August 11, 2011

The Honorable Edward J. Markey
United States House of Representatives
Washington, D.C. 20515

Dear Congressman Markey:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter of April 15, 2011, regarding NRC actions being taken in response to the recent events in Japan. Answers to your specific questions are included as an enclosure to this letter. Please note that documents in this submittal have not been released to the public and have been marked “Not For Public Disclosure.” I respectfully ask that you honor these markings.

If you have any additional questions, please contact me or Ms. Rebecca Schmidt, Director of the Office of Congressional Affairs, at (301) 415-1776.

Sincerely,

/RA/

Gregory B. Jaczko

Enclosures:
As stated

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.

TEMPORARY INSTRUCTION 2515/183

FOLLOWUP TO THE FUKUSHIMA DAIICHI NUCLEAR STATION FUEL DAMAGE EVENT

CORNERSTONE: INITIATING EVENTS AND MITIGATING SYSTEMS

APPLICABILITY: This Temporary Instruction (TI) applies to all holders of operating licenses for nuclear power reactors, except plants which have permanently ceased operations.

2515/183-01 OBJECTIVES

The objective of this TI is to independently assess the adequacy of actions taken by licensees in response to the Fukushima Daiichi nuclear station fuel damage event. The inspection results from this TI will be used to evaluate the industry's readiness for a similar event and to aid in determining whether additional regulatory actions by the U.S. Nuclear Regulatory Commission are warranted. Therefore, the intent of this TI is to be a high-level look at the industry's preparedness for events that may exceed the design basis for a plant. If necessary, a more specific followup inspection will be performed at a later date.

2515/183-02 BACKGROUND

On March 11, 2011, the Tohoku-Taiheiyou-Oki Earthquake occurred near the east coast of Honshu, Japan. This magnitude 9.0 earthquake and the subsequent tsunami caused significant damage to at least four of the six units of the Fukushima Daiichi nuclear power station as the result of a sustained loss of both the offsite and on-site power systems. Efforts to restore power to emergency equipment have been hampered or impeded by damage to the surrounding areas due to the tsunami and earthquake. The following background information is current as of March 18, 2011.

Units 1 through 3, which had been operating at the time of the earthquake, scrammed automatically, inserting their neutron absorbing control rods to ensure immediate shutdown of the fission process. Following the loss of electric power to normal and emergency core cooling systems and the subsequent failure of back-up decay heat removal systems, water injection into the cores of all three reactors was compromised, and reactor water levels could not be maintained. Tokyo Electric Power Company (TEPCO), the operator of the plant, resorted to injecting sea water and boric acid into the reactor vessels of these three units, in an effort to cool the fuel and ensure the reactors remained shutdown. However, the fuel in the reactor cores became partially uncovered. Hydrogen gas built up in Units 1 and 3 as a result of exposed, overheated fuel reacting with water. Following gas venting from the primary containment to relieve pressure, hydrogen explosions occurred in both units and damaged the secondary containments. It appears that primary containments for Units 1 and 3 remained functional, but the primary containment for Unit 2 may have been damaged. TEPCO cut a hole in the side of the Unit 2 secondary containment to prevent hydrogen buildup following a sustained period when there was no water injection into the core.

In addition, problems were encountered with monitoring and maintaining Units 3 and 4 spent fuel pool (SFP) water levels. Efforts continue to supply seawater to the SFPs for Units 1 through 4 using various methods. At this time, the integrity of the SFPs for Units 3 and 4 is unknown.

Fukushima Daiichi Units 4 through 6 were shutdown for refueling outages at the time of the earthquake. The fuel assemblies for Unit 4 had been offloaded from the reactor core to the SFP. The SFPs for Units 5 and 6 appear to be intact.

The damage to Fukushima Daiichi nuclear power station appears to have been caused by initiating events that may have exceeded the design basis for the facilities.

2515/183-03 INSPECTION REQUIREMENTS AND GUIDANCE

NRC inspection staff should assess the licensee's activities and actions to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. These inspections should occur at the operating power reactor facilities. Licensee emergency preparedness will not be assessed by this TI.

This TI may be completed all at once or in phases as the licensee verifies its capability to respond to such an event. The inspector(s) should coordinate the inspection effort with the licensee in accordance with the licensee's verification schedule.

The events at the Fukushima Daiichi plant appear to be caused by factors directly impacting nuclear safety that may have exceeded the design basis for the facility. While details on the full extent of damage to these units remain unknown, the damage poses a significant challenge to the nuclear safety of these units. Immediate actions by the U.S. industry are appropriate to assess and take corrective actions to address potential vulnerabilities that would challenge response to events that are beyond site design bases.

03.01 Assess the 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 issued February 25, 2002, and severe accident management guidelines and as required by Title 10 of the Code of Federal Regulations (10 CFR) 50.54(hh). Use Inspection Procedure (IP) 71111.05T, "Fire Protection (Triennial)," Section 02.03 and 03.03 as a guideline. If IP 71111.05T was recently performed at the facility the inspector should review the inspection results and findings to identify any other potential areas of inspection. Particular emphasis should be placed on strategies related to the spent fuel pool. The inspection should include, but not be limited to, an assessment of any licensee actions to:

a. Verify through test or inspection that equipment is available and functional. Active equipment shall be tested and passive equipment shall be walked down and inspected. It is not expected that permanently installed equipment that is tested under an existing regulatory testing program be retested.

b. Verify through walkdowns or demonstration that procedures to implement the strategies associated with B.5.b and 10 CFR 50.54(hh) are in place and are executable. Licensees may choose not to connect or operate permanently installed equipment during this verification.

c. Verify the training and qualifications of operators and the support staff needed to implement the procedures and work instructions are current for activities related to Security Order Section B.5.b and severe accident management guidelines as required by 10 CFR 50.54 (hh).

d. Verify that any applicable agreements and contracts are in place and are capable of meeting the conditions needed to mitigate the consequences of these events.

e. Review any open corrective action documents to identify vulnerabilities that may not have yet been addressed.

03.02 Assess the licensee's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63, "Loss of All Alternating Current Power," and station design, is functional and valid. Refer to TI 2515/120, "Inspection of Implementation of Station Blackout Rule Multi-Plant Action Item A-22" as a guideline. It is not intended that TI 2515/120 be completely reinspected. The inspection should include, but not be limited to, an assessment of any licensee actions to:

a. Verify through walkdowns and inspection that all required materials are adequate and properly staged, tested, and maintained.

b. Demonstrate through walkdowns that procedures for response to an SBO are executable.

03.03 Assess the licensee's capability to mitigate internal and external flooding events required by station design. Refer to IP 71111.01, "Adverse Weather Protection," Section 02.04, "Evaluate Readiness to Cope with External Flooding" as a guideline. The inspection should include, but not be limited to, an assessment of any licensee actions to verify through walkdowns and inspections that all required materials and equipment are adequate and properly staged. These walkdowns and inspections shall include verification that accessible doors, barriers, and penetration seals are functional.

03.04 Assess the thoroughness of the licensee's walkdowns 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. Assess the licensee's development of any new mitigating strategies for identified vulnerabilities (e.g., entered it in to the corrective action program and any immediate actions taken). As a minimum, the licensee should have performed walkdowns and inspections of important equipment (permanent and temporary) such as storage tanks, plant water intake structures, and fire and flood response equipment; and developed mitigating strategies to cope with the loss of that important function. Use IP 71111.21, "Component Design Basis Inspection," Appendix 3, "Component Walkdown Considerations," as a guideline to assess the thoroughness of the licensee's walkdowns and inspections.

2515/183-04 REPORTING REQUIREMENTS

The inspection results, including both observations and findings, of this TI should be in a stand-alone report. NOTE: This TI will be updated with a template which will provide specific guidance on reporting and documenting observations and findings.

The inspection report containing the results should be forwarded to NRR/DIRS/IRIB, Attention: Tim Kobetz via e-mail at timothy.kobetz@nrc.gov. Mr. Kobetz can also be reached at (301) 415-1932. The inspection results from this TI will be used to evaluate industry's readiness for a similar event and to aid in determining whether additional NRC regulatory actions are warranted.

2515/183-05 COMPLETION SCHEDULE

This TI is to be initiated upon issuance. Inspection activities are to be completed by April 29, 2011 and the inspection report issued by May 13, 2011.

2515/183-06 EXPIRATION

The TI will expire on June 30, 2012.

2515/183-07 CONTACT

Any technical questions regarding this TI should be addressed to Tim Kobetz at 301-415-1932 or timothy.kobetz@nrc.gov.

2515/183-08 STATISTICAL DATA REPORTING

All direct inspection effort expended on this TI is to be charged to 2515/183 with an IPE code of TI. All indirect inspection effort expended on this TI for preparation and documentation should be attributed to activity codes TIP and TID respectively.

2515/183-9 RESOURCE ESTIMATE

The estimated average time to complete the TI inspection requirements is 40 hours per site. Where applicable, inspectors should credit the baseline inspection program for samples reviewed during this TI assessment.

2515/183-10 TRAINING

No additional training is required.

END

ATTACHMENT 1

Revision History for TI 2515/183
FOLLOWUP TO FUKUSHIMA DAIICHI NUCLEAR STATION FUEL DAMAGE EVENT

Commitment Tracking Number	Issue Date	Description of Change	Training Needed	Training Completion Date	Comment Resolution Accession Number
N/A	ML11077A007 03/23/11	Researched commitments for 4 years and found none. This is a new document issued for inspections related to the industry response to the Fukushima Daiichi Nuclear Station Fuel Damage Event.	No	N/A	N/A

Summary of Observations

Temporary Instruction 2515/183, "Followup to the Fukushima Daiichi Fuel Damage Event"

Summary of Observations:

The following are some general observations made during the performance of TI 2515/183. While individually, none of these observations posed a significant safety issue, they indicate a potental industry trend of failure to maintain equipment and strategies required to mitigate some design and beyond design basis events.

Nuclear plants have multiple, redundant, strategies for which the overall function is to mitigate damage to the facility's fuel elements and containment. The failure of a strategy due to equipment failure, procedure inadequacy, inadequate training, etc., does not mean that the other redundant strategies would not have successfully performed their function. During this inspection, while some deficiencies were identified that would have caused a single strategy to be compromised or fail, no functions were compromised that would have resulted in damage to the fuel elements or containment.

The results of the inspections are being assessed in greater detail through the NRC's Reactor Oversight Process and will also be examined by the NRC's Task Force's examining the agency's regulatory requirements, programs, processes, and implementation in light of information from the Fukushima Daiichi event.

Licensee Capability to Mitigate Fires in Large Areas of the Plant in accordance with 10 CFR 50.54(hh)(2)

- Some equipment (mainly pumps) would not operate when tested or lacked test acceptance criteria
- Some equipment was missing or dedicated to other plant operations
- In some cases plant modifications had rendered strategies unworkable
- Fuel for pumps was not always readily available

Licensee Capability to Mitigate Station Blackout (SBO) Conditions

- In a few cases procedural or training deficiencies existed.

Licensee Capability to Mitigate Design Basis Internal and External Events

- Some equipment (mainly pumps) would not operate when tested or lacked test acceptance criteria
- Some discrepancies were identified with barrier and penetration seals

Licensee Capability to Respond to Beyond Basis Events involving Fires, Floods, and Seismic Events

- Some equipment to mitigate fires and SBO was stored in areas that were not seismically qualified or could be flooded

Matrix of Observations by Facility:

The attached matrix is a per-site summary of observations associated with TI 2515/183. The matrix was developed to provide NRC and external stakeholders with a quick method to review the observations, however, the matrix is not an in-depth assessment of the findings. As noted above, NRC is currently performing a thorough assessment of the identified issues to provide to the Task Force with insights on the U.S. nuclear industry's readiness to cope with beyond design basis events.

Using the Matrix:

While the TI was not designed to ask "yes/no" questions, the matrix provides basic answers in this way to help guide the user to information regarding a facility that they may be interested in. The inspection reports should be reviewed for additional information on the observation.

Plant / Site	Region	Section 03.01: B.5.b/50.54(hh)(2) Were the licensee's mitigation capabilities[1] satisfactory at the time of the initial review?	Section 03.02: Station Blackout (SBO) Were the licensee's mitigation capabilities[1] satisfactory at the time of the initial review?	Section 03.03: Flooding Were the licensee's mitigation capabilities[1] satisfactory at the time of the initial review?	Section 03.04: Seismic activity coincident with large fires or flooding Was the licensee's review and assessment satisfactory?
Beaver Valley	I	YES	YES	YES	YES
Calvert Cliffs	I	YES	YES	YES	YES
Fitzpatrick	I	YES	YES	YES	YES
Ginna	I	YES	YES	YES	YES
Hope Creek	I	YES	YES	YES	YES
Indian Point 2	I	YES	YES	YES	YES
Indian Point 3	I	YES	YES	YES	YES
Limerick	I	YES	YES	YES	YES
Millstone	I	YES	YES	YES	YES
Nine Mile Point	I	YES	YES	YES	YES
Oyster Creek	I	Function Met; Strategy Not Demonstrated (corrected)	YES[2]	YES	YES
Peach Bottom	I	YES	YES	YES	YES
Pilgrim	I	Function Met; Strategy Not Demonstrated (resolution in progress)[2]	YES	YES	YES
Salem	I	YES	YES	YES	YES
Seabrook	I	YES	YES	YES	YES
Susquehanna	I	YES	YES	YES	YES
Three Mile Island	I	YES	YES	YES	YES
Vermont Yankee	I	YES	YES	YES	YES
Browns Ferry	II	YES	YES[2]	YES	YES
Brunswick	II	YES	YES	YES	YES
Catawba	II	YES	YES	YES	YES
Crystal River	II	YES	YES	YES	YES
Farley	II	YES	YES	YES	YES
Harris	II	YES	YES	YES	YES
Hatch	II	Function Met; Strategy Not Demonstrated (resolution in progress)[2]	YES	YES	YES
McGuire	II	YES	YES	YES	YES
North Anna	II	YES	YES	YES[2]	YES
Oconee	II	Function Met; Strategy Not Demonstrated (resolution in progress)[2]	YES	YES[2]	YES
Robinson	II	YES	YES	YES	YES
Sequoyah	II	YES[2]	YES	YES	YES
St. Lucie	II	YES	YES	YES	YES
Summer	II	YES	YES	YES	YES
Surry	II	YES[2]	YES	YES	YES
Turkey Point	II	YES	YES	YES	YES
Vogtle	II	YES	YES	YES	YES
Watts Bar	II	YES[2]	YES	YES	YES
Braidwood	III	Function Met; Strategy Not Demonstrated (resolution in progress)[2]	YES	YES	YES
Byron	III	YES	YES	YES[2]	YES
Clinton	III	Function Met; Strategy Not Demonstrated (resolution in progress)	YES	YES	YES
D.C. Cook	III	YES	YES	YES	YES
Davis-Besse	III	YES	YES	YES	YES
Dresden	III	YES	YES	YES	YES
Duane Arnold	III	YES	YES	YES	YES
Fermi	III	YES	YES	YES	YES
Kewaunee	III	Function Met; Strategy Not Demonstrated (resolution in progress)	YES	YES	YES
LaSalle	III	YES	YES[2]	Function Met; Strategy Not Demonstrated (resolution in progress)	YES
Monticello	III	YES	YES	YES	YES
Palisades	III	YES	YES	YES	YES
Perry	III	YES	YES	YES	YES
Point Beach	III	YES	YES	Function Met; Strategy Not Demonstrated (resolution in progress)[2]	YES
Prairie Island	III	YES	YES	YES	YES
Quad Cities	III	YES	YES	YES	YES
Arkansas Nuclear	IV	YES	YES	YES	YES
Callaway	IV	YES	YES	YES	YES
Columbia	IV	YES	YES	YES	YES
Comanche Peak	IV	YES	YES[2]	YES[2]	YES
Cooper	IV	YES	YES	YES	YES
Diablo Canyon	IV	YES[2]	YES[2]	YES	YES
Fort Calhoun	IV	YES	YES[2]	YES[2]	YES
Grand Gulf	IV	YES[2]	YES[2]	YES	YES
Palo Verde	IV	YES	YES	YES	YES
River Bend	IV	Function Met; Strategy Not Demonstrated (corrected)[2]	Function Met; Strategy Not Demonstrated (corrected)	YES	YES
San Onofre	IV	YES	YES	YES	YES
South Texas	IV	YES[2]	YES	YES	YES
Waterford	IV	YES[2]	Function Met; Strategy Not Demonstrated (corrected)[2]	YES	YES
Wolf Creek	IV	Function Met; Strategy Not Demonstrated (resolution in progress)	Function Met; Strategy Not Demonstrated (resolution in progress)	YES	YES

1. There are multiple and redundant mitigation strategies. A "Function Met; Strategy Not Demonstrated" response in this column means that a mitigation strategy was not demonstrated, however, the overall mitigation function would still have worked and protected against fuel and contaiment damage.

2. The response for this item was determined through both a review of the inspection report and ongoing evaluations.

May 13, 2011

Mr. A. Lincoln, President
Great Lakes Electric Company, LLC
4300 Lostfield Road
Anywhere, IL 60555

SUBJECT: (Plant) – NRC TEMPORARY INSTRUCTION 2515/183 INSPECTION REPORT (Report number)

Dear Mr. Lincoln:

On April 29, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your (name of facility), using Temporary Instruction 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event.". The enclosed inspection report documents the inspection results which were discussed on April xx, 2011, with Mr. xxxxx and other members of your staff.

The objective of this inspection was to promptly assess the capabilities of (plant name) to respond to extraordinary consequences similar to those that have recently occurred at the Japanese Fukushima Diaichi Nuclear Station. The results from this inspection, along with the results from this inspection performed at other operating commercial nuclear plants in the United States will be used to evaluate the U.S. nuclear industry's readiness to safely respond to similar events. These results will also help the NRC to determine if additional regulatory actions are warranted.

All of the potential issues and observations identified by this inspection are contained in this report. The NRC's Reactor Oversight Process will further evaluate any issues to determine if they are regulatory findings or violations. Any resulting findings or violations will be documented by the NRC in a separate report (or you can state " the next quarterly report"). You are not required to respond to this letter.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

,
Branch Chief
Division of Reactor Projects

Docket Nos.
License Nos.

Enclosure: Inspection Report xxxxxxx

cc w/encl:

Enclosure

U. S. NUCLEAR REGULATORY COMMISSION

REGION X

Docket(s):	05000XXX
License:	NPF-XX
Report:	05000XXX/2011XXX
Licensee:	
Facility:	
Location:	
Dates:	March 23, 2011 through April 29, 2011
Inspectors:	, Senior Resident Inspector , Resident Inspector
Approved By:	, Chief, Project Branch Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000xxx/20110xx, 03/23/2011 – 04/29/2011; xxxxxxx{plant name} Temporary Instruction 2515/183 - Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event

This report covers an announced Temporary Instruction inspection. The inspection was conducted by Resident and Region xx inspectors. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006."

INSPECTION SCOPE

The intent of the TI is to provide a broad overview of the industry's preparedness for events that may exceed the current design basis for a plant. The focus of the TI was on (1) assessing the licensee's capability to mitigate consequences from large fires or explosions on site, (2) assessing the licensee's capability to mitigate station blackout (SBO) conditions, (3) assessing the licensee's capability to mitigate internal and external flooding events accounted for by the station's design, and (4) assessing the thoroughness of the 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. If necessary, a more specific follow-up inspection will be performed at a later date.

INSPECTION RESULTS

All of the potential issues and observations identified by this inspection are contained in this report. The NRC's Reactor Oversight Process will further evaluate any issues to determine if they are regulatory findings or violations. Any resulting findings or violations will be documented by the NRC in a separate report (..or you can state the next quarterly report).

03.01 Assess the 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 issued February 25, 2002, and severe accident management guidelines and as required by Title 10 of the Code of Federal Regulations (10 CFR) 50.54(hh). Use Inspection Procedure (IP) 71111.05T, “Fire Protection (Triennial),” Section 02.03 and 03.03 as a guideline. If IP 71111.05T was recently performed at the facility the inspector should review the inspection results and findings to identify any other potential areas of inspection. Particular emphasis should be placed on strategies related to the spent fuel pool. The inspection should include, but not be limited to, an assessment of any licensee actions to:

<table>
<tr><td>Licensee Action</td><td></td><td>Describe what the licensee did to test or inspect equipment.</td></tr>
<tr><td rowspan="5">a. Verify through test or inspection that equipment is available and functional. Active equipment shall be tested and passive equipment shall be walked down and inspected. It is not expected that permanently installed equipment that is tested under an existing regulatory testing program be retested.

This review should be done for a reasonable sample of mitigating strategies/equipment.</td><td rowspan="5"></td><td></td></tr>
<tr><td>Describe inspector actions taken to confirm equipment readiness (e.g., observed a test, reviewed test results, discussed actions, reviewed records, etc.).</td></tr>
<tr><td></td></tr>
<tr><td>Discuss general results including corrective actions by licensee.</td></tr>
<tr><td></td></tr>
</table>

Licensee Action		Describe the licensee’s actions to verify that procedures are in place and can be executed (e.g. walkdowns, demonstrations, tests, etc.)

b. Verify through walkdowns or demonstration that procedures to implement the strategies associated with B.5.b and 10 CFR 50.54(hh) are in place and are executable. Licensees may choose not to connect or operate permanently installed equipment during this verification. This review should be done for a reasonable sample of mitigating strategies/equipment.		
		Describe inspector actions and the sample strategies reviewed. Assess whether procedures were in place and could be used as intended.
		Discuss general results including corrective actions by licensee.

Licensee Action		Describe the licensee's actions and conclusions regarding training and qualifications of operators and support staff.
c. Verify the training and qualifications of operators and the support staff needed to implement the procedures and work instructions are current for activities related to Security Order Section B.5.b and severe accident management guidelines as required by 10 CFR 50.54 (hh).		 Describe inspector actions and the sample strategies reviewed to assess training and qualifications of operators and support staff

		Discuss general results including corrective actions by licensee.

Licensee Action		Describe the licensee's actions and conclusions regarding applicable agreements and contracts are in place.
d. Verify that any applicable agreements and contracts are in place and are capable of meeting the conditions needed to mitigate the consequences of these events. This review should be done for a reasonable sample of mitigating strategies/equipment.		
		For a sample of mitigating strategies involving contracts or agreements with offsite entities, describe inspector actions to confirm agreements and contracts are in place and current (e.g., confirm that offsite fire assistance agreement is in place and current).
		Discuss general results including corrective actions by licensee.

Licensee Action		Document the corrective action report number and briefly summarize problems noted by the licensee that have significant potential to prevent the success of any existing mitigating strategy.
e. Review any open corrective action documents to assess problems with mitigating strategy implementation identified by the licensee. Assess the impact of the problem on the mitigating capability and the remaining capability that is not impacted.		

03.02 Assess the licensee's capability to mitigate station blackout (SBO) conditions, as required by 10 CFR 50.63, "Loss of All Alternating Current Power," and station design, is functional and valid. Refer to TI 2515/120, "Inspection of Implementation of Station Blackout Rule Multi-Plant Action Item A-22" as a guideline. It is not intended that TI 2515/120 be completely reinspected. The inspection should include, but not be limited to, an assessment of any licensee actions to:

Licensee Action		Describe the licensee's actions to verify the adequacy of equipment needed to mitigate an SBO event.
a. Verify through walkdowns and inspection that all required materials are adequate and properly staged, tested, and maintained.		
		Describe inspector actions to verify equipment is available and useable.
		Discuss general results including corrective actions by licensee.

Licensee Action		Describe the licensee's actions to verify the capability to mitigate an SBO event.
b. Demonstrate through walkdowns that procedures for response to an SBO are executable.		
		Describe inspector actions to assess whether procedures were in place and could be used as intended.

		Discuss general results including corrective actions by licensee.

03.03 Assess the licensee's capability to mitigate internal and external flooding events required by station design. Refer to IP 71111.01, "Adverse Weather Protection," Section 02.04, "Evaluate Readiness to Cope with External Flooding" as a guideline. The inspection should include, but not be limited to, an assessment of any licensee actions to verify through walkdowns and inspections that all required materials and equipment are adequate and properly staged. These walkdowns and inspections shall include verification that accessible doors, barriers, and penetration seals are functional.		
Licensee Action		Describe the licensee's actions to verify the capability to mitigate existing design basis flooding events.
a. Verify through walkdowns and inspection that all required materials are adequate and properly staged, tested, and maintained.		
		Describe inspector actions to verify equipment is available and useable. Assess whether procedures were in place and could be used as intended.
		Discuss general results including corrective actions by

		licensee.

03.04 Assess the thoroughness of the licensee’s walkdowns 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. Assess the licensee’s development of any new mitigating strategies for identified vulnerabilities (e.g., entered it in to the corrective action program and any immediate actions taken). As a minimum, the licensee should have performed walkdowns and inspections of important equipment (permanent and temporary) such as storage tanks, plant water intake structures, and fire and flood response equipment; and developed mitigating strategies to cope with the loss of that important function. Use IP 71111.21, “Component Design Basis Inspection,” Appendix 3, “Component Walkdown Considerations,” as a guideline to assess the thoroughness of the licensee’s walkdowns and inspections.		
Licensee Action		Describe the licensee’s actions to assess the potential impact of seismic events on the availability of equipment used in fire and flooding mitigation strategies.
a. Verify through walkdowns that all required materials are adequate and properly staged, tested, and maintained.		
		Describe inspector actions to verify equipment is available and useable. Assess whether procedures were in place and could be used as intended.
		Discuss general results including corrective actions by licensee. Briefly summarize any new mitigating strategies identified by the licensee as a result of their reviews.

Meetings

1. Exit Meeting (IMC 0612, Section 14.07: Do not include characterization of licensee response. For contested violations, refer to the Enforcement Manual for proper handling.)

   The inspectors presented the inspection results to Mr. J. DuPage and other members of licensee management at the conclusion of the inspection on January 4, 2009. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee (should include those individuals who provided information with respect to findings. Discretion is advised as this list should not be pages long. Order can vary. This Branch Chief prefers SVP, PM then alphabetical.)

Nuclear Regulatory Commission

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety but rather that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

03.01 Assess the licensee's capability to mitigate conditions that result from beyond design basis events

Number	Description or Title	Date or Revision

03.02 Assess the licensee's capability to mitigate station blackout (SBO) conditions

Number	Description or Title	Date or Revision

03.03 Assess the licensee's capability to mitigate internal and external flooding events required by station design

Number	Description or Title	Date or Revision

03.04 Assess the thoroughness of the licensee's walkdowns 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

Number	Description or Title	Date or Revision

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents Access and Management System
AF	Auxiliary Feedwater
ARM	Area Radiation Monitors
CAM	Continuous Air Monitors
CC	Component Cooling Water
CFR	Code of Federal Regulations
NRC	United States Nuclear Regulatory Commission

TEMPORARY INSTRUCTION 2515/184

AVAILABILITY AND READINESS INSPECTION OF SEVERE ACCIDENT MANAGEMENT GUIDELINES (SAMGs)

CORNERSTONE: MITIGATING SYSTEMS

APPLICABILITY: This Temporary Instruction (TI) applies to all holders of operating licenses for nuclear power reactors, except plants which have permanently ceased operations.

2515/184-01 OBJECTIVES

The objectives of this TI are to:

a. Determine that the severe accident management guidelines (SAMGs) are available and how they are being maintained.

b. Determine the nature and extent of licensee implementation of SAMG training and exercises.

2515/184-02 BACKGROUND

On March 30, 2011, the Executive Director for Operations chartered a task force to conduct a near-term evaluation of the need for agency actions following the events in Japan. During the task force's deliberations, the importance of severe accident management guidelines (SAMGs) has been highlighted. The SAMGs were implemented as a voluntary industry initiative in the 1990s and are not part of the agency's routine Reactor Oversight Program. In order to evaluate the current status of SAMGs onsite and determine the need for any further recommendations, the task force is requesting the enclosed information regarding SAMGs at operating power reactors be gathered, assessed, and summarized.

2515/184-03 INSPECTION REQUIREMENTS AND GUIDANCE

03.01 Assess the availability and readiness of the licensee's ability to access and implement the SAMGs at their facility. Answer the following questions by filling out the attached datasheet.

a. When were the SAMGs last updated? Are controlled copies of the SAMG located in the technical support center (TSC) (Y/N), emergency operations facility (EOF) (Y/N), control room (Y/N)? For licensees that use one common EOF for multiple reactor sites, one review of the EOF will serve for all applicable sites.

b. Are SAMGs covered by the licensee's procedure control and document management system, including the requirements for periodic review and revision (Y/N)?

c. Does the licensee's configuration control and change management systems (e.g., 10CFR50.59 process) cause the licensee to update SAMGs to reflect design changes (Y/N/Partially – describe)?

d. Perform a high-level comparison of the site's SAMGs with available industry guidance (e.g., owner's group guidance document and other industry standards as applicable). Are the SAMGs consistent with the owners group guidance (Y/N/comments)?

   1. A high-level comparison means that the inspector should determine whether the major sections of the guidance documents are covered. It is not meant to be a step-by-step review of the SAMGs.

   2. The owners group guidance documents were normally the basis for the development of the SAMGs, however, other industry standards may also have been used.

   3. Some variations from the guidance documents may have been made to accommodate site specific plant design differences.

   4. The inspectors should not access the adequacy of the SAMGs as it is beyond the scope of this TI.

e. How is training conducted on the SAMGs? Who is trained on the SAMGs? What is the periodicity of training?

   1. There are various training methods that may be used by the licensee (e.g., table top exercise, classroom training, reading of training materials).

   2. Whichever training method is used the licensee should be able to provide documentation (e.g., training records) that the training was completed.

f. Interview 4 licensed operators (2 reactor operators and 2 senior reactor operators (shift technical advisor may be substituted for one of the 4 operators), 2 TSC staff, and 2 TSC managers designated to implement the SAMGs during an emergency to determine: (1) did they receive initial (Y/N) and periodic (Y/N/document periodicity) training on the SAMGs and how they relate to their assigned duties, and (2) can they articulate their responsibilities with respect to the use of SAMGs (Y/N/document who would actually implement the licensee's SAMGs)?

g. Have there been periodic exercises on the use of SAMGs by individuals who would implement them during an emergency (Y/N/document periodicity)?

2515/184-02 REPORTING REQUIREMENTS

The inspection results of this TI should be forwarded by each Region via memorandum to NRR/DIRS/IRIB, Attention: Tim Kobetz. The memorandum should be sent via e-mail to timothy.kobetz@nrc.gov no later than May 27, 2011. Mr. Kobetz can also be reached at (301) 415-1932. The memorandum should be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

In addition, the next quarterly inspection report should document completion of the TI and reference the ADAMS accession number. The inspection results from this TI will be used to aid in determining whether additional NRC regulatory actions are warranted in this area.

2515/184-03 COMPLETION SCHEDULE

This TI is to be initiated upon issuance and completed by May 27, 2011.

2515/184-04 EXPIRATION

The TI will expire on June 30, 2012.

2515/184-05 CONTACT

Any technical questions regarding this TI should be addressed to Tim Kobetz at 301-415-1932 or timothy.kobetz@nrc.gov.

2515/184-08 STATISTICAL DATA REPORTING

All direct inspection effort expended on this TI is to be charged to 2515/184 with an IPE code of TI. All indirect inspection effort expended on this TI for preparation and documentation should be attributed to activity codes TIP and TID respectively.

2515/184-09 RESOURCE ESTIMATE

The estimated average time to complete the TI inspection requirements will be 16-20 hours per site.

END

ATTACHMENTS: Exhibit 1: Template for Inspection Results for TI 2515/184
Attachment 1: Revision History Page

EXHIBIT 1

TABLE OF RESULTS

Letter or Number	Inspection Item	Yes	No	Response/Comments
a	When were the SAMGs last updated?			
	Are controlled copies of the SAMG located in the technical support center (TSC)? (Y/N)			
	Are controlled copies of the SAMG located in the emergency operations facility (EOF)? (Y/N)			
	Are controlled copies of the SAMG located in the control room? (Y/N)			
b	Are SAMGs covered by the licensee's procedure control and document management system, including the requirements for periodic review and revision? (Y/N)			
c	Does the licensee's configuration control and change management systems cause the licensee to update SAMGs to reflect design changes? (Y/N/Partially-describe)			
d	Perform a high-level comparison of the site's SAMGs with available industry guidance (e.g., owner's group guidance document and other industry standards as applicable). Are the SAMGs consistent with the owners group guidance (if any) having been incorporated (Y/N/comments)?			
e	How is training conducted on the SAMGs? Who is trained on the SAMGs? What is the periodicity of training?			
f	Interview 4 licensed operators (2 reactor operators and 2 senior reactor operators), 2 TSC staff, and 2 TSC managers assigned to apply the SAMGs during an emergency to determine:			
	(1) Did they receive **initial** training on the SAMGs? (Y/N)			
	Did they receive **periodic** training (Y/N/document periodicity) on the SAMGs and how they relate to their assigned duties?			
	(2) Can they articulate their responsibilities with respect to the use of SAMGs (Y/N/document who would actually implement the licensee's SAMGs)?			
g	Have there been periodic exercises on the use of SAMGs by individuals who would implement them during an emergency (Y/N/document periodicity)?			

ATTACHMENT 1

Revision History for TI 2515/184
Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)

Commitment Tracking Number	Issue Date	Description of Change	Training Needed	Training Completion Date	Comment Resolution Accession Number
N/A	ML11115A053 04/29/11 CN 11-008	Researched commitments for 4 years and found none. This is a new document issued for inspections related to the NRC's followup to the Fukushima Daiichi Nuclear Event.	No	N/A	N/A

Summary of Observations
Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)"

Summary of Observations:

The SAMGs were implemented as a voluntary industry initiative in the 1990s and are not part of the agency's routine Reactor Oversight Program. On March 30, 2011, the Executive Director for Operations chartered a task force to conduct an evaluation of the need for agency actions following the events in Japan. The task force will evaluate the current status of SAMGs at operating power reactors and determine the need for any further agency actions.

The following are some general observations made during the performance of TI 2515/184. While individually, none of these observations posed a significant safety issue, they indicate that while the SAMG procedures are available at every site, there appears to be an inconsistent implementation of some aspects of this voluntary SAMG program.

SAMGs are typically available in plant locations critical to combating a potentially severe accident. However in some cases the procedures were either not available in all expected areas or not properly controlled. In addition, while SAMGs appear to be updated to reflect design changes at a facility, there does not appear to be a consistent approach to conducting periodic reviews. Finally, while personnel do appear to be properly trained and knowledgeable on SAMGs, exercises on SAMGs do not appear to be periodically conducted at all sites. This summary is being provided to the Near Term Task Force for use during its review of agency regulatory requirements in light of the Japan Fukushima event.

Matrix of Observations by Facility:

The attached matrix is a per-site summary of observations associated with TI 2515/184. The matrix was developed to provide NRC and external stakeholders with a quick method to review the observations; however, the matrix is not an in-depth assessment of the findings. As noted above, NRC is evaluating the current status of SAMGs at operating power reactors in order to determine the need for any further agency actions.

Using the Matrix:

The matrix provides basic answers in a way to help guide the user to information regarding a facility that they may be interested in. The first column under Item 03.01(a) indicates that all licensees have SAMG procedures. The inspection reports should be reviewed for additional information on when the SAMGs were last updated. Specific answers to training questions associated with Inspection Item 03.01(e) are not included in the matrix; however an overview of the training program is included in the matrix under Item 03.01(f). The inspection reports should be reviewed for additional information on the observations.

Summary of Inspection Results for TI 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)

		03.01(a)				03.01(b)	03.01(c)	03.01(d)	03.01(f) (see footnote 2)			03.01(g)
Plant / Site	Region	Does the licensee have SAMG procedures? (see footnote 1)	Are controlled copies of the SAMG located in the technical support center (TSC)? (Y/N)	Are controlled copies of the SAMG located in the emergency operations facility (EOF)? (Y/N)	Are controlled copies of the SAMG located in the control room? (Y/N)	Are SAMGs covered by the licensee's procedure control and document management system, including the requirements for periodic review and revision? (Y/N)	Does the licensee's configuration control and change management systems (e.g., 10CFR50.59 process) cause the licensee to update SAMGs to reflect design changes? (Y/N/Partially)	Are the SAMGs consistent with the owners group guidance (if any) having been incorporated? (Y/N)	Did personnel receive initial training on the SAMGs? (Y/N)	Did personnel receive periodic training on the SAMGs and how they relate to their assigned duties? (Y/N)	Can personnel articulate their responsibilities with respect to the use of SAMGs? (Y/N)	Have there been periodic exercises on the use of SAMGs by individuals who would implement them during an emergency? (Y/N)
Beaver Valley	I	Y	Y	N	Y	Y	Y	Y	Y	N	N	N
Calvert Cliffs	I	Y	Y	Y	Y	Y	Y	Y	Y	N	Y	Y
Fitzpatrick	I	Y	Y	Y	Y	N	Y	Y	N	N	N	N
Ginna	I	Y	Y	N	Y	Y	N	Y	Y	Y	Y	N
Hope Creek	I	Y	Y	N	Y	Y	Y	Y	Y	Y	Y	Y
Indian Point 2	I	Y	Y	N	Y	N	Y	Y	Y	Y	Y	Y
Indian Point 3	I	Y	Y	N	Y	N	Y	Y	Y	Y	Y	Y
Limerick	I	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	Y
Millstone	I	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	N
Nine Mile Point	I	Y	Y	Y	Y	N	Y	Y	N	N	Y	Y
Oyster Creek	I	Y	Y	N	Y	N	Y	Y	Y	Y	Y	Y
Peach Bottom	I	Y	Y	Y	Y	N	Y	Y	Y	Y	N	Y
Pilgrim	I	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	N
Salem	I	Y	Y	N	Y	Y	Y	Y	Y	Y	Y	Y
Seabrook	I	Y	Y	Y	Y	Y	N	Y	Y	Y	Y	N
Susquehanna	I	Y	Y	Y	N	Y	Y	Y	Y	N	Y	N
Three Mile Island	I	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	Y
Vermont Yankee	I	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	Y
Browns Ferry	II	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	Y
Brunswick	II	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y
Catawba	II	Y	Y	Y	Y	N	N	Y	Y	Y	Y	Y
Crystal River	II	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	N
Farley	II	Y	Y	N	Y	N	Y	Y	Y	Y	Y	Y
Harris	II	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	N
Hatch	II	Y	Y	N	Y	Y	Y	Y	Y	Y	Y	N
McGuire	II	Y	Y	Y	Y	N	N	Y	Y	Y	Y	Y
North Anna	II	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	N
Oconee	II	Y	Y	N	N	N	N	Y	Y	Y	Y	Y
Robinson	II	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	Y
Sequoyah	II	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	Y
St. Lucie	II	Y	Y	Y	N	Y	Y	Y	Y	Y	Y	Y
Summer	II	Y	Y	Y	Y	Y	N	N	Y	Y	Y	N
Surry	II	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	Y
Turkey Point	II	Y	Y	Y	Y	N	N	Y	Y	Y	Y	N
Vogtle	II	Y	Y	N	Y	Y	Y	Y	Y	Y	Y	Y
Watts Bar	II	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y
Braidwood	III	Y	Y	N	Y	N	Y	N	Y	Y	Y	Y
Byron	III	Y	Y	N	Y	N	Y	N	Y	Y	Y	N
Clinton	III	Y	Y	N	Y	N	Y	Y	Y	Y	Y	Y
D.C. Cook	III	Y	Y	Y	Y	N	N	Y	Y	Y	Y	N
Davis-Besse	III	Y	Y	N	Y	Y	N	Y	Y	Y	Y	N
Dresden	III	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	N
Duane Arnold	III	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y
Fermi	III	Y	Y	N	Y	N	Y	Y	Y	N	N	N

Summary of Inspection Results for TI 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)

		03.01(a)				03.01(b)	03.01(c)	03.01(d)	03.01(f) (see footnote 2)			03.01(g)
Plant / Site	Region	Does the licensee have SAMG procedures? (see footnote 1)	Are controlled copies of the SAMG located in the technical support center (TSC)? (Y/N)	Are controlled copies of the SAMG located in the emergency operations facility (EOF)? (Y/N)	Are controlled copies of the SAMG located in the control room? (Y/N)	Are SAMGs covered by the licensee's procedure control and document management system, including the requirements for periodic review and revision? (Y/N)	Does the licensee's configuration control and change management systems (e.g., 10CFR50.59 process) cause the licensee to update SAMGs to reflect design changes? (Y/N/Partially)	Are the SAMGs consistent with the owners group guidance (if any) having been incorporated? (Y/N)	Did personnel receive initial training on the SAMGs? (Y/N)	Did personnel receive periodic training on the SAMGs and how they relate to their assigned duties? (Y/N)	Can personnel articulate their responsibilities with respect to the use of SAMGs? (Y/N)	Have there been periodic exercises on the use of SAMGs by individuals who would implement them during an emergency? (Y/N)
Kewaunee	III	Y	Y	Y	Y	N	Y	N	Y	N	Y	N
LaSalle	III	Y	Y	Y	Y	Y	Y	Y	N	N	Y	Y
Monticello	III	Y	Y	Y	Y	N	Y	Y	Y	N	N	N
Palisades	III	Y	Y	Y	N	N	N	Y	N	N	N	N
Perry	III	Y	Y	Y	Y	Y	N	Y	Y	Y	Y	N
Point Beach	III	Y	Y	Y	Y	Y	Y	N	Y	Y	Y	Y
Prairie Island	III	Y	Y	Y	Y	N	Y	N	N	N	Y	Y
Quad Cities	III	Y	Y	Y	Y	N	Y	Y	Y	Y	Y	Y
Arkansas Nuclear	IV	Y	Y	Y	Y	N	Partially	Y	Y	Y	Y	Y
Callaway	IV	Y	Y	Y	N	N	N	Y	Y	Y	Y	N
Columbia	IV	Y	Y	N	Y	Y	Y	Y	Y	Y	Y	Y
Comanche Peak	IV	Y	N	N	N	N	N	N	Y	Y	Y	N
Cooper	IV	Y	Y	Y	Y	Y	Y	Y	Y	N	Y	Y
Diablo Canyon	IV	Y	Y	Y	Y	Y	Partially	Y	Y	Y	Y	Y
Fort Calhoun	IV	Y	Y	Y	Y	N	N	Y	Y	Y	Y	Y
Grand Gulf	IV	Y	Y	Y	Y	Y	Y	Y	Y	N	Y	Y
Palo Verde	IV	Y	Y	Y	Y	Y	Y	Y	Y	N	Y	N
River Bend	IV	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y
San Onofre	IV	Y	Y	Y	Y	Y	N	Y	Y	N	Y	Y
South Texas	IV	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	N
Waterford	IV	Y	N	N	N	N	Partially	Y	Y	Y	Y	Y
Wolf Creek	IV	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y

(1) See the individual plant data sheets regarding information on "when the SAMGs were last updated" as referred to in TI 2515/184 Section 03.01(a).
(2) Specific answers to training questions associated with Inspection Item 03.01(e) are not included in the matrix, however, an overview of the training program is included in the matrix under Item 03.01(f). The

OMB Control No.: 3150-0012

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

May 11, 2011

NRC BULLETIN 2011-01: MITIGATING STRATEGIES

ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operation and have certified that fuel has been removed from the reactor vessel.

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to achieve the following objectives:

1. To require that addressees provide a comprehensive verification of their compliance with the regulatory requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(hh)(2),

2. To notify addressees about the NRC staff's need for information associated with licensee mitigating strategies under 10 CFR 50.54(hh)(2) in light of the recent events at Japan's Fukushima Daiichi facility in order 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, and

3. To require that addressees provide a written response to the NRC in accordance with 10 CFR 50.54(f).

BACKGROUND

Following the terrorist events of September 11, 2001, the readiness of NRC-regulated facilities to manage challenges to core cooling, containment and spent fuel pool cooling (SFP) following large explosions or fires was enhanced through a series of orders and imposition of license conditions. These requirements were formalized in the rulemaking of March 27, 2009, resulting in 10 CFR 50.54(hh)(2).

The NRC conducted a comprehensive inspection of the implementation of the mitigating strategies developed by licensees in 2008. Subsequently the NRC incorporated this inspectable area into the baseline reactor oversight process on a sample basis as part of the triennial fire protection inspection.

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Events at the Fukushima - Daiichi Nuclear Power Station following the March 11, 2011, earthquake and tsunami highlight the potential importance of B.5.b mitigating strategies in responding to beyond design basis events.

The Commission has established a task force to consider the need for agency actions following the events in Japan and is considering further actions that could improve operational safety.

DISCUSSION

The events in Japan highlight the importance and potential versatility of B.5.b mitigating strategies. Therefore, the NRC seeks comprehensive confirmation that licensees are maintaining equipment and strategies to satisfy 10 CFR 50.54(hh)(2).

In addition, the existing guidance on the implementation of the strategies, which was adopted by all licensees to meet the regulatory requirements for mitigating strategies, does not describe in detail the practices necessary for maintenance and testing of the equipment, training requirements, and validation of feasibility of the strategies. Based upon the information submitted by licensees in response to this bulletin, the NRC will determine if additional efforts are needed to ensure compliance with existing regulatory requirements and/or whether enhancement to the existing regulations and guidance are necessary.

APPLICABLE REGULATORY REQUIREMENTS

As a result of the terrorist events of September 11, 2001, the NRC issued EA-02-026, "Order for Interim Safeguards and Security Compensatory Measures" (the ICM Order) dated February 25, 2002. The ICM Order, which is designated as Safeguards Information (SGI), modified then-operating licenses for commercial power reactor facilities to require compliance with specified interim safeguards and security compensatory measures. Section B.5.b of the ICM Order requires licensees to adopt mitigation strategies using readily available resources to maintain or restore core cooling, containment, and SFP cooling capabilities to cope with the loss of large areas of the facility due to large fires and explosions from any cause, including beyond-design-basis aircraft impacts.

By letter dated February 25, 2005, the NRC staff provided guidance for implementing Section B.5.b of the ICM Order. This Phase 1 Guidance, designated as SGI, included best practices for mitigating losses of large areas of the plant and measures to mitigate fuel damage and minimize releases. Following issuance of the B.5.b Phase 1 Guidance, the NRC staff conducted inspections at operating reactor sites using TI 2515/164 (SGI) and subsequently TI 2515/168 (SGI) to ensure compliance with Section B.5.b of the ICM Order.

In December 2006, the Nuclear Energy Institute (NEI) issued NEI 06-12, Revision 2, "B.5.b Phase 2 & 3 Submittal Guideline," formerly designated for Official Use Only – Security Related Information (OUO-SRI), Agencywide Documents Access and Management System (ADAMS) Accession No. ML070090060. The NRC endorsed NEI 06-12, Revision 2, by letter dated December 22, 2006, as an acceptable means for developing and implementing the mitigation strategies requirement in Section B.5.b of the ICM Order. NEI 06-12, Revision 2, provides guidance for implementing a set of strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under the circumstances associated with the loss of a large area of the plant due to explosions or fire. NEI 06-12 provides guidance in the following areas:

- Adding make-up water to the SFP,
- Spraying water on the spent fuel,
- Enhanced initial command and control activities for challenges to core cooling and containment, and
- Enhanced response strategies for challenges to core cooling and containment.

The specific strategies covered in NEI 06-12, Revision 2, were developed based on the results of assessments conducted at currently licensed power reactor facilities for the purpose of enhancing plant specific mitigation capability for damage conditions caused by a large explosion or fire. These assessments identified a wide spectrum of potential plant specific strategies. NEI 06-12, Revision 2, specifies one set of strategies applicable to all pressurized-water reactors and another set applicable to all boiling-water reactors. Both sets are derived from the results of the plant specific assessments.

The B.5.b Phase 1 Guidance and NEI 06-12, Revision 2, were used by each licensee in preparing information submitted to the NRC that describes a plant specific approach to implementing mitigating strategies and supports each plant specific license condition. The NRC staff completed its review of the information submitted by each licensee, as well as information obtained during prior NRC inspections, and issued an OUO-SRI safety evaluation (SE) that documents the bases for its approval of the license condition for each facility. The SE issued for each licensee includes regulatory guidance in Section 3.0 of Appendix A, "Phase 1 Assessment," that recites the generic B.5.b Phase 1 Guidance, as clarified in TI 2515/168, in an OUO-SRI form rather than SGI.

By publishing new requirements in the Federal Register dated March 27, 2009 (74 FR 13926), the NRC amended 10 CFR Part 50, 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," and 10 CFR Part 73, "Physical Protection of Plants and Materials." This rulemaking added paragraph (i) to 10 CFR 50.34, "Contents of Applications; Technical Information," and paragraph (d) to 10 CFR 52.80 "Contents of Applications; Additional Technical Information," to require submittal of a "description and plans for implementation of the guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with the loss of large areas of the plant due to explosions or fire as required by § 50.54(hh)(2) of this chapter." This rulemaking also added 10 CFR 50.54(hh)(2) to impose mitigating strategies requirements similar to those imposed by the ICM Order and associated license conditions on all reactor applicants and licensees.

REQUESTED ACTION

In order to confirm continued compliance with 10 CFR 50.54(hh)(2), within 30 days of the date of this bulletin, the NRC requests that licensees provide the following information on their mitigating strategies programs.

1. Is the equipment necessary to execute the mitigating strategies, as described in your submittals to the NRC, available and capable of performing its intended function?

2. Are the guidance and strategies implemented capable of being executed considering the current configuration of your facility and current staffing and skill levels of the staff?

Within 60 days of the date of this bulletin, the NRC requests that licensees provide information regarding their mitigation strategies programs for 10 CFR 50.54(hh)(2).

In responding to the following questions, licensees should provide information that addresses measures that are currently in place, noting any additional planned actions with expected completion dates.

1. Describe in detail the maintenance of equipment procured to support the strategies and guidance required by 10 CFR 50.54(hh)(2) in order to ensure that it is functional when needed.

   Examples of the types of information to include when providing your response to Question (1) are:

   a. Measures implemented to maintain the equipment, including periodicity.

   b. Basis for establishing each maintenance item (e.g., manufacturer's recommendation, code or standard applicable to the craft). This should include consideration of storage environment impact on the maintenance necessary.

   These examples are not meant to limit your response if you use other methods to address the issues described above.

2. Describe in detail the testing of equipment procured to support the strategies and guidance required by 10 CFR 50.54(hh)(2) in order to ensure that it will function when needed.

   Examples of the types of information to include when providing your response to Question (2) are:

   a. A description of any testing accomplished to ensure the strategies were initially feasible.

   b. A description of any periodic testing instituted for the equipment, along with the basis for establishing that test requirement.

   c. A description of the corrective action process used when the equipment fails to adequately perform its test.

These examples are not meant to limit your response if you use other methods to address the issues described above.

3. Describe in detail the controls for assuring that the equipment is available when needed.

   Examples of the types of information to include when providing your response to Question (3) are:

   a. A description of any inventory requirements established for the equipment.

   b. A listing of deficiencies noted in inventories for the equipment and corrective actions taken to prevent loss.

   These examples are not meant to limit your response if you use other methods to address the issues described above.

4. Describe in detail how configuration and guidance management is assured so that strategies remain feasible.

   Examples of the types of information to include when providing your response to Question (4) are:

   a. Measures taken to evaluate any plant configuration changes for their effect on feasibility of the mitigating strategies.

   b. Measures taken to validate that the procedures or guidelines developed to support the strategies can be executed. These measures could include drills, exercises, or walk through of the procedures by personnel that would be expected to accomplish the strategies.

   c. Measures taken to ensure procedures remain up-to-date and consistent with the current configuration of the plant.

   d. A description of the training program implemented in support of the mitigating strategies and the manner in which you evaluate its effectiveness.

   These examples are not meant to limit your response if you use other methods to address the issues described above.

5. Describe in detail how you assure availability of off-site support.

   Examples of the types of information to include when providing your response to Question (5) are:

   a. A listing of off-site organizations you rely on for emergency response.

   b. Measures taken to ensure the continuity of memoranda of agreement or understanding or other applicable contractual arrangements. This should include a listing of periods of lapsed contractual arrangements.

   c. A listing of any training or site familiarization provided to off-site responders. This should include any measures taken to ensure continued familiarity of personnel of the off-site responders in light of turnover and the passage of time.

   These examples are not meant to limit your response if you use other methods to address the issues described above.

REQUIRED RESPONSE

Licensees should address the required written response to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, 11555 Rockville Pike, Rockville, MD 20852, pursuant to the provisions of 10 CFR 50.54(f). In addition, submit a copy of the response to the appropriate Regional Administrator. Before submitting responses to the NRC, licensees must evaluate them for proprietary, sensitive, safeguards, or classified information and mark such information appropriately. The addressees have two options for submitting responses:

1. Addressees may choose to submit written responses providing the information requested above within the requested time periods.

2. Addressees, who cannot meet the requested completion date must submit written responses within 15 days of the date of this bulletin that address any alternative course of action proposed, including the basis for the acceptability of the proposed alternate course of action.

REASONS FOR INFORMATION REQUEST

The NRC is requesting information to confirm compliance with Order EA-02-026, the subsequently imposed license conditions, and 10 CFR 50.54(hh)(2) and on the status of licensee mitigating strategies programs. The staff will use the information received to inform the Commission and to determine if further regulatory action is warranted.

RELATED DOCUMENTATION

- Information Notice No. 11-05 “Tohoku-Taiheiyou-Oki Earthquake Effects on Japanese Nuclear Power Plants,” March 18, 2011

BACKFIT DISCUSSION

Under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f), this bulletin transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements (see the Applicable Regulatory Requirements section of this bulletin) and gathering information to determine the need for additional regulatory action. No backfit is either intended or approved by the issuance of this bulletin, and the staff has not performed a backfit analysis. If, as a result of information received in response to this bulletin, the NRC determines that new guidance, orders or regulations are needed, the NRC will prepare the necessary documentation to comply with the requirements of the Backfit Rule and/or any applicable finality provisions in 10 CFR Part 52 as part of the development of the new guidance, orders or regulations.

FEDERAL REGISTER NOTIFICATION

The NRC did not publish a notice of opportunity for public comment on a draft of this bulletin in the *Federal Register* because the agency is requesting information from affected licensees on an expedited basis to assess the adequacy and consistency of regulatory programs. There is no legal requirement that the NRC publish for public comment such information requests.

CONGRESSIONAL REVIEW ACT

The NRC determined that this bulletin is not a rule under the Congressional Review Act.

PAPERWORK REDUCTION ACT STATEMENT

This bulletin contains information collections that are covered by the Office of Management and Budget clearance number 3150-0012, which expires January 31, 2013. This collection of information is mandatory under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). The burden to the public for these information collections is estimated to average 200 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection.

Send comments regarding this burden estimate or any other aspect of these information collections, including suggestions for reducing the burden, to the Information Services Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail to Infocollects.Resource@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0012), Office of Management and Budget (OMB), Washington, DC 20503.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

CONTACT

Please direct any questions about this matter to the technical contact listed below.

/RA/

Timothy J. McGinty, Director
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Note: NRC Generic Communications may be found on the NRC public Web site, http://www.nrc.gov, under Electronic Reading Room/Document Collections