Report

A comparison of U.S. and Japanese regulatory requirements in effect at the time of the Fukushima accident

November 2013

EXECUTIVE SUMMARY

In the Staff Requirements Memo (SRM) to SECY-12-0110 "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," dated March 20, 2013, the Commission directed that the staff "should document its comparison of U.S. and Japanese regulatory requirements that were in effect at the time of the accident, focused on those areas most relevant to the sequence of events and accident mitigation capabilities at Fukushima." In response, staff's comparison (with contractor support) focuses on the phenomena that were especially pertinent to the Fukushima event. The staff's comparison was comprehensive within these areas, but was not an exhaustive review of all requirements for the design and licensing of nuclear power plants or their operations that may have played a role in determining how the accident unfolded. Specifically, the comparison has primarily focused on the following areas:

- Protection from design basis natural phenomena such as earthquakes, tsunami, and floods
- Loss of ultimate heat sink
- Loss of electrical power
- Containment venting
- Severe accident management

The results of this study are based principally on the review of a large amount of literature that was available in English. Two types of literature have been reviewed: the Japanese legal and regulatory documents in place at the time of the Fukushima accident, and the reports on the Fukushima accident. The first type of document provides the information needed to accomplish the goals of the study. The second type of document was very helpful in a number of ways: providing insights on how certain matters were treated in practice; confirming the observations and understanding of the Japanese legal and regulatory system and requirements; ensuring all important Japanese legal/regulatory documents are captured; etc.

Staff believes, based upon this review, that the following general observations regarding Japanese regulatory approaches can be made:

- In Japan, the regulatory framework was defined by a hierarchy of legal and regulatory documents: national laws, regulations (or Ministerial Orders and Ordinances), and regulatory guides issued by the Nuclear Safety Commission (NSC).

- Reactor design requirements and safety criteria are primarily contained in the regulatory guides of the NSC of Japan. The NSC regulatory guides are not regulatory requirements or legally binding. However, Japanese counterparts have indicated that in practice licensees would comply with these guides as though they constituted requirements.
- The NSC regulatory guides consist of Level I and Level II guides. Level I guides establish requirements that are comparable to the design basis requirements in NRC regulations for nuclear power plants. Level II guides provide additional guidance in specific technical areas, but most Level II guides are not available in English.
- The regulatory provisions of some specific NRC requirements (i.e., GDC, 10 CFR 50.46) were addressed in Japanese regulatory guides.
- Japanese regulatory requirements apparently did not consider beyond design basis events such as station blackout (SBO) events, anticipated transients without scram (ATWS) and terrorist attacks. There were also no apparent regulatory guidance documents on tsunamis and design basis floods.
- In Japan, severe accident management measures apparently did not consider natural phenomena. Japanese utilities also adopted a different containment venting strategy (delayed containment venting) than that adopted by U.S. utilities. Extensive Damage Mitigation Guidelines (EDMGs) were not required.
- Prior to the Fukushima accident, both Japanese regulators and industry publicly stated that the possibility of severe accidents was sufficiently low, to the extent that a severe accident could not occur from an engineering viewpoint.

In summary, the US and Japanese had many similarities in design bases requirements and guidance at the time of the event. There were also differences between the US and Japan in the approach to beyond design bases events and severe accidents. Staff cautions, however, that there should be no implication that the Fukushima accident and associated consequences could or would have been completely avoided assuming Japan had the same U.S. regulatory framework prior to the accident.

It should be noted that this study was performed to evaluate the similarities and differences between U.S. and Japanese regulatory requirements that were in effect at the time of the accident at Fukushima Daiichi. Since that time, Japan has re-organized the nuclear regulatory body and developed many new safety standards and regulations to improve regulatory oversight of its nuclear power plants. The NRC has also imposed additional requirements and undertaken numerous activities to address the lessons learned from the accident.

Contents

1. Introduction

In the Staff Requirements Memo (SRM) to SECY-12-0110 "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," dated March 20, 2013, the Commission directed that the staff "should document its comparison of U.S. and Japanese regulatory requirements that were in effect at the time of the accident, focused on those areas most relevant to the sequence of events and accident mitigation capabilities at Fukushima." In response, staff's comparison (with contractor support) focuses on the phenomena that were pertinent to the Fukushima event. Specifically, the comparison has primarily focused on the following areas:

- Protection from design basis natural phenomena such as earthquakes, tsunami, and floods
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- Containment venting
- Severe accident management

This report presents the findings of the comparison study. Section 2 describes briefly the literature review conducted in the course of this study. Section 3 provides an overview of the Japanese legal and regulatory documents and discusses major documents. Appendices 1 and 2 supplement Section 3 with additional information. Section 4 focuses on the comparison of U.S. requirements and Japanese requirements in the aforementioned areas. Section 5 summarizes conclusions reached in this study.

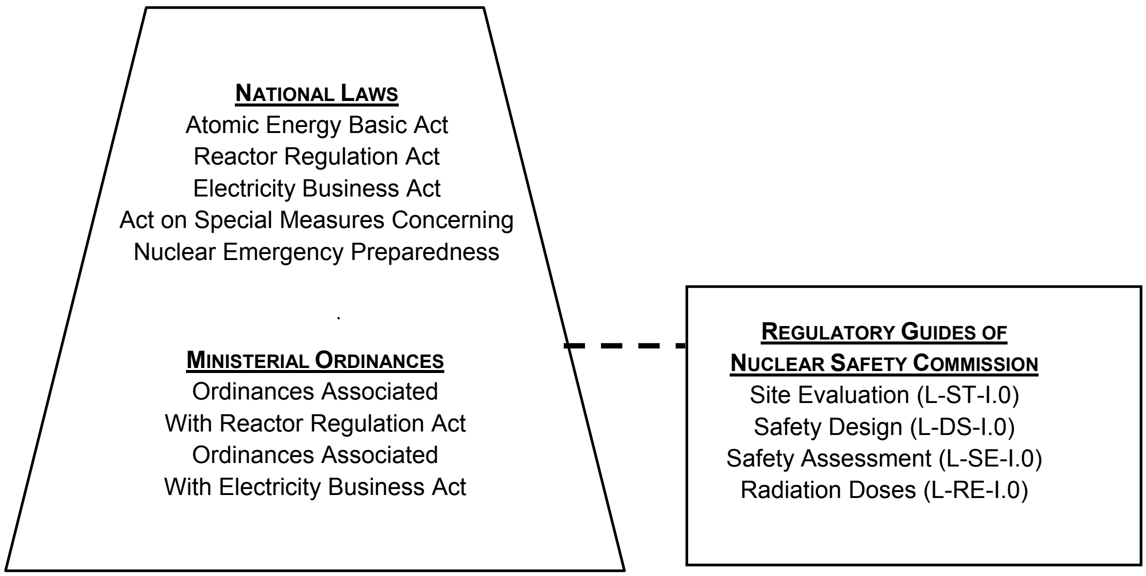
2. Literature Research

The primary approach of this study is based principally on the review of a large amount of literature that was available in English. Two types of literature have been reviewed: the Japanese legal and regulatory documents in place at the time of the Fukushima accident, and the reports on the Fukushima accident. The first type of document is the subject of the study. The second type of document was very helpful in a number of ways: providing insights on how certain matters were treated in practice; confirming the observations and understanding of the Japanese legal/regulatory system and requirements; ensuring all important Japanese legal/regulatory documents are captured; etc. In particular, the NRC Near-Term Task Force report was used as the basis for discussing NRC regulations and guidance in the selected areas. [1]

After the Fukushima accident, a large number of reports and articles have been published on the accident itself and on the lessons learned by Japan governmental organizations, international organizations, professional societies and others. In this study, over one hundred documents related to the Fukushima accident were collected and screened. Around a dozen documents were found particularly useful, some of which are referenced in this report and listed in the References section.

3. Overview of Japanese Legal and Regulatory Documents

The Japanese regulatory framework is defined by a hierarchy of legal and regulatory documents, as illustrated in the diagram below:



Some major legal and regulatory requirements are shown in the diagram. Appendix 1 provides a list of legal and regulatory documents in place in Japan at the time of the Fukushima accident. Appendix 2 contains a summary description of major Japanese legal and regulatory documents.

As illustrated in the diagram, national laws are at the top of the hierarchy. The Atomic Energy Basic Act, Reactor Regulation Act, Electricity Basic Act and Act on Special Measures concerning Nuclear Emergency Preparedness are the main laws on reactor regulation. There are other national laws. The national laws collectively define Japan's policy for utilization of atomic energy, establish organizations for regulation of nuclear reactors, delineate their responsibilities, set up licensing processes, etc. The Reactor Regulation Act, Electricity Business Act, and various regulations also establish the regulatory inspection regime and licensee safety management programs.

Below the national laws are Ministerial Orders and Ordinances which amplify certain matters addressed in the national laws and implement national laws. These Ministerial Orders and Ordinances have the effect of regulation and are legally binding. They are promulgated by the Ministries of the Japanese Government, for example, Ministry of Economy, Trade and Industry (METI), which, at the time of the event, had the primary responsibility for regulation of nuclear power reactors. The Reactor Regulation Act and Electricity Business Act, which are two main laws on reactor regulation, are supported by a number of Ministerial Orders and Ordinances.

Regulatory Guides were developed by the Nuclear Safety Commission (NSC) of Japan. The NSC was a statutory body established by the Atomic Energy Basic Act within the Cabinet Office. The Act describes NSC's responsibility as "plan, deliberate on and determine the matters related to ensuring safety among the matters related to the research, development and utilization of nuclear energy." In the licensing process, the NSC conducted its own safety review in parallel to that by the Nuclear and Industry Safety Agency (NISA) of METI, which was Japan's nuclear regulator prior to the accident. The NSC had the oversight responsibility of NISA/METI. Pursuant to the Reactor Regulation Act, METI was required to provide the NSC a quarterly report on its licensing and inspection activities.

The NSC Regulatory Guides were used by the NSC and NISA to conduct safety reviews. The NSC Regulatory Guides are not requirements, nor are they legally binding. However, in practice they were considered as such. [2] In this sense the words "require and requirement" will be used in the subsequent discussion associated with the NSC regulatory guides.

The NSC guides for power reactors have five divisions for siting (ST), design (DS), safety evaluation (SE), radiation exposure dose limit target (RE) and accident management (AM). Each division has Level I and Level II guides except for the AM division, which has one Level II document (L-AM-II.01). Level I regulatory guides are identified as main regulatory guides in [3]. Each division also has a general guide with a guide number in the format L-xx-I.0. The general guides are supported by additional Level I guides and Level II guides which typically address more specific topics. The discussion here will be primarily focused three general guides:

- Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities (L-DS-I.0)
- Regulatory Guide for Reviewing Nuclear Reactor Site Evaluation and Application Criteria (L-ST-I.0)
- Evaluating Safety Assessment of Light Water Reactor Facilities (L-SE-I.0)

As indicated earlier, the national laws and regulations (or Ministerial Orders and Ordinances) do not contain design requirements and specific safety standards for nuclear power reactors. These three guides and some supporting Level I guides were considered to provide design basis requirements. The safety design guide (L-DS-I.0) was issued in 1970 and revised in 1977, 1981, 1990 and 2001. The siting guide (L-ST-I.0) was issued in 1964 and revised in 1990. The "requirements" of these two guides were in effect at the time of the construction permit of Fukushima Daiichi Nuclear Power Plants (from 1966 to 1972 for units 1-6). [2]

The safety design guide (L-DS-I.0) contains fifty-nine general "requirements" (or "guidelines"), which are essentially equivalent to the General Design Criteria (GDC) of Appendix A to 10 CFR Part 50 with variations in organization and use of language or terms (which may be due to translation). The stated purpose for this guide is to provide

> *"the basis of the judgment for adequacy of the design to ensure safety at the Safety Review related to the application for the establishment license (includes the application of alteration of an establishment license) of the individual light water nuclear power reactor (hereafter referred to as "LWR")."*

In other words, the approval of a construction permit requires that the proposed reactor design meets the requirements of the regulatory guide (L-DS-I.0). Meeting the requirements of L-DS-I.0 would "ensure" adequate design safety.

The siting guide (L-ST-I.0) was used in evaluation of the suitability of reactor site conditions. The guide provides a site exclusion zone, a low population zone, and sites being located away from densely populated regions. Dose acceptance criteria are specified for these areas and are used to assess the acceptability of a site. L-SE-I.0, "Regulatory Guide for Evaluating Safety Assessment of Light Water Reactor Facilities," provides dose assessment guidelines. In accordance with these two guides, the suitability of a site is primarily based on the estimated doses from selected design basis accidents meeting design basis dose criteria.

The current U.S. reactor site criteria are contained in Subpart B of 10 CFR Part 100 of the U.S. NRC regulations, which address a broader scope of issues compared to L-ST-I.0. In addition, the dose criteria (at site boundary and in the low population zone) were relocated in 10 CFR 50.34. The dose assessment is primarily used to evaluate the adequacy of design, particularly, the performance of containment systems (vs. site suitability).

The safety evaluation guide (L-SE-I.0) was initially issued in 1990 and revised in 2001. The stated purpose of this guide is to provide

> *"the basis of the judgment for adequacy of the design to ensure safety at the Safety Review related to the application for the establishment license (includes the application of alteration of an establishment license) of the individual light water nuclear power reactor (hereafter referred to as "LWR")."*

In other words, this guide is used in a licensing review to evaluate the adequacy of the safety assessment provided in the application for permission to install (or modify) LWRs. This guide requires two types of safety assessments: "safety design assessment" and "site assessment". Guidance and acceptance criteria are provided for conducting these assessments.

“Safety design assessment” evaluates the response and performance of the plant safety systems to selected anticipated operating occurrences and postulated accidents (or design basis events). The scope is similar to that of Chapter 15 “Transient and Accident Analyses” of the Safety Analysis Report (SAR) in U.S. However, SARs for plants in U.S. also include discussions of the Station Blackout (SBO) and Anticipated Transient without Scram events (ATWS).

“Site assessment,” which determines the suitability of a site, evaluates the radiological consequences that could result from major accidents (selected postulated or design basis accidents). The scope of “site assessment” is similar to the “Design Basis Accident Radiological Consequence Analysis” part of Chapter 15 of the SAR in the U.S.

Another safety evaluation guide worth mentioning is L-SE-I.02, “Regulatory Guide for Evaluating Emergency Core Cooling System Performance of LWR.” This guide contains the same acceptance criteria contained in 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.” This guide provides additional guidance for conducting LOCA analyses.

Overall, from the review of the material presented above, the following observations can be made:

- In Japan, the regulatory framework is defined by a hierarchy of legal and regulatory documents: national laws, regulations (or Ministerial Orders and Ordinances), and regulatory guides issued by the NSC.
- Reactor design requirements and safety criteria are contained in the NSC regulatory guides. The NSC regulatory guides are not regulatory requirements or legally binding. However, Japanese counterparts have indicated that in practice licensees would comply with these guides as though they constituted requirements.
- The NSC Level I guides establish requirements that are comparable to the design basis requirements in NRC regulations for nuclear power plants. The safety assessment is within the scope of design basis analyses.
- The NSC guides do not contain provisions for beyond design basis events such as SBO, ATWS, and loss of large plant areas due to a terrorist attack.

4. Comparison of U.S. Requirements and Japanese Requirements

This Section will compare the U.S. requirements and Japanese requirements in place at the time of the Fukushima accident in the areas discussed in Section 1, namely,

- Protection from design basis natural events such as earthquakes, tsunami, and floods
- Loss of ultimate heat sink
- Loss of electrical power
- Containment venting
- Severe accident management

As mentioned earlier in this report, the discussion of the U.S. requirements is primarily based on the information in the NRC Near-Term Task Force report [1]. The review covers all the aforementioned areas.

The table below lists relevant regulations and guides in the U.S. and Japan that were in the areas that were part of this study.

List of Relevant Regulations and Guides

	Regulations		Regulatory / industry guides	
	U.S.	Japan[a]	U.S.	Japan
Protection from design basis natural events	GDC 2 10 CFR 100.20 10 CFR 100.23 Appendix S to 10 CFR 50	Guideline 2, L-DS-I.0 L-ST-I.0 L-DS-I.02	RG 1.29, 1.59, 1.60, & 1.208 (seismic) RG 1.102 & 1.125 (flood) USI A-46	
Electrical power	GDC 17 & 18 10 CFR 50.63	Guidelines 27 & 48, L-DS-I.0	RG 1.155 NUREG 0800	
Ultimate Heat Sink	GDC 45	Guideline 26, L-DS-I.0		
Mitigation features	10 CFR 50.54(hh)		EDMGs GL 89-16[1]	NSC Decision on May 28, 1992 [8] (L-AM-II.01)
Severe accident management			GL88-20[2] NEI 91-04[3]	NSC Decision on May 28, 1992 [8] (L-AM-II.01)

a. Because Japan's regulations do not provide technical requirements on related matters and based on the discussion on the role of the NSC Regulatory Guides in Section 3 of this report, the Level I guides of Japan NSC are treated as "regulatory requirements" for the comparison purpose

[1] GL 89-16, "Installation of a Hardened Wetwell Vent"
[2] GL 88-20, Supplement 2, "Accident Management Strategies for Consideration in the Individual Plant Examination Process," April 4, 1990
[3] NEI 91-04 (Revision 1), Severe Accident Issue Closure Guidelines, December 1994

Protection from Design Basis Natural Phenomena (Earthquake, Floods)

The importance of protection of nuclear power plants from natural phenomena has long been recognized both in the U.S. and in Japan. This is reflected in the fact that the protection from natural phenomena was included in the GDC when the U.S. Atomic Energy Commission (AEC) established the GDC in 1971 and in the regulatory guide L-DS-I.0 developed by Japan's NSC when the guide was initially published in 1970. The information presented below looks at regulations and programs addressing this aspect in the U.S. and Japan respectively. The discussion will be limited to earthquake and floods.

<u>U.S.</u>

The relevant NRC regulations and regulatory guides on earthquake and floods are listed below:

- GDC 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A to 10 CFR Part 50
- 10 CFR 100.20, "Factors to Be Considered When Evaluating Sites"
- 10 CFR 100.23, "Geologic and Seismic Siting Criteria"
- Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50
- RG 1.29, "Seismic Design Classification," issued in 1972 and updated in 1973, 1976, 1978, and 2007
- RG 1.59, "Design Basis Floods for Nuclear Power Plants," issued in 1973 and updated in 1976 and 1977
- RG 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," issued in 1973
- RG 1.102, "Flood Protection for Nuclear Power Plants," issued in 1975 and updated in 1976
- RG 1.125, "Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants," issued in 1977 and updated in 1978 and 2009
- RG 1.208, "A Performance-Based Approach To Define the Site-Specific Earthquake Ground Motion," issued in 2007

The U.S. regulations on natural phenomena have been continuously evolving and enhanced with new information developed, operating experience, and advances in technology and analytical techniques. Currently the NRC staff is updating RG 1.59 and it is expected to address tsunamis and advances in flood analysis. Seismic hazard analysis has transitioned to probabilistic and risk informed approaches.

In the past several decades, the NRC and the industry have undertaken a number of initiatives to address potential plant vulnerabilities to natural phenomena. In 1977, the NRC initiated the Systematic Evaluation Program (SEP) to review the designs of older operating nuclear reactor plants in order to reconfirm and document their safety. The SEP covered topics including seismic events, floods, high winds, and tornadoes.

In 1980, the NRC was concerned that licensees had not conducted the seismic qualification of electrical and mechanical equipment in some older nuclear reactor plants in accordance with the licensing criteria for the seismic qualification of equipment acceptable at that time. As a result, the NRC established the Unresolved Safety Issue (USI) A-46, “Seismic Qualification of Mechanical and Electrical Equipment in Operating Nuclear Power Plants,” program in December 1980. The objective of USI A-46 was to develop alternative seismic qualification methods and acceptance criteria that could be used to assess the capability of mechanical and electrical equipment in operating nuclear power plants to perform their intended safety functions. The scope of the review was limited to equipment required to bring each affected plant to hot shutdown and maintain it for a minimum of 72 hours.

In 1991, the NRC issued Supplement 4 to Generic Letter (GL) 88-20, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f).” This GL requested that “each licensee perform an individual plant examination of external events to identify vulnerabilities, if any, to severe accidents and report the results together with any licensee determined improvements and corrective actions to the Commission.” The external events considered in the IPEEE program include seismic events, internal fires, high winds, and floods.

<u>Japan</u>

Japan’s regulatory requirements and guidance on natural phenomena are included in the following:

- Guideline 2 of L-DS-I.0: Design Considerations against Natural Phenomena
- L-ST-I.0: Regulatory Guide for Reviewing Nuclear Reactor Site Evaluation and Application Criteria (L-ST-I.0)
- L-DS-I.02: Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities

Guideline 2 of L-DS-I.0 is comparable to GDC 2 of Appendix A to 10 CFR Part 50. L-ST-I.0 is discussed in Section 3 of this report. In accordance with this guide, the suitability of a site was primarily based on design basis radiological consequence dose assessment.

L-DS-I.02 was initially issued in 1977 and revised in 2006. It provides guidance on seismic classification, seismic hazard analysis (determining design basis ground motion), and seismic design criteria (e.g., load combinations, stress limits). The seismic hazard analysis methodology is of a deterministic nature. NISA requested that all NPPs should undergo a back check on the basis of the 2006 version of the guide, which includes the need to re-evaluate the seismic hazards at the Nuclear Power Plant (NPP) sites to obtain the new Design Basis Earthquake Ground

Motion (DBGM) and to implement the necessary upgrades. This study has not looked into what upgrades were implemented at Japanese nuclear power stations as a result of this back check.

With respect to tsunami, the IAEA mission report points out that [2]

> *"After the issuance of the Construction Permit (Daiichi plant) about forty years ago, the regulatory authority did not provide any requirements or guidance regarding tsunami safety. The guidance provided in 2006 as part of the Seismic Safety Guidelines, does not contain any concrete criteria or methodology that could be used in re-evaluation. The only re-evaluation was performed in 2002 by TEPCO on a voluntary basis. Even this work was not reviewed by NISA. Therefore an effective regulatory framework was not available to provide for the tsunami safety of the NPPs through their operating life."*

Though the regulatory authority requested licensees to conduct back checks on special occasions, backfitting is not institutionalized. [3] Prior to the Fukushima accident, the prevailing view in Japan had been that Japanese nuclear installations were designed and built with sufficient margin to prevent events beyond design basis accidents or to mitigate such events. At the same time, voluntary modifications of facilities were not restricted. In this context, "backfit" is understood as "retroactive application of new rules to existing reactors." [5]

Loss of Ultimate Heat Sink

NRC design basis requirements for removal of heat to the ultimate heat sink are contained in GDC 44 - 46 of Appendix A to 10 CFR Part 50. The Japan NSC requirements are given in Guideline 26 of L-DS-I.0. There is no apparent difference in design basis requirements for the heat removal systems in the NRC requirements and Japanese requirements. Similarly, emergency operation procedures (EOPs) exist to respond to a loss of ultimate heat sink event at U.S. plants and Japanese plants. As discussed in the following, Japanese plants had incorporated features to provide diverse means to remove heat from the core and containment. After the 9/11 terrorist attack, all licensees in the U.S. have implemented the extensive damage guidelines (EDMGs) and enhanced the plant capability to maintain or restore core cooling, containment and spent fuel pool cooling capabilities. At the time of the accident, Japanese utilities had not developed and implemented EDMGs.

Loss of Electrical Power

NRC design basis requirements for electrical power are provided in GDC 17 and 18 of Appendix A to 10 CFR 50. In Japan, NSC design basis requirements for electrical power systems are provided in Guidelines 27 and 48 of the safety design guide (L-DS-I.0). There is no apparent difference in the design basis requirements for electrical power systems in the U.S. requirements and Japanese requirements. The major difference existed in the requirement for the plant capability to accommodate the SBO event, in which a simultaneous loss of the offsite power and onsite emergency AC power takes place.

<u>U.S.</u>

In the U.S., the NRC station blackout rule, "10 CFR 50.63 Loss of All Alternating Current (AC) Power" was promulgated in 1988. The implementing guidance is provided in Regulatory Guide 1.155 "Station Blackout." As the result of implementing this rule, the operating nuclear power plants in the U.S. had 4 – 8 hours of coping capability. However, the RG 1.155 did not take into account loss of offsite power caused by flood or earthquake.

After the 9/11 terrorist attack, the licensees are required to implement the EDMGs to enhance the plant mitigation capability as the result of loss of a large number of important equipment in a large plant area due to events such as a terrorist attack involving a large commercial airplane. The EDMGs and associated equipment could conceivably further enhance plant capability to cope with an SBO event. However, the equipment associated with the EDMGs is not designed to be protected from design basis or beyond design basis events.

Japan

Japanese regulations did not have a station blackout rule. Guideline 27 of L-DS-I.0 requires:

> *"The nuclear reactor facilities shall be so designed that safe shutdown and proper cooling of the reactor after shutting down can be ensured in case of a short-term total AC power loss."*

The commentary on this guideline of the guide further explains:

> *"No particular considerations are necessary against a long-term total AC power loss because the repair of troubled power transmission line or emergency AC power system can be expected in such case.*
>
> *The assumption of a total AC power loss is not necessary if the emergency AC power system is reliable enough by means of system arrangement or management (such as maintaining the system in operation at all times)."*

The duration for "a short-term AC power loss" or "a long-term total AC power loss" is not given. According to a report by Institute of Nuclear Power Operations (INPO), [6] the TEPCO EOP procedures were developed assuming that a loss of all AC power would not last for more than 30 minutes and that the coping time could be extended up to eight hours using station batteries. This assumption was based on the multiple off-site transmission lines, the availability of backup diesel generators, and extensive features to cross-tie and share electrical power sources among the units.

Containment Venting

U.S.

At the time of the Fukushima accident, all boiling water reactor (BWR) facilities with Mark I containment designs and three of the eight BWR units with Mark II containments in the U.S. had voluntarily installed harden vents. No regulatory requirement was imposed. The updated final safety analysis report for each facility includes a description of the hardened vent, but the vent is not a required design feature for that facility. The designs of those vents varied from plant to plant. At some facilities, the hardened vent relied on ac-operated valves, some relied on dc-operated or air-operated valves, and some incorporated passive rupture disks along with isolation valves. Each different design has different operational complexities during a prolonged SBO scenario.

Japan

In Japan, a filtered or wetwell vent system was considered as one severe accident mitigation feature in the NSC document L-AM-II.01[4]. However, in the 1980s and afterward, Japanese utilities and vendors made decisions to deviate from accident management strategies developed by the U.S. BWR Owners Group and adopted a different containment venting strategy. [6] In general, the primary containment vessel (PCV) venting was designed to delay venting as long as possible to avoid the release of radioactive materials. In keeping with this strategy, vent lines include rupture disks sized not to fail until containment pressure reaches the maximum operating value. If fuel damage has occurred, accident management guidelines indicate that venting is warranted when pressure is expected to reach the established threshold (e.g., Fukushimi Daiichi NPP guidance set at two times the maximum operating value), there is no prospect for the recovery of containment spray, and the water injection amount has not covered the torus vent line. Site superintendent permission is needed to vent the containment.

However, this strategy of delayed venting does not appear to have considered the likelihood of increased hydrogen leakage during periods with high containment pressure. In addition, potential impacts on the effectiveness of low-pressure injection under accident conditions also does not appear to have been considered.

In contrast, the U.S. BWRs typically do not have rupture disks that would prevent early venting, and emergency operating procedures require that venting be initiated before the containment design pressure is reached.[6] If fuel damage has occurred, procedure guidance calls for earlier venting based on hydrogen concentration inside containment to reduce the potential for explosions inside the PCV. The decision to initiate venting is made by the shift manager, with consultation and advice from the site emergency response center.

[4] Accident Management for Severe Accidents at Light Water Power Reactor Installations

Accident Management

U.S.

In the U.S., a number of guidelines and procedures govern onsite emergency actions by the operators depending on the nature and extent of events at the plant, including EOPs, severe accident management guidelines (SAMGs) and EDMGs. The NRC and industry have established the command and control responsibility for each of these programs.

EOPs have been part of the NRC safety requirements and in the administrative controls section of the technical specifications (TS) for each plant. In implementing the EOPs, the command and control functions remain in the control room under the direction of the shift supervisor and plant manager.

In response to the TMI accident and follow-up activities, the nuclear industry voluntarily developed SAMGs during the 1980s and 1990s. The SAMGs are voluntary. In Supplement 2 to GL 88-20[5], the NRC encouraged but did not require licensees to develop and implement SAMGs. The SAMGs are intended for use by plant technical support staff, usually located in the plant's Technical Support Center (TSC), and are meant to enhance the ability of the operators to manage accident sequences that progress beyond the point where EOPs and other plant procedures are applicable and useful. EOPs typically cover accidents to the point of loss of core cooling and initiation of inadequate core cooling. The objective of the EOPs is to prevent core damage. The objectives of the SAMGs are described as: [7]

- Terminate core damage progression once it begins;
- Maintain the capability of the containment as long as possible; and
- Minimize on-site and off-site releases and their effects.

When implementing the SAMGs, the command and control functions are shifted to the TSC and typically to the emergency coordinator or shift technical advisor or both.

Following the terrorist events of September 11, 2001, the NRC issued security advisories, orders, license conditions, and ultimately a new regulation (10 CFR 50.54(hh)) to require licensees to develop and implement guidance and strategies intended to maintain or restore core cooling and containment and spent fuel pool cooling capabilities under the circumstances associated with the loss of large areas of the plant due to fire or explosion. These requirements have led to the development of EDMGs at all U.S. nuclear power plants. The requirements of 10 CFR 50.54(hh) have been added to the agency's routine inspection program as part of the triennial fire protection inspections. In terms of command and control, control room, plant, TSC, or emergency operations facility (EOF) staff could make EDMG decisions.

Japan

[5] GL 88-20, Supplement 2, "Accident Management Strategies for Consideration in the Individual Plant Examination Process," April 4, 1990

In Japan, the NSC issued its policy or decision regarding severe accident management on May 28, 1992 and revised it on October 20, 1997. [8] The decision recommended that the licensees should develop and implement severe accident management strategies on a voluntary basis. The purpose was to further reduce the risk from operation of nuclear power reactors, which was considered already sufficiently low. In the decision, the NSC viewed the possibility of severe accidents as sufficiently low to the extent that severe accidents could not occur from an engineering viewpoint as a result of implementing the then-current regulatory framework.

The NSC document or guide L-AM-II.01 (1992) provides an overview of the status of development and implementation of accident management (AM) measures in Japan at that time. Further information on this document is provided in Appendix 2. Based on this document, various Phase I AM measures (or EOPs) had been developed and implemented at the nuclear power stations in Japan. Phase II AM measures (or SAMGs) were being considered or studied by the utilities. The following discussion of the implementation of severe accident measures is primarily based on several reports on Fukushima accident since the NSC 1992 decision. [2, 5, 6, 9, 10]

- Perception on severe accident risk

Prior to the Fukushima accident, the prevailing perception on the severe accident risk of the regulatory bodies and the industry in Japan had been that the probability of a severe accident was considered to be too low to occur in reality from an engineering point of view. [8] It is believed that this perception of severe accident risk had significant influence on how severe accident management was treated in Japan. It also appears that there may be confusion between the concept of risk and the concept of probability of occurrence.

In the U.S., the NRC believes, on one hand, the existing regulatory framework provides adequate protection and the operating plants pose no undue risk to the safety and health of the public. On the other hand, the NRC recognizes that severe accidents,

> *"which are beyond the substantial coverage of the design basis events, constitute the major risk to the public associated with the release of radioactive materials from nuclear power plant accidents. A fundamental objective of the Commission's severe accident policy is that the Commission intends to take all reasonable steps to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur."* [11]

- Development and implementation of severe accident management measures

Following the NSC's decision on May 28, 1992, [8] Japanese utilities had implemented severe accident management measures on a voluntarily basis. Typical modifications to plants include installation of alternative pumps or procedures for core flooding using containment spray systems, water injection using fire pumps, and additional provisions for ac power using a tie-line from a neighboring installation. These measures provide diversity for essential safety functions but are vulnerable to a large external event since they are located on the site. In May 2002, Japanese licensees reported the accident management measures developed for each unit. [2]

In the aftermath of the Fukushima accident, both Japanese and international investigations revealed a number of shortcomings in the severe accident management strategies in Japan.

- Technical bases for severe accident measures

There are a number of shortcomings in the technical bases on which the severe accident management measures were in part based. As discussed earlier, the Japanese utilities decided in 1980s and 1990s to depart from the SAMGs developed by U.S. BWR Owner Group and to adopt a delayed containment venting strategy. The decision to adopt this strategy does not appear to have considered the possibility of leakage of hydrogen out of the containment vessel and the impact of the resulting high containment pressure on low pressure injection.

Probabilistic safety assessments (PSAs) used in development of severe accident management measures were of limited scope. Level 1 PSAs were available for internal events. No Level 2 PSA results were available, though a Level 2 PSA is crucial to understand accident progression and the response and conditions of the containment following core damage. In May 2002, Japanese licensees reported the accident management measures developed for each unit together with PSAs of internal events for representative reactor types for the purpose of quantifying safety. The PSA of internal events for all commercial reactor facilities under operation were reported to NISA by the licensees in March 2004. No PSAs for external events were required by NISA.

Severe accident measures were based on potentially non-conservative assumptions. For example, SAMGs and associated procedures generally assume that instruments, lighting and power are available. These documents do not consider the possible state of the plant and the local environmental conditions such as radiation fields that may preclude manual actions from being taken.

5. Conclusions

Staff believes, based upon this review, that the following general observations regarding Japanese regulatory approaches can be made:

- In Japan, the regulatory framework is defined by a hierarchy of legal and regulatory documents: national laws, regulations (or Ministerial Orders and Ordinances), and regulatory guides issued by the Nuclear Safety Commission (NSC).
- Reactor design requirements and safety criteria are primarily contained in the regulatory guides of the NSC of Japan. The NSC regulatory guides are not regulatory requirements or legally binding. However, Japanese counterparts have indicated that in practice licensees would comply with these guides as though they constituted requirements.
- The NSC regulatory guides consist of Level I and Level II guides. Level I guides establish requirements that are comparable to the design basis requirements in NRC regulations for nuclear power plants. Level II guides provide additional guidance in specific technical areas, but most Level II guides are not available in English.
- The regulatory provisions of some specific NRC requirements (i.e., GDC, 10 CFR 50.46) were addressed in Japanese regulatory guides.
- From our review, Japanese regulatory requirements apparently did not specifically address beyond design basis events such as station blackout (SBO) events, anticipated transients without scram (ATWS) and terrorist attacks. It was not apparent that Japanese guidance documents at the time of the accident addressed tsunamis or design basis floods.
- From our review, it was not apparent that Japanese regulatory requirements for severe accident management measures considered natural phenomena. Japanese utilities also adopted a different containment venting strategy (delayed containment venting) than that adopted by U.S. utilities. Extensive Damage Mitigation Guidelines (EDMGs) were not required.
- Prior to the Fukushima accident, both Japanese regulators and industry publicly stated that the possibility of severe accidents was sufficiently low, to the extent that a severe accident could not occur from an engineering viewpoint.

In summary, the US and Japanese had similarities in design bases requirements and guidance at the time of the event. There were also differences between the US and Japan in the approach to beyond design bases events and severe accidents. Staff cautions, however, that there should be no implication that the Fukushima accident and associated consequences could or would have been completely avoided assuming Japan had the same U.S. regulatory framework prior to the accident.

This review and previous evaluations have highlighted some similarities and some differences in specific regulatory requirements imposed on nuclear power plants in Japan and the U.S. Following the Fukushima accident, the NRC evaluated regulatory requirements and practices in the U.S. to determine what improvements were warranted given the insights from the events in Japan. The NRC focused on the course of events leading up to, during, and after the Fukushima accident to determine if its regulations were sufficient to ensure that U.S. plants would be able to prevent or mitigate the types of conditions that contributed to core damage and the release of radioactive materials following the earthquake and tsunami in Japan. Although programs related to severe accident management and mitigating strategies had been implemented in the U.S., the NRC determined that additional improvements in these areas were needed to address the lessons learned from the Fukushima accident. As an example, the mitigating strategies implemented at U.S. nuclear plants following the terrorist attacks of September 11, 2001, to cope with large fires and explosions may have helped in responding to an extended loss of electrical power and core cooling capability that occurred at Fukushima if the equipment was stored in an area of the plant that was not inundated by the tsunami. However, NRC requirements did not require this equipment to be designed or located to handle multi-unit events or survive extreme natural phenomena, such as a beyond design basis flood. Upon identifying these limitations, the NRC's response was to issue orders to U.S. plants to have additional portable power supplies and pumps that would be protected against flooding and other natural hazards. The staff's efforts to identify and implement the remaining Fukushima lessons learned follow a similar methodology.

References

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3. Convention on Nuclear Safety National Report of Japan for the Fifth Review Meeting, 2010
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6. Lessons Learned from the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station, INPO, August 2012
7. NEI 91-04 (Revision 1), Severe Accident Issue Closure Guidelines, December 1994
8. NSC Decision: Accident Management of Severe Accidents at Power Generating Light Water Reactor Facilities, May 28, 1992 (Amended on October 20, 1997)
9. Technical Knowledge of the Accident at Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Co., Inc., NISA, March 2012
10. Convention on Nuclear Safety: National Report of Japan for the Second Extraordinary Meeting, July 2012
11. Policy Statement on Severe Reactor Accidents Regarding Future Design s and Existing Plants, U.S. Nuclear Regulatory Commission, August 1985

Appendix 1 - List of Japanese Laws, Regulations, and Regulatory Guides

In Japan, the nuclear regulatory framework includes a hierarchy of legal and regulatory documents, including national laws, Cabinet orders and Ministerial Ordinances (or regulations), and regulatory guides. The following are a list of the laws, regulations and regulatory guides related to regulation of nuclear power reactors in Japan prior to the Fukushima accident. [3, 4]

<u>National Laws</u>

- Atomic Energy Basic Act
- The Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (or Reactor Regulation Act)
- The Electricity Utilities Industry Law (or Electricity Business Act)
- Act on Special Measures Concerning Nuclear Emergency Preparedness
- Environmental Impact Assessment Act
- Act Concerning Prevention from Radiation Hazards due to Radioisotopes, etc. (or Radiation Hazard Prevention Act)
- Law on Compensation for Nuclear Damage

<u>Ministerial Ordinances</u>

In Japan, Cabinet Orders and Ministerial Ordinances have the effect of regulations, which are enacted by the Government to implement the public laws. This section lists the Ministerial Ordinances associated with Reactor Regulatory Act, Electricity Business Act, and Act on Special Measures Concerning Nuclear Emergency.

Reactor Regulatory Act

- The Rule for the Installation, Operation, etc., of Commercial Nuclear Power Reactors
- Ordinance for the Enforcement of the Reactor Regulation Act
- Ordinance for Implementation of Inspection, Etc. by Japan Nuclear Energy Safety Organization (JNES)
- Notification for Radiation Dose Rate Limits, etc.
- Rules Concerning Report to Nuclear Safety Commission

Electricity Business Act

- Ministerial Order for Performing Inspections, Etc. by the JNES
- Rules for the Enforcement of the Electricity Business Act
- Ordinance of Establishing Technical Standards for Nuclear Power Generation Equipment
- Technical Standards on Dose Equivalent, etc. due to Radiation Relating to Nuclear Power Generation Equipment

Act on Special Measures Concerning Nuclear Emergency Preparedness

- Ordinance for the Enforcement of the Special Law of Emergency Preparedness for Nuclear Disaster
- Ordinance for the Enforcement of the Basic Law on Emergency Preparedness
- Basic Plan for Emergency Preparedness

<u>Regulatory Guides</u>

Regulatory guides were established by the Nuclear Safety Commission (NSC) of Japan. The regulatory guides were not regulations. The regulatory guides established design basis requirements and safety evaluation criteria. The regulatory guides related to nuclear power reactor are listed in the following, which cover siting, design, safety evaluation, dose limits and accident management. [4] The underlined regulatory guides are identified as main regulatory guides. [3] The main guides are also referred to as Level I guides or documents. Most of Level II regulatory guides are not available in English.

Siting

- <u>Regulatory Guide for Reviewing Nuclear Reactor Site Evaluation and Application Criteria (L-ST-I.0)</u>

Design

- <u>Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities (L-DS-I.0)</u>
- <u>Reviewing Classification of Importance of Safety Function of Light Water Nuclear Power Reactor Facilities (L-DS-I.01)</u>
- <u>Reviewing Seismic Design of Nuclear Power Reactor Facilities (L-DS-I.02)</u>
- <u>Regulatory Guide for Reviewing Fire Protection of Light Water Nuclear Power Reactor Facilities (L-DS-I.03)</u>
- <u>Regulatory Guide for Reviewing Radiation Monitoring in Accidents of Light Water Nuclear Power Reactor Facilities (L-DS-I.04)</u>
- Fuel assemblies for Boiling Water Reactors (L-DS-II.01)
- 17x17 Fuel Assemblies for Pressurized Water Reactors (L-DS-II.02)
- Core Thermal Design and Thermal Operation Limit Setting Methods (L-DS-II.03)
- Application of "Core Thermal Design and Thermal Operation Limit Setting Methods (L-DS-II.04)
- Guide for Geological and Ground Safety Examination of Nuclear Power Plants L-DS-II.05)
- Fuel Design Methodology of Power Generating Light Water Reactors (L-DS-II.06)
- Design Considerations for Internally Originated Flying Objects" Caused by Pipe Rupture Accidents (L-DS-II.07)
- 9x9 Fuel Assemblies for Boiling Water Reactors (L-DS-II.09)
- Mixed Oxide Fuels for Power Generating Light Water Reactors (L-DS-II.10)
- Full Core Loading of Mixed Oxide Fuels in Advanced Boiling Water Reactors (L-DS-II.11)
- Validity Evaluation Fuel Integrity Criteria under Boiling Transition Conditions (L-DS-II.12)

- Guidelines for Reviewing Seismic Safety Concerning Geological Faults (L-DS-II.13)

Safety Evaluation

- Evaluating Safety Assessment of Light Water Reactor Facilities (L-SE-I.0)
- Evaluating Core Thermal Design of Pressurized Water Cooled Nuclear Power Reactors (L-SE-I.01)
- Evaluating Emergency Core Cooling System Performance of Light Water Power Reactors (L-SE-I.02)
- Evaluating Reactivity Insertion Events of Light Water Nuclear Power Reactor Facilities (L-SE-I.03)
- Evaluating Dynamic Loads on BWR MARK-I Containment Pressure Suppression Systems (L-SE-I.04)
- Evaluating Dynamic Loads on BWR MARK-II Containment Pressure Suppression Systems (L-SE-I.05)
- Meteorologica1 Observation for Safety Analysis of Nuclear Power Reactor Facilities (L-SE-I.06)
- Clarified Interpretation of "No Mechanical Failures of Fuel Cladding" (L-SE-II.01)
- Decay Heat Data to be Used in Performance Evaluation of the Emergency Core Cooling Systems of Light Water Power Reactors(L-SE-II.02)
- Evaluation of High-Burnup Fuels for Reactivity Insertion Events in Light Water Power Reactors (L-SE-II.03)
- Safety Evaluation of Reload Cores (of Light Water Power Reactors) (L-SE-II.04)
- Application Guidance of "Reference Dose for Plutonium for Site Evaluation of Nuclear Reactors Using Plutonium Fuel (L-SE-II.05)

Radiation Exposure Dose Limit Target (RE)

- Annual Dose Target for the Public in the Vicinity of Light Water Power Reactor Facilities (L-RE-I.0)
- Reviewing Evaluation of Dose Target for Surrounding Area of Light Water Nuclear Reactor Facilities (L-RE-I.01)
- Radiation Monitoring of Effluent Released from Light Water Nuclear Power Reactor Facilities (L-RE-I.02)
- Assessment of Exposure Dose of the Public in Safety Examination of Power Generating Light Water Reactor Facilities (L-RE-II.01)
- Radiation Energy and Other Relevant Data for Exposure Dose Calculations (L-RE-II.02)

Accident Management (AM)

- Accident Management for Severe Accidents at Light Water Power Reactor Installations (L-AM-II.01)

Appendix 2 - Summary Description of Main Japanese Laws, Regulations, and Regulatory Guides

Appendix 1 provides a list of Japan's legal and regulatory documents in place in Japan at the time of the Fukushima accident. This Appendix provides a summary description of four main laws and associated Ordinances, including Atomic Energy Basic Act, Reactor Regulation Act, Electricity Business and Act on Special Measures Concerning Nuclear Emergency Preparedness, and main regulatory guides. These documents were reviewed for the primary purpose of identifying Japan's regulatory requirements in the areas discussed in Section 1 of the main report, namely, protection from design basis natural phenomena, loss of the ultimate heat sink, loss of electrical power, containment venting and severe accident management. This Appendix supplements Section 3, by providing an overview of the Japanese regulatory requirements in place at the time of the accident. The full-text English version of the Ordinances is not available. Therefore, the summary of the Ordinances is based on their excerpts.

- **Atomic Energy Basic Act**

The Atomic Energy Basic Act of 1955 is the basic law of Japan on utilization of nuclear energy and materials. The Act states its purpose as securing a future energy source, promotion of development and utilization of the nuclear energy, and advancement of research and technology. The Act also sets the basic policy which requires, among other things, nuclear technology be limited to peaceful use and safety be ensured. It requires activities related to nuclear materials, nuclear reactors and radiation hazard prevention be subject to the Government regulation.

The Act establishes within the Cabinet Office, the Japan Atomic Energy Commission (JAEC) and NSC. The former is charged with the responsibility of development and implementation of national policies on promotion and development of nuclear technology. The latter is charged with the responsibility of development and implementing national policy on nuclear safety issues.

- **Reactor Regulation Act and Associated Ordinances**

<u>Reactor Regulation Act</u>

The Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (or Reactor Regulation Act) of 1957, is a comprehensive law which implements the Atomic Energy Basic Act. The Reactor Regulation Act has fourteen chapters, covering control and use of radiation sources and a variety of facilities, i.e., nuclear fuel fabrication, nuclear reactors, spent fuel storage facilities, processing facilities, etc.

Chapter IV of the Reactor Regulation Act provides for regulations related to commercial power reactors, reactors in the development stage, and research and test reactors. The Act establishes licensing proceedings for different types of reactors (e.g., commercial power reactor, reactor under development, research and test reactors), including permitting establishment of a reactor, license amendment, license transfer, license termination, decommissioning, etc. The Act gives the METI the authority of licensing commercial power reactors. The Act establishes the allegation system and penalties for violation of the Act. According to the Act, the METI must approve the operation plan of the licensee of a commercial reactor before commencing operations, is required to conduct Facility Periodic Inspections concerning the performance of the facility, and is responsible for qualifying and approving the Chief Engineer of Reactors.

The Act does not contain specific design and safety requirements (or technical standards) for nuclear reactors. It is of an administrative nature.

The Rule for the Installation, Operation, etc., of Commercial Nuclear Power Reactors

The Ministerial Ordinance was enacted to implement the provisions of the Reactor Regulation Act. The Ordinance specifies the scope of an application for establishment of nuclear reactors. Similar to the Reactor Regulation Act, this Ordinance does not provide specific design or safety requirements.

Notification for Radiation Dose Rate Limits, etc.

This Ordinance specifies the radiation exposure doses, concentration, and density in controlled areas and the environmental monitoring area, the surface contamination limit, radiation exposure dose limits of radiation workers, radioactivity concentration limits outside of the environmental monitoring area.

Ordinance for Implementation of Inspection Etc. by JNES

Pursuant to the Reactor Regulation Act, the JNES is given the responsibility by the METI for conducting inspections prescribed in the Act. This Ordinance amplifies the requirements of the Act for qualifications of the JNES inspectors, i.e., education, knowledge, training, experience.

Ordinance for the Enforcement of the Reactor Regulation Act

The Ordinance supplements the Reactor Regulation Act with requirements on specific matters. For example, it clarifies that establishment of both land-based reactors and reactors installed on ships requires a license from the Government; it prescribes the plant systems, structures and components that are subject to periodic inspections pursuant to the Act; it establishes the number and the qualifications of Nuclear Safety Inspectors of the METI.

- **Electricity Business Act and Associated Ordinances**

Electricity Business Act

The stated objectives of the Electricity Business Act are, through regulation of construction, maintenance and operation of electric facilities, to protect the interests of the electricity users; to achieve sound development of the electricity business; to assure public safety and to promote environmental preservation. The Electricity Business Act applies to all forms of electricity generation installations, i.e., nuclear, hydro, fossil. The Electricity Business Act regulates nuclear power plants in Japan as one type of electricity business.

Provisions of the Act specific to nuclear power reactor are primarily on inspection. It requires the METI commission the JNES to conduct various inspections for nuclear reactors, i.e., pre-operation (pre-service) inspection, periodic facility inspection, safety management inspection, fuel element inspection, welding inspection. It mandates that nuclear fuel must be inspected for approval by the METI before the fuel assemblies can be used in a reactor. In addition, the Act contains reporting requirements, prescribes penalties for violation of the Act, and ensures access of the regulator and its inspectors to the regulated facilities and facility information, etc.

The Act does not contain specific design or safety requirements.

Rules for the Enforcement of the Electricity Business Act

The Ordinance provides specific processes and procedures to implement the Electricity Business Act. With respect to nuclear power reactors, the Ordinance supplements the Act with requirements on conducting the inspections required by the Act. It defines the scope (and timing) for the inspections required by the Act and delineates the inspection activities carried out by the JNES. It provides that the METI shall conduct its own inspections and coordinate its inspection activities with JNES. It requires that inspection methods, instructions and procedures be developed and available for performing inspections. It contains procedural provisions on reporting and on the interactions between different organizations, i.e., METI, JNES, NSC, and licensees.

The Ordinance does not contain specific design and safety requirements.

Ministerial Order for Performing Inspection, Etc. by the JNES (Pursuant to the Electricity Business Act Law)

Pursuant to the Act, the JNES inspectors must be certified for performing the inspections prescribed in the Act for nuclear reactors. This Ministerial Order delineates qualifications for certified inspectors, including electric structure examiners, welding safety management reviewers, and periodic safety management reviewers.

Ordinance of Establishing Technical Standards for Nuclear Power Generation Equipment

The Ordinance prescribes reactor design requirements and reactor safety standards.

- **Act on Special Measures Concerning Nuclear Emergency Preparedness and Associated Ordinance**

This Act prescribes special measures for nuclear preparedness and response. It addresses organizational elements, communication/notification, radiation monitoring, offsite emergency response center, emergency declaration, protective measures, restoration measures, etc.

This Act does not cover severe accident management.

- **NSC Regulatory Guides**

Appendix 1 provides a list of the NSC Regulatory Guides. The Guides were grouped according to siting, design, safety evaluation, dose limits and accident management. The siting group contains one regulatory guide (L-ST-I.0). The regulatory guide groups for design, safety evaluation, and dose limits each contain a general guide (L-DS-I.0, L-SE-I.0, and L-RE-I.0) supported by the guides on more specific topics. The accident management group contains one guide L-AM-II.01. It is the only regulatory document on accident management. The NSC decision on severe accident management was attached to this document. [8] In the following, summaries are provided for Level I guides and L-AM-II.01:

  - Regulatory Guide for Reviewing Nuclear Reactor Site Evaluation and Application Criteria (L-ST-I.0)
  - Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities (L-DS-I.0)
  - Regulatory Guide for Reviewing Classification of Importance of Safety Functions of Light Water Nuclear Power Reactor Facilities (L-DS-I.01)
  - Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities (L-DS-I.02)
  - Regulatory Guide for Reviewing Fire Protection of Light Water Nuclear Power Reactor Facilities (L-DS-I.03)
  - Regulatory Guide for Reviewing Radiation Monitoring in Accidents of Light Water Nuclear Power Reactor Facilities (L-DS-I.04)
  - Regulatory Guide for Evaluating Safety Assessment of Light Water Reactor Facilities. (L-SE-I.0)
  - Regulatory Guide for Evaluating Core Thermal Design of Pressurized Water Cooled Nuclear Power Reactors (L-SE-I.01)
  - Regulatory Guide for Evaluating Emergency Core Cooling System Performance of Light Water Power Reactors (L-SE-I.02)
  - Regulatory Guide for Evaluating Reactivity Insertion Events of Light Water Nuclear Power Reactor Facilities (L-SE-I.03)

- Regulatory Guide for Evaluating Dynamic Loads on BWR MARK-I Containment Pressure Suppression Systems (L-SE-I.04)
- Regulatory Guide for Evaluating Dynamic Loads on BWR MARK-II Containment Pressure Suppression Systems (L-SE-I.05)
- Regulatory Guide for Meteorological Observation for Safety Analysis of Nuclear Power Reactor Facilities(L-SE-I.06)
- Regulatory Guide for the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities (L-RE-I.0)
- Regulatory Guide for Reviewing Evaluation of Dose Target for Surrounding Area of Light Water Nuclear Reactor Facilities (L-RE-I.01)
- Guide for Radiation Monitoring of Effluent Released from Light Water Nuclear Power Reactor Facilities (L-RE-I.02)
- Accident Management for Severe Accidents at Light Water Power Reactor Installations (L-AM-II.01)

The primary focus of the discussion below is on the scope and key concepts of these guides and not on technical details.

Regulatory Guide for Reviewing Nuclear Reactor Site Evaluation and Application Criteria (L-ST-I.0)

This guide was used in evaluation of the suitability of reactor site conditions. The guide provides for a site exclusion zone, a low population zone, and sites being located away from densely populated regions. Dose acceptance criteria are specified for these areas and are used to assess the suitability of a site. L-SE-I.0, "Regulatory Guide for Evaluating Safety Assessment of Light Water Reactor Facilities," provides dose assessment guidelines. In accordance with these two guides, the suitability of a site is essentially based on the estimated doses from selected design basis accidents.

Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities (L-DS-I.0)

The purpose of this guide is stated in the very beginning of the guide: "This guide provides the basis of the judgment for adequacy of the design to ensure safety at the Safety Review related to the application for the establishment license (includes the application of alteration of an establishment license) of the individual light water nuclear power reactor (hereafter referred to as "LWR")."

The guide contains fifty nine general "requirements" or "guidelines." These guidelines are essentially equivalent to the General Design Criteria of Appendix A to 10 CFR Part 50 with variations in arrangement of items and use of language or terms (which may be due to translation).

The guide also includes a commentary section, which provides clarification or interpretation of selected guidelines.

Regulatory Guide for Reviewing Classification of Importance of Safety Functions of Light Water Nuclear Power Reactor Facilities (L-DS-I.01)

This guide prescribes a safety classification scheme, which includes three safety classes (Class 1, Class 2 and Class 3) and one Nonsafety Class. The guide also groups the SSC performing safety functions into two categories: abnormality prevention systems (PS) and abnormality mitigation systems (MS). PS SSCs include those whose failures may lead to potential release of radioactive materials. MS SSCs are those designed to provide safety shutdown, emergency core cooling and containment functions. Thus, PS-1, PS-2 and PS-3 are used to denote Class 1, Class 2 and Class 3 PS SSCs; and MS-1, MS-2, and PS-3 denote Class 1, Class 2 and Class 3 MS SSCs respectively.

The guide provides criteria (or definition), examples, and considerations for classifying SSCs as well as design considerations for SSCs of different safety classes. Nonetheless, the guide does not specify applicable industry codes and standards for SSCs of different Safety Classes.

Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities (L-DS-I.02)

This document provides guidance on seismic classification, seismic hazard analysis (determining design basis ground motion), and seismic design criteria (e.g., load combinations, stress limits). The guide also calls for attention to tsunami induced by earthquake. Seismic hazard analysis is of deterministic nature.

Regulatory Guide for Reviewing Fire Protection of Light Water Nuclear Power Reactor Facilities (L-DS-I.03)

This guide provides that design of nuclear power plants incorporate provisions to prevent occurrence of fires, to detect/extinguish fires should they occur and to mitigate the consequences from plant fires. A fire protection program is required for plant operation. But there is not guidance given on the fire protection program. The guide requires that the function of the fire protection systems not be severely affected by a major earthquake and design take into consideration of earthquake induced fires.

Regulatory Guide for Reviewing Radiation Monitoring in Accidents of Light Water Nuclear Power Reactor Facilities (L-DS-I.04)

This regulatory guide provides guidance used in reviewing the radiation monitoring systems. The radiation monitoring systems are used to measure radiation levels and radioactive releases during and after an accident, and to provide information on integrity of radiation barriers and condition of the plant and the surrounding area. In accordance to this guide, the radiation monitoring systems are classified into Category 1, Category 2 and Category 3. Category 1 systems are those used to monitoring integrity of radiation barriers. Category 2 systems are those used to provide information on changing status of the accident and for dose estimates. Category 3provides information supplemental to the information obtained by Category 1 and Category 2 systems. The guide includes design considerations and requirements for the Category 1 and Category 2 systems

Regulatory Guide for Evaluating Safety Assessment of Light Water Reactor Facilities (L-SE-I.0)

This guide is used in licensing review to evaluate the adequacy of the safety assessment in the application for permission to install (or modify) LWRs. The licensing review is required to verify the safety assessment of the application fully meet the requirements specified in this guide. This guide requires two types of safety assessments: safety design assessment and site assessment. Guidance and acceptance criteria are provided for conducting these assessments.

“Safety design assessment” evaluates the response of the plant systems to selected anticipated operating occurrences and postulated accidents. The scope is similar to that of Chapter 15 “Transient and Accident Analyses” of the SAR in U.S.

“Site assessment,” which determines the suitability of a site, evaluates the radiological consequences that could result from major accidents (selected postulated or design basis accidents. The scope of “site assessment” is similar to “Design Basis Accident Radiological Consequence Analysis” of Chapter 15 of the SAR in the U.S. However, in the U.S., the primary purpose of design basis accident radiological analysis of Chapter 15 is to assess the plant design (vs. site suitability), in particular, the performance of the containment systems.

This guide is a general guide on safety evaluation. Next are introduced a number of supporting guides addressing specific issues or design basis events.

Regulatory Guide for Evaluating Core Thermal Design of Pressurized Water Cooled Nuclear Power Reactors (L-SE-I.01)

This guide provides acceptable methods for developing departure from nucleate boiling (DNB) correlations, and establishes DNB ratio (DNBR) acceptance criterion for transient analysis.

Regulatory Guide for Evaluating Emergency Core Cooling System (ECCS) Performance of Light Water Power Reactors (L-SE-I.02)

This guide establishes acceptance criteria for evaluation of ECCS performance. The acceptance criteria on cladding temperature, cladding oxidation, hydrogen generation and coolable core geometry are the same as those in 10 CFR 50.46.

This guide provides guidance on loss of coolant accident (LOCA) analyses (i.e., break sizes and locations, initial conditions, boundary conditions, and actuation of safety systems and operator actions credited), and on evaluation models (i.e., code models or correlations, model development and assessment).

Regulatory Guide for Evaluating Reactivity Insertion Events of Light Water Nuclear Power Reactor Facilities (L-SE-I.03)

Design basis event analyses include analysis of the plant response to reactivity anomalies, i.e., uncontrolled control rod withdrawal, control rod ejection in Pressurized Water Reactors (PWRs) or control rod drop in BWRs. This Guide establishes acceptance criteria for this type of events with regard to maximum fuel enthalpy and reactor coolant system pressure. The Guide also provides guidelines on conducting the analyses, i.e., initial conditions, assumptions, plant inputs, etc.

Regulatory Guide for Evaluating Dynamic Loads on BWR MARK-I Containment Pressure Suppression Systems (L-SE-I.04)

This guide provides for acceptable methods for evaluating dynamic loads on BWR Mark-I containment pressure suppression caused by LOCAs and by actuation of safety valves. This Guide does not address challenges that may arise from severe accidents.

Regulatory Guide for Evaluating Dynamic Loads on BWR MARK-II Containment Pressure Suppression Systems (L-SE-I.05)

This Guide provides for acceptable methods for evaluating dynamic loads on BWR Mark-II containment pressure suppression caused by LOCAs and by actuation of safety valves. This Guide does not address challenges that may arise from severe accidents.

Regulatory Guide for Meteorological Observation for Safety Analysis of Nuclear Power Reactor Facilities (L-SE-I.06)

This Guide provides guidance on the meteorological measurements, methods for data analysis and atmospheric dispersion model. The purpose is to calculate atmospheric dispersion coefficients used in dose assessments. The Guide does not address the meteorological monitoring requirements to support radiological emergency response.

In the U.S., the meteorological monitoring system is required to be capable of providing near real-time meteorological information for use in dose projections during radiological emergencies.

Regulatory Guide for the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities (L-RE-I.0)

This guide establishes an annual dose target of 50µSv for the population living in the vicinity of a nuclear power station during normal plant operations. The associated guides L-RE-I.01 and L-RE-I.02 provide detailed guidance on determining annual releases and models and methods for calculating annual doses received by individuals living in the vicinity of a nuclear power plant.

Accident Management for Severe Accidents at Light Water Power Reactor Installations (L-AM-II.01)

This document was published in 28 May 1992. This document provides an overview of the status of development and implementation of accident management (AM) measures in Japan, describes the technical basis information which takes into account development in other countries such as US, Europe, and Japan), and makes recommendations for further development accident management measures in Japan. The document attempts to use Level 1 and Level 2 PSA results from the plants in Japan to show the effectiveness of proposed AM measures and the plants meeting probabilistic safety goals.

The AM measures are divided into two groups: Phase I measures and Phase II measures. The Phase I measures are described as various operations to restore safety functions. Although not explicitly stated, it seems that the Phase I measures has the primary goal of preventing core damage, which is the goal for the EOP in the U.S. A list of the Phase I measures is provided, which are characterized as manual operations to perform or restore plant functions, i.e., restoring AC power in the event of SBO, manual scram of reactor or boron injection, manual start ECCS injection, feed and bleed, etc.

The Phase II measures are described as those taken to cool a damaged core and to prevent containment damage from overpressure by venting the containment. The stated objectives for the Phase II measures are to protect the containment in the severe accident conditions and mitigate radioactive release into environment. At the time of the publication of this report, the Phase II measures were being studied by the licensees.

NSC's decision on severe accident management was made on May 28, 1992 and amended on October 20, 1997. It determined that under the existing regulatory framework "The possibility of severe accidents is sufficiently low due to these measures, to the extent that such accidents could not occur from an engineering viewpoint, and thus the risk from reactor facilities is considered to be sufficiently low." The purpose of developing severe accident management measures was "significant in further reducing the risk, which is already low." Therefore, effective accident management should be developed by licensees on a voluntary basis and properly implemented in the event of an emergency.