Convention on Nuclear Safety
National Report of Japan
for the Fourth Review Meeting

September, 2007

Government of Japan

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D. Safety of Installations

Annexes

Acronym and Abbreviation Used in this Report

ABWR	Advanced Boiling Water Reactor
ACNRE	Advisory Committee for Natural Resources and Energy
AEC	Atomic Energy Commission
AESJ	Atomic Energy Society of Japan
ALARA	as low as reasonably achievable
ANRE	Agency of Natural Resources and Energy
APWR	Advanced Pressurized Water Reactor
BSS	Basic Safety Standards
BWR	Boiling Water Reactor
Comprehensive Check	Comprehensive Check of Electric Power Facilities
DNB	Departure from Nucleate Boiling
Dose Limit Notification	Notification for Dose Limits on the basis of the Rules for Commercial Power Reactors
Electric facility	Facility or equipment that is installed for power generation, transformation, supply, distribution and utilization
ERSS	Emergency Response Support System
FY	Fiscal Year
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
INES	International Nuclear Event Scale
INPO	Institute of Nuclear Power Operations
IRRS	Integrated Regulatory Review Service
JAEA	Japan Atomic Energy Agency
JANTI	Japan Nuclear Technology Institute
JCO Criticality Accident	Criticality Accident at JCO Co. Uranium Fuel Fabrication Facility
JEA	Japan Electric Association
JEAC(G)	Japan Electric Association Code (Guideline)
JNES	Japan Nuclear Energy Safety Organization
JPDR	Japan Power Demonstration Reactor
JSME	Japan Society of Mechanical Engineers
LCO	Limiting Conditions for Operation
METI	Ministry of Economy, Trade and Industry
MEXT	Ministry of Education, Culture, Sports, Science and Technology
Minister of METI	Minister of Economy, Trade and Industry
Minister of MEXT	Minister of Education, Culture, Sports, Science and Technology

MITI	Ministry of International Trade and Industry (METI at present)
MLIT	Ministry of Land , Infrastructure and Transportation
Monju	Prototype fast breeder reactor owned by JAEA
MOX	Mix Oxide
NISA	Nuclear and Industrial Safety Agency
NISS	Nuclear and Industrial Safety Subcommittee
NSC	Nuclear Safety Commission
NS Network	Nuclear Safety Network
NTC	Nuclear Power Training Center
NUSS	Nuclear Safety Standards, IAEA
OECD/NEA	Organization of Economic Co-operation and Development/Nuclear Energy Agency
OSART	Operational Safety Assessment Review Team
PAZ	Precautionary Action Zone
PSA	Probabilistic Safety Assessment
PSR	Periodic Safety Review
PWR	Pressurized Water Reactor
QA	Quality Assurance
R & D reactor	Power reactors at the stage of research and development
Reactor Regulation Law	Law on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors
Regulatory Guide for Reactor Siting	Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and Application Criteria
Regulatory Guide for Reviewing Safety Assessment	Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities
Regulatory Guide for Reviewing Seismic Design	Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities etc
Regulatory Guide for Safety Design	Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities
SCC	Stress Corrosion Cracking
Special Law for Nuclear Emergency	Special Law of Emergency Preparedness for Nuclear Disaster
RM	Relationship Management
SCAP	Stress Corosion Cracking and Cable Ageing Project
SSC	Structures, Systems and Components
UPZ	Urgent Protective Action Planning Zone
WANO	World Association of Nuclear Operators

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Preface

1. Preparation of the Report

Organizations involved in the report preparation

This report was prepared by the Nuclear and Industrial Safety Agency (hereinafter referred to as “NISA”) of the Ministry of Economy, Trade and Industry (hereinafter referred to as “METI”). It was prepared in consultation with the relevant government agencies, as well as in the support of the Incorporated Administrative Agency Japan Nuclear Energy Safety Organization (hereinafter referred to as “JNES”) *and with the cooperation of Japan Nuclear Technology Institute*. Moreover, the report was deliberated by the Nuclear and Industrial Safety Subcommittee of the Advisory Committee for Natural Resources and Energy under METI and it also reflects the opinion of the Nuclear Safety Commission.

Items to be considered in preparation of this report

In preparation of this report, special attention was paid to respond appropriately to the items for which reporting was recommended in the “Summary Report” of the 3rd Review Meeting(RM), questions on the previous report of Japan raised by other Contracting Parties, and also items presented as questions or comments to the reporting of Japan in the 3rd RM.

Fundamental policies of the descriptions of this report

In composing this report, results of implementation by the national government and industry were inserted corresponding to each article of the previous report, and description of it without any alteration was concisely duplicated so that the whole aspect of the present framework for ensuring safety could be understood. However, concerning the operating experience, which is actual evidence for ensuring the safety and improvement of nuclear installations, only operating experience of three years after the 3rd review meeting (in April 2005) were provided as the subject of this report, so as to avoid duplication of the last report.

Description style of this report

In description of style, content revisions from the last report are shown in italics and where no changes were made the script remains as in the original (including editorial amendments). This is so that the reader can easily identify the revisions corresponding to the three years after the 3rd RM.

Under the legislative and regulatory framework of Japan, nuclear installations in the scope of this Convention (land-based commercial nuclear power stations) correspond to commercial power reactors and power reactors at the stage of research and development, of which the safety regulations for these two types of nuclear installations are fundamentally the same. For this reason, detailed description in this report is focused on the commercial power reactors as examples, which have plenty of experience with siting, design, construction and operation, etc.

In addition, the response to “SYNOPSIS OF THE RELEVANT IAEA SAFETY REQUIREMENTSTATEMENTS REFLECTING THE ISSUES ADDRESSED BY ARTICLES 6 TO 19 OF THE CONVENTION ON NUCLEAR SAFETY” is provided in Appendix 5 of this report.

2. Current Status of Nuclear Energy Utilization in Japan

(1) Situation of utilization of nuclear energy in general

Currently, there are a total of 56 nuclear installations in Japan in the scope of the Convention, which include 55 units in operation and one unit under construction that has attained criticality. They have become the main power source supplying about 30.6% of the electric power production in the 2006 fiscal year.

In the Framework for Nuclear Energy Policy decided by the Cabinet in October 2005, it is stated that the basic concepts of nuclear power generation is 1) aiming to maintain or increase the current level (around 30 to 40%) of electric power production in 2030 or later, 2) promote nuclear fuel cycles, 3) utilize fast breeder reactors, etc. METI deliberated concrete measures to realize these basic concepts, and adjusted the Nuclear Power National Plan in August 2006. This plan was determined to be promoted intensively and to 1) realize construction of new and additional nuclear power stations even under the circumstances of deregulation of the power industry, 2) utilize the established nuclear power stations by ensuring safety into its major premises, 3) put the Fast Breeder Reactor Cycle into commercial use in as early a date as possible, 4) secure quantity and quality of the technical human resources to sustain the next generation, 5) support industries for the export of the nuclear power generation facility, 6) participate positively in making an international framework with the aim to coexist with expansion of nuclear power generation and nuclear nonproliferation, etc.

(2) Current Status of Nuclear installations in Japan

As of August 2007, there are a total of 56 nuclear installations in Japan in the scope of the Convention, which include 55 units in operation and one unit under construction that has attained criticality.

In FY 2006, the total capacity of 49.47GWe of nuclear power generation accounted for about 20.7 percent of the nation’s total capacity of electricity generation, and nuclear power generated 304.5 billion kWh of electricity that was about 30.6 percent of 995.9 billion kWh electricity generated in Japan. The average annual capacity factor of nuclear power plants is 70%. The average unscheduled shut-down frequency over FYs 2004 to 2006 was as small as 0.5 times per reactor-year.

3. International Activities for Ensuring Safety of Nuclear installations

Recognizing that international cooperation is essential for ensuring safety of nuclear installations, Japan has been promoting multilateral and bilateral cooperation.

As for multilateral cooperation, Japan has actively participated in the information exchange on safety regulation and study of issues concerning nuclear safety in the IAEA, the Organization for Economic Cooperation and Development/ Nuclear Energy Agency (OECD/NEA), and furthermore, the International Nuclear Regulators Association (INRA) and the International Nuclear Safety Group (INSAG). The obtained information and the study results have been utilized for the substantial regulation of Japan.

A new international activity during this reporting period is that Japan participated in the second review meeting of the "Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management." This led Japan to have opportunities to receive peer-reviews of the contracting parties on the safety of radioactive waste management etc. as well as on the safety of nuclear installations. As for bilateral cooperation, Japan has been exchanging regulatory information on nuclear safety with the regulatory authorities of China, France, Korea, Sweden, the U.K. and the U.S.A. under the bilateral agreement, has shared its knowledge and experience with them and has been making efforts to enhance the safety of each other's nuclear power plants.

On the other hand, licensees are also cooperating actively in managing the World Association of Nuclear Operators (WANO) Tokyo Center, in order to enhance the safety and reliability of the operation of nuclear power stations through the information exchange between utility operators of Asian nations.

Units 3 and 6 of the Kashiwazaki-Kariwa Nuclear Power Station of TEPCO invited the OSART of IAEA in November 2004.

4. Important Matters during Reporting Period

Revision of the Regulatory Guide for Reviewing Seismic Design

The Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities to new nuclear reactors was revised by the Nuclear Safety Commission on September 19, 2006. It requires a higher level of seismic safety resulting from the alternation of the formulation and evaluation method of earthquake ground motion etc. NISA, deciding that the seismic safety should be checked based on the new Guide for the existing nuclear installations, instructed the operators (the licensees of all the nuclear power reactors) to conduct the seismic safety evaluation and to report the results to on September 20, 2006.

Activities of the Task Force on Inspection System

While promoting firm establishment of the new inspection system which was introduced in October 2003, the Task Force on Inspection System had resumed in November 2005, in order to ensure the safety of the aged nuclear power station, and the report, "Improvement in the Inspection System for Nuclear Power Generation Facilities," was issued in September 2006. The main points are (1) to encourage the licensees to conduct their maintenance activities *reflecting the individual characteristics of each nuclear installation with emphasis on aging management, (2) to strengthen the operational safety activities by including the inspection during reactor operation in addition to the activities performed during reactor shutdown, and (3) to include thorough remedial actions into the inspection system and to avoid non-conformity by the licensees*

Stipulating performance-code of the technical standard

Heretofore, the technical standard which defines technical requirements of the nuclear installations included and not only items_related to performance but also provisions which define detailed specifications. In order to enable flexible response to technical innovations or latest knowledge, the technical standard should be focused only to the performance requirements necessary for ensuring safety, and the academic society and association standards should be utilized as detailed specifications which meet the performance requirements.

Promotion of safety research

NISA established the Nuclear Safety Infrastructure Subcommittee for ensuring nuclear safety under the Nuclear and Industrial Safety Subcommittee in July 2006, and developed the framework for industries and regulatory organizations to perform planning, implementation and evaluation of the nuclear safety infrastructure study program. The subcommittee prepared the roadmap of the nuclear safety infrastructure study program so that the program could be implemented in a well-planned and efficient manner towards ensuring safety. In addition, considering that many research facilities for nuclear safety studies are on a global scale in the crisis of possible shutdown in recent years, it has been proposed that the Japan Materials Testing Reactor (JMTR) of the Japan Atomic Energy Agency is to be positioned strategically as an essential facility for the safety infrastructure study program. It is decided that the international joint study should also be strongly promoted. For example, the special fund business SCAP project, which reviews stress corrosion cracking (SCC) and the aging of cables, was entrusted from Japan to OECD/NEA.

Establishment of the Japan Nuclear Technology Institute

In April 2005, the nuclear industry (licensees, nuclear fuel fabricators, plant manufacturers, etc.) established the Japan Nuclear Technology Institute (JANTI) who is actively encouraged to further improve the voluntary safety activities by the industry and to share and upgrade the safety culture. Objectives of JANTI are (1) to act for the safety-culture dissemination in support of licensees, (2) to perform peer review to the activities of licensees, (3) to analyze and evaluate information from the Nuclear Information Archives "NUCIA" etc., and (4) to evaluate the safety culture of the members objectively with the third-party's view-points .

Measures for aging management

Implementation of Technical Assessment of Aging and development and implementation of the Ten Years Maintenance Program were defined as requirements for aged nuclear installations *in laws and ordinances of October 2003. NISA has been confirming their implementing status.*

In order to enhance the response to aged nuclear installations, NISA formed the Technical Information Coordination Committee to share domestic and foreign technical information among the industrial society, the academic society and the government organizations, and to utilize them effectively. In addition, the Special Committee was established under the Atomic Energy Society of Japan, with the participation of NISA, JNES, universities, research organizations, power utilities, nuclear plant manufacturers and plant engineering companies. And a road map to the measures for aging management and life-extension with safe-operation of light water reactors was developed by the Special Committee.

Securing human resources in the nuclear field

In Japan, sustaining human resources becomes one of the key issues, due to the retirement of expert engineers and decreasing construction opportunities of nuclear installations in recent years. Consequently, the securing human resources in the future have been reviewed and the national government and the industrial and academic societies are working together for personnel training and succession of technical tradition.

Comprehensive check of the electric power facilities

The data falsification in the hydroelectric power station of The Chugoku Electric Power Co., Inc. in October 2006 led to the implementation of the Comprehensive Check of electric power facilities including hydroelectric power, thermal power and nuclear power stations. As a result, 316 cases on the whole, of which 98 cases were of nuclear power, of falsification and procedural defect were discovered and reported by 12 electric power companies in March 2007.

METI responded to the cases of nuclear installations by (1) invoking the order to the nuclear power stations of Tokyo Electric Power Co., Inc., Hokuriku Electric Power Co., The Chugoku Electric Power Co., Inc., and The Japan Atomic Power Co. to amend their Operational Safety Programs, (2) moving forward or extending the Periodic Inspections of the nuclear power stations of these electric power companies, (3) implementing special Operational Safety Inspections and Periodic Inspections with the assignment of the Special Nuclear Installation Management Supervisor, (4) promoting information sharing on international accidents and troubles, and (5) directing planning of an action plan for the prevention of recurrence.

The action plan for the prevention of recurrence was submitted from each electric power company to METI. The main base examples include participation of the executive officers, intensive education program on operational safety, recordkeeping of alarm typewriters, cooperation with free access of Nuclear Safety Inspectors, enhancement of independency of the chief engineer of reactors, promotion of information sharing through the NUCIA etc., notification of deviations from limiting conditions for operation, information sharing between electric power companies, etc. NISA determined that these action plans are appropriate in general, and they will verify the situation through the Operational Safety Inspection etc. in the future. Moreover, NISA will also revise the required legislation corresponding to enforcement *of the action plan.*

Receiving the report from NISA about the falsification and concealment concerning nuclear facilities, the NSC has investigated and has tried to understand the whole story. Responding to the actions taken by METI, the NSC determined "Actions to take, concerning the malicious conduct of unreported alteration and concealment at nuclear facilities" (See Attachment to the report on Article 6) on April 23, 2007.

Invitation of IRRS

NISA and the NSC requested the Integrated Regulatory Review Service (IRRS) of the IAEA in 2007. The main objectives of the conventional review by the International Regulatory Review Team (IRRT) was the verification of conformance with the safety standards of the IAEA, the IRRS to Japan included, in addition to that, extensive policy dialogue with the participating senior regulators about current issues on regulation. A preparatory meeting was held in February 2007 and the IRRS plenary meeting was held in June of the same year.

Niigataken Chuetsu-oki Earthquake

On July 16, 2007, a magnitude 6.8 earthquake occurred offshore Chuetsu, in Niigata Prefecture about 16 km away from the Kashiwazaki Kariwa nuclear power station of Tokyo Electric Power Co., Inc. Units 2, 3, 4 and 7, which were in operation, automatically shut down with the scram signal due to this earthquake. Units 1, 5 and 6, which were in shutdown for the Periodic Inspection.

After this earthquake, a very little amount of radioactive material was released through the discharge path to the sea in Unit 6 and from the vent stack to the air in Unit 7. The released amount of activity was $9x10^4$ *Bq (*$2x10^{-10}$ *Bq/*cm^3*, limit defined by law: 0.2 Bq/*cm^3*) to the sea and* $4x10^8$ *Bq (*$2x10^{-7}$*mSv, limit defined by law: 1 mSv/year) into the air, respectively. It was much less than the limit defined by law, and resulted in no significant radiation exposure to the public. Tokyo Electric Power Co., Inc. is investigating the impact on the equipment, etc. in the power station by this earthquake (as of the end of July 2007).*

NISA accepted the IAEA investigation team, in order to achieve international information sharing of the impact of this earthquake on the nuclear power station.

Moreover, the Investigation/Response on Offshore Chuetsu Earthquake Subcommittee was established under the Nuclear and Industrial Safety Subcommittee of the Advisory Committee for Natural Resources and Energy in order to investigate the specific impact on the Kashiwazaki Kariwa nuclear power station by this earthquake and also examine issues to and responses to be addressed to improve safety after this earthquake by the national government and the licensee.

The NSC commissioners, immediately after the earthquake, had visited the site and carefully observed the plants. The NSC received the report from NISA, held a meeting of the "Investigation Project Team on Seismic Safety of Nuclear Facilities" and worked for fact *finding, trying to deliberate future actions. On July 30, 2007, the NSC determined the viewpoint about the impact of this earthquake and the action plans and published the report "The NSC views on, and future actions to take for, the impacts due to the Niigata-ken Chuetsu-oki Earthquake in 2007"(See Annex 4 of this National Report).*

A. General Provisions

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Article 6 Existing Nuclear Installations

Each Contracting Party shall take the appropriate steps to ensure that the safety of nuclear installations existing at the time the Convention enters into force for that Contracting Party is reviewed as soon as possible. When necessary in the context of this Convention, the Contracting Party shall ensure that all reasonably practicable improvements are made as a matter of urgency to upgrade the safety of the nuclear installation. If such upgrading cannot be achieved, plans should be implemented to shut down the nuclear installation as soon as practically possible. The timing of the shut-down may take into account the whole energy context and possible alternatives as well as the social, environmental and economic impact.

There are a total of 56 nuclear installations in Japan in the scope of the Convention as of the end of June 2007, which include 55 units in operation and one unit under construction that has attained criticality.

Since the previous report, Unit 5 of the Hamaoka Nuclear Power Station, Chubu Electric Power Co. Unit 1 of the Higashidoori Nuclear Power Station, Tohoku Electric Power Co., Inc. and Unit 2 of the Shika Nuclear Power Station, Hokuriku Electric Power Co., which were under construction, have attained criticality and they enter the scope of the Convention. The decommissioning plan of Fugen that had been under preparation for decommissioning has been authorized and the plant has come out of the scope of the Convention.

Matters on ensuring safety for existing nuclear installations are described in the following.

In addition, for reference, a list of accidents and failures reported since the previous report is shown in Table 6-1 together with the INES evaluation results.

6.1 Existing Nuclear Installations in the Scope of this Convention

There are a total of 56 existing nuclear installations in Japan in the scope of the Convention as of the end of *June 2007.* The breakdown is shown in the following:

Breakdown of Nuclear Installations in the Scope of the Convention

Type		Status	Number of units	Remarks
Commercial power reactor	Boiling water reactor (BWR)	in operation	*32*	
	Pressurized water reactor (PWR)	in operation	23	
Power reactor in research and development stage		under construction*	1	Monju

* Unit under construction, which has attained criticality

The existing nuclear installations are listed in Annex 1, and their locations are shown in Fig.

6-1.

6.2 The Safety of Existing Nuclear Installations

Major studies and measures implemented for ensuring safety of existing nuclear installations since the previous report are; 1) measures taken for the secondary-system pipe rupture accident at Unit 3 of the Mihama Power Station (the event was partly reported in the report for the 3rd review meeting and at the review meeting), 2) measures taken for the crack etc. of the hafnium-plate-type control rods of BWR nuclear power plants, 3) measures taken for the strainer plugging issue in BWR and the containment sump clogging issue in PWR, 4) measures taken for falsification of the test data on the feed water flow instrumentation, 5) safety review and assessment of power plants loading mixed-oxide fuel partially in the core, 6) measures taken for faulty design of the ABWR turbine blades, 7) check on the status of implementing the measures against sodium leakage of the Monju power station, 8) check on seismic safety of the Onagawa Nuclear Power Station, 9) measures taken for the partial omission of Periodic Inspections conducted by the Incorporated Administrative Agency, Japan Nuclear Energy Safety Organization, 10) implementation of the Comprehensive Check on power generation facilities, etc.

These cases are summed up as below.

(1) Measures Taken for the Secondary-System Pipe Rupture Accident at Unit 3 of the Mihama Power Station

1) Summary of the Accident and Investigation Results

On August 9, 2004, at Unit 3 of the Mihama Power Station, the Kansai Electric Power Co., Inc (hereinafter referred to as "KEPCO")., a main condensate pipe ruptured during the steady-state of operation at rated thermal power, discharged secondary-system steam in the turbine building. Workers near the ruptured opening were exposed to the discharged steam, and five persons died and six persons got injured.

The Nuclear and Industrial Safety Agency (hereinafter referred to as "NISA"), aiming at an investigation of cause and prevention of recurrence of a similar accident, established an accident investigating committee immediately after the accident and compiled the final report on March 30, 2005.

The direct cause of the accident was wall thinning of the pipe concerned due to erosion and corrosion, overlooked for years since KEPCO and Mitsubishi Heavy Industries (hereinafter referred to as "MHI") had not included the ruptured pipe section in the checklist, and the root cause was the inadequacy of the maintenance management and quality assurance system of these business operators.

The efforts made by NISA and the Nuclear Safety Commission (hereinafter referred to as "the NSC") for this case is described in the following.

2) Action by the Nuclear and Industrial Safety Agency

(a) Establishment of the Accident Investigating Committee (the Investigating Committee for the Secondary-System Pipe Rupture Accident at Unit 3 of the Mihama Power Station)

On August 10, 2004, the Accident Investigating Committee was established under the Reactor Safety Subcommittee (it has held the meeting 11 times henceforth).

On September 27, 2004, the interim report was compiled and the special Operational Safety Inspection by NISA and the strict Audit of Licensee's Periodic Check System by the incorporated administrative agency, Japan Nuclear Energy Safety Organization (hereinafter referred to as "JNES") of the licensee were implemented based on the report.

On March 30, 2005, the final report was compiled.

On March 28, 2006, the implementation status of the measures to prevent recurrence was evaluated (the Investigating Committee meeting was held).

(b) Administrative Actions against Licensees concerning the Secondary-System Pipe Rupture Accident at Unit 3 of the Mihama Power Station

- *In September 2004, under the name of the Minister of Economy, Trade and Industry (hereinafter referred to as the "Minister of METI"), a severe warning in writing, the order for conformity to technical standards concerning Unit 3 of Mihama Power Station and a notification to downgrade the evaluation of the Audit of Licensee's Periodic Check System were issued to the Kansai Electric Power Co., Inc.*

- *The Minister of METI directed JNES to perform a strict review at the Audit of Licensee's Periodic Check System of the nuclear power stations of the Kansai Electric Power Co., Inc.,*

- *The Director General of NISA issued a notification to licensees other than the KEPCO, to inform his strong expectation to reflect the accident in their preventive actions.*

(c) Implementation of the special Operational Safety Inspection (from the 3rd calendar quarter, 2004 to the 4th calendar quarter, 2005)

In order to comprehensively confirm the implementation of the measures to prevent recurrence, the inspection was carried out with the inspection system fully enhanced by inter-changing the inspectors of the Nuclear Safety Inspector's Offices of NISA in the Wakasa area and dispatching the Nuclear Safety Inspectors of NISA. Moreover, on July 1, 2005, the Regional Nuclear Safety General Manager (in charge of the Wakasa area) was newly assigned, and the subsequent inspections were conducted with the General

Manager involved in it.

(d) Confirmation by the Usual Operational Safety Inspection (from the first quarter, 2006)

During the usual Operational Safety Inspection conducted by the Mihama Nuclear Safety Inspector's Office of NISA, the implementing status of the measures to prevent recurrence of the accident has been confirmed. In addition, for the matters beyond the responsibility of the Mihama Power Station, and which is under the responsibility of the Nuclear Power Division, KEPCO Head Office, confirmation was carried out by the inspectors from three offices; namely, the Takahama Nuclear Safety Inspectors Office, Mihama Nuclear Safety Inspectors Office and Ohi Nuclear Safety Inspectors Office, and the Regional Nuclear Safety General Manager.

(e) Confirmation by the Nuclear Maintenance Confirmation Committee (established inside KEPCO on June 17, 2005)

NISA participates as an observer in the Nuclear Maintenance Confirmation Committee, which is an internal audit system of KEPCO, and consists of third-party members, and checks implementation of the measures to prevent recurrence.

(f) Confirmation of the measures to prevent recurrence of MHI

NISA has checked the status of the measures to prevent recurrence of MHI by periodic hearing (about once / quarter) since August 29, 2005.

The measures to prevent recurrence of MHI has been already implemented, since it is required to steadily promote the evaluation and improvement of activities, NISA keeps a close watch on MHI's measures to prevent recurrence whether or not they have autonomously implemented checking of the improvement status of the procurement control by KEPCO.

3) The action status by the NSC

The NSC, immediately after the occurrence of the accident, the NSC held an extra conference, to investigate the outline of the accident, and, in order to help in the response to be taken, compiled a report, "Accident at Unit 3 of the Mihama Nuclear Power Station of KEPCO" In order to investigate and review the technical matters etc. for ensuring safety of the secondary-system piping of nuclear installations, the Subcommittee on the Secondary System Piping Rupture at Unit 3 of the Mihama Nuclear Power Station was established under the Special Committee on Analysis and Evaluation of Nuclear Accidents and Failures, and the final report was compiled on April 28, 2005. The Accident Investigation Subcommittee received the accident investigation progress reports from NISA, compiled the items to be followed, and the views of the NSC, and informed NISA of them.

In addition, the NSC reviewed the responses taken by the government agencies for the accident of Unit 3 of the Mihama Power Station, through the "Subsequent regulation reviews" of regulatory body's programs concerning the regulations of periodic licensee's inspections of commercial nuclear power generating facilities and the "Subsequent regulation reviews" of the regulatory body's programs concerning the quality assurance in safety preservation activities at the operating commercial nuclear power generating facilities and the NSC presented their own opinion in a written report issued in February 2005.

Also, the NSC conducted the "Subsequent regulation reviews" of the regulatory body's checking actions for recurrence prevention concerning the secondary piping rupture accident of Unit 3, Mihama Nuclear Power Station, The Kansai Electric Power Company, Inc. They checked the system of the government agencies to confirm implementation status of the measures to prevent recurrence of the KEPCO, and compiled the report in March, 2006.

4) Events Associated with the Accident and the Measures Taken

Through the investigation of the rupture accident due to wall thinning of the secondary-system piping of the Mihama Unit 3, the importance of the maintenance management of aged plants and the necessity of measures taken for degradation of the organizational climate are recognized, and the measures for those issues have been improved and enhanced (refer to Section 14.3). The steam, which discharged at the accident, spread into the central control room, which revealed that the air-tightness of the central control room was inadequate. Therefore, the habitability of the central control rooms is currently under study (refer to Section 18.7).

(2) Measures Taken for the Crack etc. of the Hafnium-Plate-Type Control Rods of BWR Nuclear Power Plants

In January 2006, cracks of the hafnium-plate-type control rods in-service were found at Unit 6 of the Fukushima Daiichi Nuclear Power Station of Tokyo Electric Power Co., Inc(hereinafter referred to as "TEPCO")., and the licensee reported the event to NISA. NISA informed the licensees who own BWRs (hereinafter referred to as the "BWR licensees") of the investigation and the report of the results on the usage of the same-type control rods, on the operability of the same-type control rods in operating reactors, and on the identity of the cracks etc. of the same-type control rods in reactors during shutdown.

In February 2006, NISA, as a result of the evaluations of the above-mentioned reports, informed the BWR licensees using the same-type control rods to use them in the fully-inserted positions to avoid any potential of not being able to insert them due to cracks, when they are to be used until their neutron irradiation reaches more than a certain level. (The subject units, as of the end of March 2006, are the following 23 nuclear reactors using the same-type control rod; Tsuruga Unit 1, Higashidori Unit 1, Onagawa Units 1, 2 and 3, Fukushima Daiichi Units 2, 3, 4, 5 and 6, Fukushima Daini Units 1, 2, 3 and 4, Kashiwazaki Kariwa Units 1, 2, 3 and 6, *Hamaoka Units 1, 2, 3, 4 and 5.)*

NISA received the reports from TEPCO and Chubu Electric Power Co., Inc(hereinafter referred to as "Chubu") concerning the causes analysis and countermeasures for hafnium-plate-type control rods on which the cracks were found in the in-service or spent control rods in May 2006. In response to this, NISA publicly released the NISA's investigation report, and directed to perform a visual inspection of all same-type control rods at the annual periodic licensee's check as short-term measures. In addition, NISA and licensees identified the issues to be studied for medium- to long-term. These measures were reported to the Working Group for Accident and Failure Countermeasures, Subcommittee for Nuclear Emergency Preparedness of the Nuclear and Industrial Safety Subcommittee and the NSC.

(3) Measures Taken for the Strainer Plugging Issue in BWR and the Containment Sump Clogging Issue in PWR

In June 2004, concerning the issue that the recirculation function of the emergency core cooling system could be affected by the plugging of the strainers or by containment sump screen clogging during a loss of reactor coolant accident of BWRs and PWRs, NISA, as a result of this study based on an overseas review status, directed licensees to make a report of investigation on as-build status of thermal insulation etc about BWRs and PWRs and evaluation results on the effectiveness of strainers or screen about BWRs.

In April 2005, the investigation results on the thermal insulation etc. of BWRs and PWRs and interim measures such as a revision of operation procedures, were reported by several licensees. NISA considered that the interim measures reported by the licensees were adequate as they were equivalent in content to the measures of which effectiveness was verified in the U.S.A. NISA considers that full implementation of the interim measures are essential in order not to cause an important-to-safety problem until the measures on the equipment, such as larger strainers or screens etc., will be completed, and NISA instructed the licensees to implement the interim measures and has been checking the implementation status by the Operational Safety Inspection etc. For the effectiveness evaluation of the strainers of BWRs, it was reported that the effectiveness was not confirmed at some nuclear installations as a result of a conservative evaluation in accordance with the U.S. regulatory guides. NISA studied the adequacy of those evaluations and measures at the Working Group for Safety Assessment, of the Nuclear Reactor Safety Subcommittee, under Nuclear and Industrial Safety Subcommittee. As NISA, receiving the results that the larger strainers and replacement of the thermal insulation are required, instructed in October 2005, to implement the measures for BWRs. All BWR licensees have been implementing the measures in accordance with the instruction. This status of implementation was also reported to the NSC.

In October 2005, NISA instructed PWR licensees to submit reports on the result of the effectiveness evaluation of the PWR screens by August 2006. In August 2006, proposed measures for the equipment of each plant and responses to the effectiveness evaluation methods were all submitted and reported to the Working Group, and in May 2007, it was concluded these measures were necessary to be implemented.

(4) Measures Taken for the Falsification of Full-Flow Test Data on Feed Water Flow Instrumentations

NISA, on receipt of an in-house whistle-blowing on November 15, 2005: "almost entire full-flow certification examination data on the reactor feedwater flow instrumentation of Toshiba Corporation were falsified", investigated ten nuclear reactors of TEPCO and Tohoku Electric Power Co., Inc., using the flow instrumentations made by the company. As a result, it was confirmed that the fraudulent correction had been made on the test data of feed water flow instrumentations supplied to three units of both licensees. But, NISA determined after its evaluation that there was no safety or legal issue involved due to the inaccuracy of the feed water flow instrumentations.

NISA ordered Toshiba Corporation to conduct a thorough root cause analysis and to report the recurrence-preventing measures established based on the analysis results in order to never make such a falsification again. NISA requested of TEPCO and Tohoku Electric Power Co., Inc., to establish measures for quality assurance, of procurement for prevention of recurrence and to then report them. NISA conducted the on-site inspection at Toshiba Corporation, in accordance with the Reactor Regulation Law in June, October and December 2006, in order to confirm the implementation status of the recurrence-preventing measures.

(5) Safety Review and Assessment of Power Plants Loading Mixed-Oxide Fuel Partially in the Core

On this matter, the Genkai Units 3 (Kyushu Electric Power Co., Inc.: PWR) was authorized on September 7, 2005, and the Ikata Unit 3 (Shikoku Electric Power Co., Inc.: PWR) on March 28, 2006, and the Hamaoka Unit 4 (Chubu Electric Power Co., Inc.: BWR) on July 4, 2007.

(6) Measures Taken for Faulty Design of the ABWR Turbine Blades

In June 2006, at Unit 5 of the Hamaoka Nuclear Power Station, Chubu Electric Power Co., Inc. (hereinafter referred to as "Hamaoka Unit 5"), the steam turbine and nuclear reactor were automatically shutdown due to excessive turbine-shaft vibration caused by the breaking off of a blade of the 12th stage of the low pressure turbine(B) NISA performed a visual inspection of the blades etc. removed from the shaft, which belonged to the same stage as the one of the low-pressure turbine (B) where the blades came off, confirmed breakages and cracks at the fork-shape joint section of the blades, and determined that the turbine did not conform to the Technical Requirements for Nuclear Power Generation Equipment. NISA, judging that it was necessary to confirm conformity of the steam turbine of Unit 2 of the Shika Nuclear Power Station, Hokuriku Electric Power Co.,(hereinafter referred to as the "Shika Unit 2", which is of the same design as the one at Hamaoka Unit 5, according to the same Technical Requirements, directed to have the blades checked. As a result, breakages and cracks were confirmed on the blades of the low-pressure turbine (B) also at the Shika Unit 2.

In October of the same year, NISA received a report concerning the causes and countermeasures from Chubu Electric Power Co., Inc. and Hokuriku Electric Power Co., Inc. The report describes that the short-term measures are to remove all of the 12th-stage blades of *the low pressure turbines and to apply pressure plates (distributors) to the stationary blades of the same stage prior to the restart, and the long-term measure is to design and manufacture new blades of the 12th stage, taking into account the fluid-induced vibration force due to random vibration and flashback. In November of the same year, NISA released an investigation report showing that the licensees' measures for prevention of recurrence are adequate.*

NISA reported these measures to the Working Group for Accident and Failure Countermeasures of the Subcommittee for Nuclear Emergency Preparedness, under Nuclear and Industrial Safety Subcommittee and the NSC.

Chubu Electric Power Co., Inc. submitted the construction plan of installation of pressure plates to the Hamaoka Unit 5 turbine on November 8, 2006. NISA reviewed the plan. Chubu Electric Power Co., Inc. passed its pre-service inspection and resumed the commercial operation of the Hamaoka Unit 5 on March 13.

NISA, upon receiving the construction plan of installation of distributors to the Shika Unit 2 turbine on November 13, 2006 from Hokuriku Electric Power Co., examined the plan.

(7) Check on the Status of Implementing the Measures against Sodium Leakage of the Monju Power Station

At the fast breeder prototype-reactor "Monju", a sodium leak accident of the secondary cooling system occurred in December 1995, and the reactor has been kept in a low-temperature shutdown state since that time.

The licensee, Japan Nuclear Cycle Development Institute (in October 2005, the Institute was integrated with the Japan Atomic Energy Research Institute, and has become an incorporated administrative agency, Japan Atomic Energy Agency), in July 2007, completed construction work for the measures to cope with sodium leakage that started in September 2005. Monju is now conducting verification tests of the completion of the work. NISA, as the prerequisite to the restart of the Monju, is conducting the following:

1) Integrity check on equipment, system and fuel that have not been used for a long period of time, with pre-service inspection and on-site inspection

2) Check on the quality assurance relating to technical capabilities, the time required to extract the secondary-system sodium, the plant maintenance plan, with pre-service inspection and on-site inspection.

3) Safety examination for the application of change in fuel assemblies for the first core loading.

Concerning the safety examination for the application of change in fuel assemblies for the first core loading, in October 2006, Japan Atomic Energy Agency applied the usage of fuel assemblies prepared for reload of the core as the first core loading to be used at the restart of

Monju. NISA asked for consultation to the NSC in July 2007.

Various activities for ensuring the safety of Monju have been confirmed openly in the "Study Group for Confirmation of the Monju Safety", which was established under the Nuclear and Industrial Safety Subcommittee in November 2005. As of June 2007, the Study Group was held 9 times.

The NSC conducted the audit of NISA's activities based on the reports about 1) The Results of Confirmation about the Monju Integrated Safety Check and 2) The Results of Deliberation about the Important Items after the Approval of Amendment of Monju Establishing License, and the NSC appraised, in its decision in June 2007, the NISA's activities concerning confirmation of Quality Management System as appropriate.

(8) Check on Seismic Safety of the Onagawa Nuclear Power Station

When the earthquake occurred in Miyagiken-oki in August 2005, the Onagawa Nuclear Power Station of Tohoku Electric Power Co., Inc. experienced earthquake ground motion exceeding the design basis earthquake ground motion. NISA directed Tohoku Electric Power Co., Inc. to analyze the factors that caused earthquake ground motion which exceeded the design basis earthquake ground motion, to check the seismic safety in reference to the important-to-safety equipments of the power station. In accordance with the direction, Tohoku Electric Power Co., Inc. first reported the results of the check on seismic safety for Units 2 and 3 of the power station to NISA. NISA, based on the evaluation of the report, notified the licensee that the evaluation methods of seismic safety and study results of the seismic safety of the licensee are adequate. Since Unit 1 of the Onagawa Nuclear Power Station has been in operation for 22 years after commissioning, NISA directed the licensee to assess also the effect of aging on the seismic safety. In September 2006, NISA notified the licensee that the evaluation methods and results of the check on the seismic safety submitted by the licensee are adequate. For additional information, refer to the report of Article 14 (Section 14.5).

Units 1, 2 and 3 of the Onagawa Nuclear Power Station were automatically shutdown when the earthquake occurred and had undergone the Periodic Inspection after that, they resumed their operation after receiving notifications of confirmation of the equipment seismic safety and completing the Periodic Inspections one by one.

(9) Measures Taken for the Partial Omission of Periodic Inspections Conducted by the Incorporated Administrative Agency, Japan Nuclear Energy Safety Organization (JNES)

On February 22, 2007, NISA received a report from JNES, which indicated that an incomplete inspection (failure to perform a part of functional test) was discovered at the 21st Periodic Inspection of the Tokai No.2 Power Station, the Japan Atomic Power Company, conducted by JNES in 2005. On February 23, 2007, NISA ordered to report the result of the check if there are any other incomplete inspections to JNES in accordance with the Electricity Utilities Industry Law. On March 9, 2007, JNES reported to NISA that three defects in the record check were discovered at the Periodic Inspections of other power plants in addition to the above-mentioned one. The report describes that the root cause of the defects in record check *(failure to check records, failure to prepare inspection records, etc.) is an inadequate mechanism to not file or check records due to human error, and the measures to prevent recurrence are process management by making an appropriate management table and improvements in the mechanism to check by the administration department. The Director General of NISA expressed that "it is regrettable that defects were found in the Periodic Inspections conducted by JNES", and he issued a severe warning in writing and directed the President of JNES to take measures of thorough recurrence-prevention.*

(10) Comprehensive Check of Electric Power Facilities

1) Background and Circumstances of the Comprehensive Check of Electric Power Facilities

On November 21, 2006, NISA received a report from the Chugoku Electric Power Co., Inc., which describes that the company falsified data regarding a dam for a hydroelectric power plant in the past. Receiving information that other licensees have also conducted construction work on hydraulic power production plants without obtaining authorization in accordance with the River Law, NISA directed all licensees to conduct investigations into hydraulic power production plants. At the same time, at nuclear power plants, cases such as inappropriate corrections for the measured temperatures of seawater for cooling were also revealed.

Receiving a series of such reports, NISA, under direction of the Minister of METI, directed licensees on November 30, 2006 to check whether there exists any similar data falsification (hereinafter referred to as the "Comprehensive Check") concerning the facilities for hydraulic-power, thermal-power and nuclear-power generation and to make a report on the results of the check by March 31, 2007.

On March 30, 2007, the licensees reported 316 cases in total to NISA as the results of the Comprehensive Check regarding data falsification etc., and on April 6 of the same year, measures to prevent recurrence were reported.

On April 20, 2007, the Minister of METI issued a document reconfirming the purposes and objectives of the Comprehensive Check to the presidents of all electric power companies. The document explains that the main purport of the Comprehensive Check is "to disclose facts without hiding them", specifically, (1) to cut off the vicious circle of continuing to falsify records on the premise of past falsifications, (2) to establish a mechanism not to allow a falsification, (3) to share information on accidents and troubles, and utilize them for prevention of recurrence, and (4) to improve the culture and climate of electric power companies by steadily promoting such activities.

The 316 cases were classified into the following four groups according to the contents;

Group 1: Cases that conflict with legislations and regulations, and have effect on the safety

Group 2: Cases that are confirmed not to compromise safety, but conflict with *legislations and regulations and are issues in terms of compliance*

Group 3: Cases that are not necessarily related to safety requirements, but conflict with legislations and regulations and are issues in terms of compliance

Group 4: Errors in writing etc.

The Minister of METI decided to make administrative dispositions according to the contents and group of the reported cases, and in accordance with the provisions of the Administrative Procedure Law, issued a document listing notices of the dispositions, saying that an appeal against the notice of the dispositions is acceptable to the licensees. Also, the Minister issued a document instructing to submit a specific program and schedule for measures to prevent recurrence. In addition, the Minister issued a document to the manufacturers concerned, requesting them to make an action plan to improve a level of nuclear safety, including a mechanism of sharing information and to promote sharing information concerning safety technologies when responding to the requirement from electric power companies for maintenance management and procurement.

The above-mentioned administrative dispositions were implemented, since there was no overture of an appeal from the licensees. METI, based on the Comprehensive Check results on the power generation facilities, also established an action plan that specifies responses to be taken by METI itself.

2) Instructions to Improve Safety and Safe Operation of Nuclear Power Generation Facilities

The Comprehensive Check results of nuclear power generation facilities showed that there had been no falsifications since October 2003 when the inspection system was revised. Although they were the cases within the period before the revision of the inspection system, there were a total of 98 cases, including 11 cases conflicting with the Reactor Regulation Law and the Electricity Utilities Industry Law and impairing the safety that the laws intend to ensure (Group 1) (refer to Table 6-2). Major actions taken by METI after receiving the results include; (1) issuing an order to amend the Operational Safety Program of the nuclear power stations of TEPCO, Hokuriku Electric Power Co., the Chugoku Electric Power Co., Inc., and the Japan Atomic Power Co., (2) earlier-than-scheduled implementation of, or extension of period for the Periodic Inspections of these electric power companies' nuclear power stations, and implementation of special inspections of the Operational Safety Inspections and Periodic Inspections including the appointment of NISA Special Nuclear installation Management Supervisors, etc., (3) amendment of a Ministerial Order for making it mandatory to report the accidental withdrawal of control rods, etc., and the promotion of international sharing of information concerning accidents and failures.

On May 21 ,2007, all licensees concerned submitted the action plans for prevention of recurrence to METI. The major specific contents include; involvement of executives to the program, thorough implementation of education on operational safety, retention of records by alarm typewriters, supporting free access of NISA Nuclear Safety Inspectors to nuclear installations, independency of chief reactor engineer, promotion of information sharing with

NUCIA etc., report of a deviation from the limiting conditions for operation, interdepartmental information sharing, information sharing among electric power companies. NISA judged that these action plans are generally adequate as they are based on the instructions issued by METI and the time schedule and methods to implement these actions are concretely provided, and decided to check the implementation status through the Operational Safety Inspection etc. NISA also decided to make necessary amendments etc. of legislations and regulations corresponding to implementation of the action plan. These responses were reported to the NSC.

Receiving the report from NISA about the falsification and concealment concerning nuclear facilities, the NSC has investigated and has tried to understand the whole story. Responding to the actions taken by METI, the NSC determined "Actions to take concerning the malicious conducts of unreported alteration and concealment at nuclear facilities" on April 23, 2007. As a part of these actions, the NSC ; on May 15, 2007, invited the Chief Reactor Engineers from each nuclear power plants, interviewed and exchanged the opinions, ; on May 17 and 31, 2007, heard the opinions of Executive of Hokuriku Electric Power Co. Ltd, and ;on May 28 and June 4. 2007, heard from Japan BWR Owners Group and Japan PWR Owners Group respectively, about their activities for information sharing and for analyzing accident and failure.

6.3 Evaluation and Verification of Safety, and Position as to Continued Operation

NISA had implemented the necessary safety assessment and verification for existing nuclear installations at planning, licensing, construction and operation stages. They are explained in the reports of Article 7 to Article 19.

Through those assessments and verification, principles of this convention have been applied to ensure the safety of existing nuclear installations for every stage from licensing to operation.

As shown in Section 6.2, for events that occurred during the period after the previous reporting, NISA had judged appropriately whether any issue on safety existed or not, instructed licensees to take measures for ensuring safety, as necessary, and confirmed that the measures were appropriately taken. Therefore, it is appropriate to continue operation of operating nuclear installations.

For the Monju, which is a nuclear installation under construction that has attained criticality, implementation of safety measures against sodium leakage and the passing the pre-operational inspection are requirements for its operation.

Table 6-1 Accidents and Failures at Nuclear Power Stations Reported by Licensees during the Reporting Period

Name of power plant	Title	Event date	INES level
Takahama, Unit 4	Significant indication by the eddy current examination of steam generator tubes	September 6, 2004	0−
Sendai, Unit 1	Significant indication by the eddy current examination of steam generator tubes	September 10, 2004	0−
Tomari, Unit 1	Significant indication by the eddy current examination of steam generator tubes	September 21, 2004	0−
Fukushima Daiichi, Unit 2	Automatic trip of one reactor-coolant recirculation pump	September 29, 2004	0−
Mihama, Unit 1	Insufficient wall thickness of the turbine-driven auxiliary feedwater piping	October 25, 2004	0−
Sendai, Unit 2	Significant indication by the eddy current examination of steam generator tubes	December 15, 2004	0−
Hamaoka, Units 1 & 2	Cracks of Units 1 and 2 common stack duct joint	December 21, 2004	0−
Ikata, Unit 1	Cracks of ventilation stack of the reactor auxiliary building	December 23, 2004	0−
Tsuruga, Unit 2	Significant indication by the eddy current examination of steam generator tubes	January 18, 2005	0−
Kashiwazaki Kariwa, Unit 1	Steam leak from the small-bore drain pipe in the turbine building	February 4, 2005	0−
Onagawa, Unit 1	Nitrogen leak from the reactor containment	February 25, 2005	0−
Mihama, Unit 1	Failure of B make-up-pump manifold cover bolts	March 19, 2005	0−
Mihama, Unit 1	Cracks of the lower part of auxiliary building ventilation stack, and the faulty connection of a drain pipe	April 28, 2005	Out of scale event
Ikata, Unit 3	Failure of the chiller for the central control room air conditioner	May 12, 2005	0−
Kashiwazaki Kariwa, Unit 5	Automatic reactor shutdown following the turbine trip due to low condenser vacuum	July 3, 2005	0+
Shimane, Unit 1	Indication trouble for closed position of a drywell vacuum breaker valve	July 8, 2005	0−
Tokai, No.2	Valve stem failure of motor-driven reactor feedwater pump discharge valve	August 10, 2005	0−
Fukushima Daiichi, Unit 5	Valve stem failure of the test bypass valve of the core spray system	August 22, 2005	0−
Mihama, Unit 1	Make-up water leakage from the seal of primary coolant pump No. 3	September 29, 2005	0−
Fukushima Daiichi, Unit 2	Automatic trip of one reactor-coolant recirculation pump	October 9, 2005	0−
Fukushima Daini, Unit 2	Damage of a seawater strainer of the cooling system for residual heat removal equipment	November 2, 2005	0−
Tomari, Unit 1	Cracks near the weld of stiffener for emergency ventilation stack	January 6, 2006	0−
Sendai, Unit 1	Significant indication by the eddy current examination of steam generator tubes	January 13, 2006	0−

Name of power plant	Title	Event date	INES level
Shika, Unit 2	Manual shutdown of the nuclear reactor due to malfunction of a steam supply isolation valve of the reactor core isolation cooling system	January 27, 2006	0−
Fukushima Daiichi, Unit 6	Cracks etc. of hafnium-plate-type control rods	February 1, 2006	1
Fukushima Daiichi, Unit 3	Cracks etc. of hafnium-plate-type control rods	March 3, 2006	1
Fukushima Daiichi, Unit 2	Automatic trip of one reactor-coolant recirculation pump	March 14, 2006	0
Ikata, Unit 1	Cracks of the weld of steam distributor in the moisture separator and reheater	June 5, 2006	Out of scale event
Fukushima Daini, Unit 1	Cracks of a valve stem of a residual heat removal system flow control valve	June 7, 2006	0
Hamaoka, Unit 5	Automatic shutdown of the nuclear reactor following steam turbine trip	June 15, 2006	0+
Onagawa, Unit 2	Observation of water puddle containing radioactive materials in the reactor-building torus room	August 3, 2006	0−
Hamaoka, Unit 3	Cracks etc. of hafnium-plate-type control rods	August 7, 2006	1
Fukushima Daiichi	Tritium release to the outside of control zone	August 11, 2006	0−
Takahama, Unit 3	Automatic shutdown of the nuclear reactor due to low steam-generator level during power down	August 18, 2006	0+
Tsuruga, Unit 2	Cooling water leak from heat transfer tube of the reactor building closed cooling water system	October 4, 2006	0−
Shimane, Unit 1	Corrosion of condensate storage tank	October 13, 2006	0−
Shimane, Unit 1	Wall thinning of piping at condensate filter outlet header	November 9, 2006	0−
Fukushima Daiichi, Unit 2	Manual reactor shutdown following earth fault of circuit for the automatic depressurization system	January 17, 2007	0−
Genkai, Unit 2	Cracks of extraction piping of the surplus extraction water system	January 24, 2007	0−
Fukushima Daiichi, Unit 4	Reactor power change due to a misoperation	February 11, 2007	0+ (tentative)
Fukushima Daini, Unit 4	Automatic shutdown of the nuclear reactor due to an alarm of "main steam pipe radioactivity high-high trip"	February 18, 2007	0− (tentative)
Fukushima Daiichi, Unit 5	Manual reactor shut-down due to malfunction of a valve of the core spray system	February 20, 2007	0+ (tentative)
Sendai, Unit 1	Significant indication by the eddy current examination of steam generator tubes	May 10, 2007	0− (tentative)

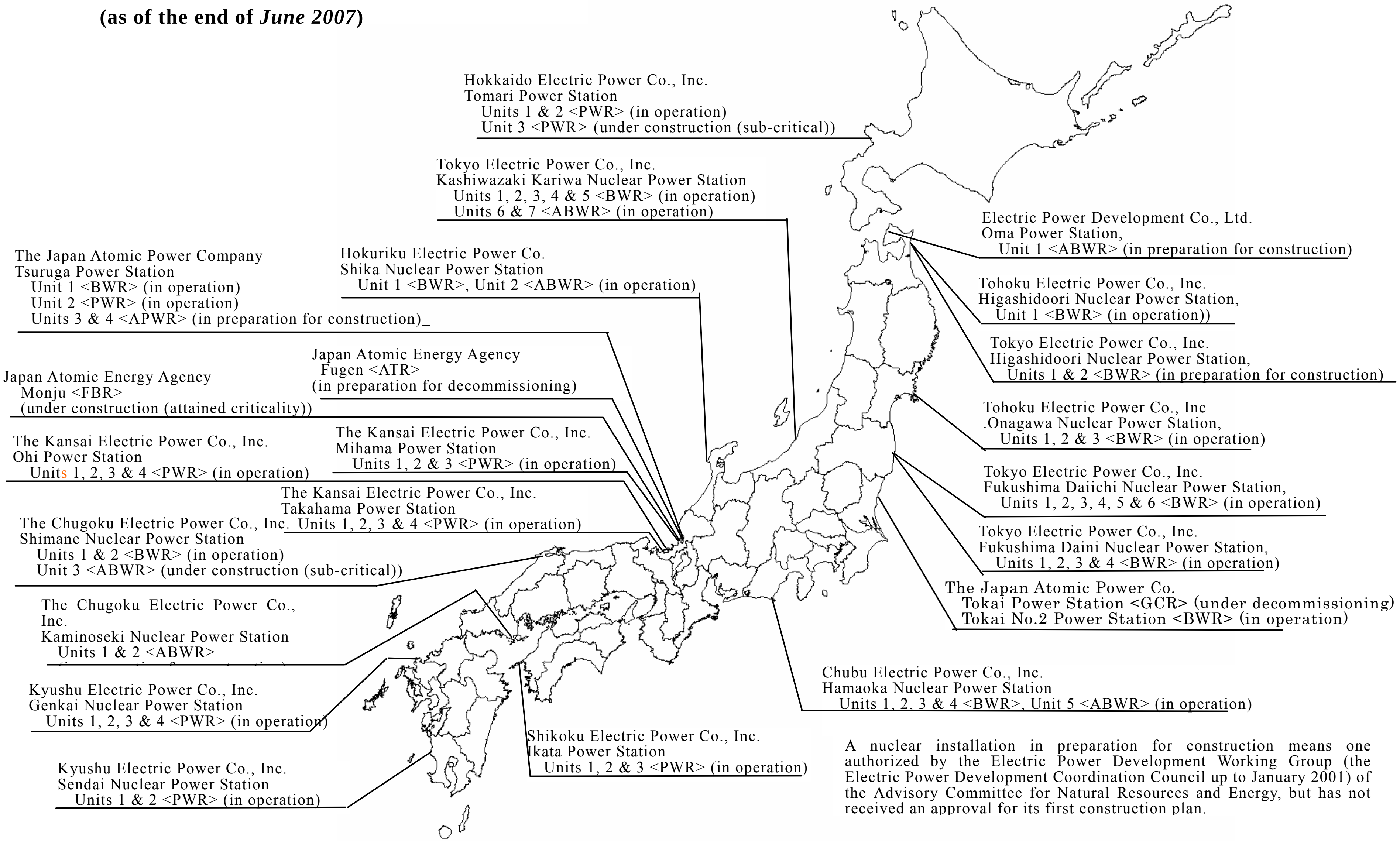
Table 6-2 List of 11 Cases Classified as Group 1 in Nuclear Power

Electric power company	Plant name	Time period	Summary
Hokuriku Electric Power Co.	Unit 1 of the Shika Nuclear Power Station	June 1999	• **Criticality accident occurred during reactor shutdown (during Periodic Inspection)** During the Periodic Inspection, a criticality accident occurred due to the withdrawal of three control rods caused by inadvertent isolation work of hydraulic control units. The logbook etc. was falsified in this case and also a report to the national government required by legislations and regulations was not made. Furthermore, no investigation into the cause was conducted and no measures to prevent recurrence taken.
Tokyo Electric Power Co., Inc.	Unit 3 of the Fukushima Daiichi Nuclear Power Station	November 1978	• **Criticality of nuclear reactor due to control-rods withdrawal and falsification of logbook, etc.** During Periodic Inspection, five control rods were withdrawn due to inadvertent isolation works of hydraulic control units, which resulted in the criticality of the reactor. Since the operation shift at that time did not recognize the occurrence of criticality and did not take any special measures, the criticality lasted over 7 and a half hours. In addition, the logbook was falsified to hide the fact.
	Unit 4 of the Fukushima Daini Nuclear Power Station	From October 1988 to January 1990	• **Unlawful actions for construction plan and pre-service inspection of a control rod drive mechanism,** At scram tests of the control rod drive mechanisms (CRD), trouble occurred at one CRD. Replacement of the CRD concerned was performed without submitting the construction plan for it. After that, unlawful action such as undergoing the pre-service inspection of the CRD while using a fake unit was also committed.
	Unit 1 of the Kashiwazaki Kariwa Nuclear Power Station	May 1992	• **Falsified indication for operation of a residual-heat-removal cooling intermediate pump (A)** Although the motor of the residual-heat-removal cooling intermediate (RHIW) pump (A) was out of order, emergency diesel generators underwent inspection with the indication lamp tampered in the central control room so as to make it look as if the pump started. Then, the nuclear reactor was started without checking the integrity of other systems, which is required by the Operational Safety Program.
The Chugoku Electric Power Co., Inc.	Unit 2 of the Shimane Nuclear Power Station	May 1998	• **Negligence of checking the operability of other trains in repairing diesel engine cooling water leakage** Although one train of the emergency diesel generators was inoperable with the reactor at rated power, the record of having conducted tests for other trains, which are required by the Operational Safety Program to continue operation, could not be confirmed.

Electric power company	Plant name	Time period	Summary
	Unit 1 of the Shimane Nuclear Power Station	June 2001	• **Negligence of checking the operability of other trains in repairing high-pressure- core-injection-system main stop valve (HPCI MSV) as it failed to open** Since the main stop valve failed to open with the reactor at rated power, the repair was performed, but the record of having checked the operability of the alternative emergency core cooling systems, which is required by the Operational Safety Program to continue operation, could not be confirmed.
The Japan Atomic Power Company	Unit 2 of the Tsuruga Power Station	January 1994	• **Negligence of checking the operability of other systems in replacing a part (gasket) for leak-tightness of emergency diesel generator** Although a water leakage occurred at the cooling water system of one emergency diesel generator, reactor operation continued without checking the integrity of other systems.
	Unit 1 of the Tsuruga Power Station	From September 1995 to March 2000	• **Hiding the corrosion event on the outside surface of condensate storage tank** Although the plate thickness of the tank lower section became less than the required minimum thickness specified in the application of construction permit due to corrosion, the tank was used with its water level lowered without checking the required strength.
	Unit 2 of the Tsuruga Power Station	From April to December 1996	• **Hiding the occurrence of a very small leak of the reactor coolant** When a leak was found in the piping in the containment, the operation should have been discontinued for repair, but the fact was hidden and the operation was continued without repair for about 8 months.
	Unit 2 of the Tsuruga Power Station	July 1997	• **Deceptive action to an equalizing valve during an inspection of containment leakage rate** An official inspection of containment leakage rate was conducted with a block plate installed at the outlet of an inboard equalizing valve with a leak identified, of the regular airlock, without following the appropriate in-house procedures. Then, the nuclear reactor was started up with the equalizing valve replaced, but the local leak rate test was not carried out prior to the start-up.
	Tokai No.2 Power Station	2001 or before	• **Falsification of flow-rate data during a functional test of the reactor building gas processing system** Since the airflow did not satisfy a specified flow rate during a functional test of the standby gas treatment system, the data was falsified with instrument adjustments so as to satisfy the specified flow rate.

Fig. 6-1 Locations of Nuclear Installations

(as of the end of *June 2007*)



A nuclear installation in preparation for construction means one authorized by the Electric Power Development Working Group (the Electric Power Development Coordination Council up to January 2001) of the Advisory Committee for Natural Resources and Energy, but has not received an approval for its first construction plan.

Attachment

The NSC 2007-D8 (Decided on 23 April 2007)

Actions to take, concerning the malicious conducts of unreported alteration and concealment at nuclear facilities (Summary)

The NSC received on 20 April 2007 a report from NISA on the "Assessment and follow-up actions concerning comprehensive checks of power generation facilities (Comprehensive Inspection Report)" and another report on the "Investigation of the criticality accident in 1999 Unit 1, Shika Nuclear Power Station, Hokuriku Electric Power Company, Inc. and other unforeseen cases of control rod dislodgement during reactor outages (Criticality Accident Report)."

Repeated malicious conducts of unreported alteration and concealment jeopardize the base of ensuring nuclear safety. It is really shameful that some serious cases were among them.

Currently nuclear licensees have been in the process of implementing recurrence prevention measures under the regulatory frames strengthened in 2002 and 2003. The actions were being taken based upon the lessons uncovered of similar data alterations in the past. Most of malicious conducts recently uncovered were before these recurrence prevention measures came into effect. Nevertheless, the recently uncovered cases again hampered the public trust in nuclear energy.

For recovering public trust, the NSC considers it really important to put into effect soonest the follow-up actions proposed by NISA. Following are the immediate actions by the NSC.

1. Criticality accidents and control rod dislodgement

The criticality accident on Unit 1, Shika Nuclear Power Station, Hokuriku Electric Power Company, Inc. is really a matter of serious concern in the context that an unforeseen criticality, which is an emergency (at a nuclear power plant), is secretly covered. The NSC shall examine closely the unforeseen cases of control rod dislodgement, in view of their risks to fundamental nuclear safety.

It is to the NSC awareness, meanwhile that, among the cases of control rod dislodgement, which came into light starting with the criticality accident of Hokuriku Electric Power Company, there are some cases irrelevant to data alterations.

(1) The ensuring of safety during reactor outages

The Criticality Accident Report (NISA) states, "Important is to examine how to ensure reactor safety, as part of operation management during outages, including hardware modifications if necessary, because safety functions of the facility could be temporarily lifted during periodic inspection outages for the purpose of inspections or tests."

It is the NSC position that the principle of defense-in-depth should be maintained for safety measures and their certain implementation even during the outages.

Therefore, the NSC will conduct necessary examinations, being informed from NISA of its investigation and examination results of safety measures during outages such as operation management including hardware modifications as needed, based on the lessons from the criticality accident at Shika or from abroad.

(2) Criticality accident on Unit 1, Shika Nuclear Power Station, Hokuriku Electric Power Company, Inc.

The Criticality Accident Report (NISA) also states that “the analysis (by the Hokuriku Electric Power Company) meets with margin the safety criteria of the abnormal transients assumed in the (earlier) safety examination process, and, hence, the fuel integrity was apparently not jeopardized.”

Upon NISA instructions, JNES is to perform an “independent (cross) check” of this analysis. The NSC requests NISA to report to the NSC on its evaluation of the case based on the JNES analyses.

Besides the regulatory requirements, the NSC plans to interview, at the earliest convenience, the top management of the operator concerning its determined measures to let such critical occurrence never concealed again.

2. Analyses of, and feedback from, the information on accidents and troubles

(1) Global sharing of the information on accidents and troubles, and its utilization

The Criticality Accident Report (NISA) also clarifies that: Six unforeseen criticalities have been reported at commercial BWR plants, three out of which occurred during the outages; all these three cases occurred when control rods were maneuvered for insertion or withdrawal, the situation being different from the recent case of Hokuriku Electric Power Company; and, for this reason, no significant information could have been withdrawn from these cases for preventing the recent case of Hokuriku Electric Power Company.

Besides the cases mentioned in the Criticality Accident Report (NISA), the criticality accident in 1987 during the outage at the Oskarshamn Nuclear Power Station, Sweden, prompted NRC to draw attention of the US BWR owners.

The NSC must admit that Japan is still insufficient to share the global information on nuclear accidents and troubles, considering not only the criticality cases but also other cases, such as the secondary circuit pipe rupture at Unit 2, Surry Nuclear Power Station, U.S. (1986), or the strainer blockage at Unit 2, Barseback Nuclear Power Station, Sweden (1992).

Needless to say, it is quite important to extract lessons for enhancing nuclear safety of our national nuclear facilities from the information on nuclear accidents and troubles in other countries. It is equally important to carefully examine our cases, in disseminating our national experience to other countries, in terms of their impacts to our national nuclear facilities as well as their user-friendliness to international recipients.

Therefore, NISA should strengthen its activities on the collection/dissemination and the analysis/utilization of international information relevant to nuclear accidents and troubles, by the use of multi-national (IAEA, for instance) or bilateral cooperation frameworks.

The NSC also examines the measures to make use of internationally shared information, such as the IAEA databases, for preventing accident recurrences in our country.

(2) Information sharing among utilities and reactor manufacturers, and its utilization

Important in the utilization of the information relevant to accidents/troubles is for electric utilities and reactor manufacturers to analyze the causes closely and share the recurrence prevention measures based thereupon. As pointed out in the Criticality Accident Report (NISA)[1], the electric utilities and reactor manufacturers should, through the association such as the Japan BWR Owners Group and the Japan PWR Owners Group, acknowledge the importance of sharing the technological information and its use for accident recurrence prevention.

With the intention of promoting such activities, the NSC plans to interview, at the appropriate timing, the Japan BWR Owners Group and the Japan PWR Owners Group concerning their activities for information sharing.

(3) Knowledge base building of reactor operating information by nuclear operators

The information on operation management should be appropriately recorded and accumulated for analysis and use for enhancing nuclear safety. However, the Comprehensive Inspections Report (NISA) has revealed a number of cases with defects in this regard.

The NSC takes a position that a station-level knowledge base on the operation management information should be the generic base condition for recurrence prevention. To this end, an electronic system at a station-level is an effective means for sharing the operation management information and for its automatic recording and preservation. The NSC requests nuclear operators to proactively tackle this issue, although it is not a regulatory requirement. The system would also contribute to the prevention of malicious conducts such as data alteration.

3. Advancement of revising the legally bound inspection regime

[1] The Japan BWR Owners Group is tackling the issue of information sharing, with participation of the electric utilities and the reactor manufacturers. Such activities should be vigorously and effectively exercised.

NISA has been examining, at its “Committee on the Inspection System,” a more scientific and rationalized inspection regime with due consideration to plant specific maintenance programs.

The NSC takes a position that the lessons from the Comprehensive Inspections Report (NISA) should be aggressively reflected in this revision process and, therefore, the Committee’s examination be advanced so that the operators’ safety measures be further promoted. The NSC expects to be informed, at the appropriate timing, of the concrete approach of NISA for advancing the revision process.

4. The NSC plans for motivating field-oriented approaches by means of site interviews of Reactor Chief Engineers

The reactor chief engineers are, as prescribed in the Reactors Regulation Law, responsible for supervising the plant safety measures. They are due to play crucial roles in ensuring reactor safety, with full use of their professionally technological knowledge and experience.

The NSC plans to promote placing higher importance to the plant site, or the very front line of ensuring safety. To this end, the NSC plans to share the problem awareness with reactor chief engineers through direct interviews at the earliest convenience and also encourage mutual information exchange between them.

5. NISA Plan of reviewing the revision of operator’s safety rules and of special safety inspections

NISA plans to execute special programs for nine units at seven nuclear power stations, which have been ranked as “I” in its Comprehensive Inspections Report. The NISA special programs: require the revision of safety rules to prescribe the reporting system, upon major accidents, to the top management for recurrence prevention; plan special inspections at the earliest periodic inspection; execute special supervision by special nuclear facility superintendents; and implement special safety inspections.

The NSC is to receive reports from NISA as appropriate and watch the development.

6. Subsequent Regulation Reviews of NISA’s follow-up actions

The NSC plans Subsequent Regulation Reviews for the regulation-related items from among NISA’s follow-up actions, in order to enhance nuclear safety.

B. Legislation and Regulation

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Article 7 Legislative and Regulatory Framework

1. Each Contracting Party shall establish and maintain a legislative and regulatory framework to govern the safety of nuclear installations.
2. The legislative and regulatory framework shall provide for:
- **(i) the establishment of applicable national safety requirements and regulations;**
- **(ii) a system of licensing with regard to nuclear installations and the prohibition of the operation of a nuclear installation without a licence;**
- **(iii) a system of regulatory inspection and assessment of nuclear installations to ascertain compliance with applicable regulations and the terms of licenses;**
- **(iv) the enforcement of applicable regulations and of the terms of licences, including suspension, modification or revocation.**

The Atomic Energy Basic Law has been established as the basic law governing the utilization of nuclear energy, and the Law on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (hereinafter referred to as the "Reactor Regulation Law"), the Electricity Utilities Industry Law *etc.* have been established as the laws to govern the safety of nuclear installations, *which are subjects to the convention.*

As a progress after the previous report, the technical standard in accordance with the Electricity Utilities Industry Law has been revised to define performance requirements, and the detailed specifications conforming to the performance requirements were defined by standards of the private sectors. And the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities etc. (hereinafter referred to as the "Regulatory Guide for Reviewing Seismic Design") has been revised, incorporating the latest knowledge into the technical requirements. The issues which are now under review and will be reflected to future revision or establishment of legislations and regulations are the new inspection system currently being reviewed by "The Taskforce on Inspection System" (see Article 19) and the use of risk information in legislations (see Article 14).

7.1 System of Legislations Governing the Nuclear installations

(1) Legislations concerning Utilization of Nuclear Energy in General

T*he overall system of legislations and regulations for utilization of nuclear energy is based on the Atomic Energy Basic Law.* The objectives of the Atomic Energy Basic Law are quoted as "to secure future energy resources, achieve progress in science and technology, and promote industry, by encouraging research, development, and utilization of nuclear energy, and thereby contribute to improvement of the welfare of human society and the people's living standard." The basic policy is prescribed as follows: "the research, development and utilization of nuclear energy shall be limited to peaceful purposes, on the basis of the highest priority of ensuring safety, and performed on a self-controlled basis under democratic administration, and the results obtained shall be made public and actively contribute to international cooperation."

In order to achieve these objectives and the basic policy, the law provides establishment of a set of laws to govern following areas:

- Establishment of the Atomic Energy Commission and the Nuclear Safety Commission, and their duties, organization, administration, and authorities.
- Regulations governing nuclear fuel materials *and nuclear source materials.*
- Regulations for construction, etc. of nuclear reactors.
- Prevention of radiation hazards.
- *Compensation for a nuclear damage*

The law also provides that those who will utilize nuclear energy shall manage their facilities with the first priority on safety under the supervision of the regulatory body in accordance with these laws.

The *basis to establish organizations related to regulation and the missions of the organizations are provided in the laws, such as the "Law for Establishment of the Atomic Energy Commission and the Nuclear Safety Commission", "Law for Establishment of the Ministry of Economy, Trade and Industry", "Law for the Japan Nuclear Energy Safety Organization, Incorporated Administrative Agency."*

(2) System of Legislations and Regulations Governing the Safety of Nuclear Installations, Subjects of the Convention on Nuclear Safety

Major legislations and regulations for safety regulation of nuclear installations are shown in Figure 7-1. As shown in the figure, *1) the Law on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors, 2) the Electricity Utilities Industry Law, 3) the Basic Law for Emergency Preparedness and the Special Law of Emergency Preparedness for Nuclear Disaster, 4) The Laws for Radiation Protection., 5) the Environment Impact Assessment Law, and 6) the Law on Compensation for Nuclear Damages have been enacted as major legislations and regulations.*

These laws provide requirements for approval, inspection and notification of licensing, construction, operation and decommissioning of establishment of nuclear installations. They are enacted after deliberations at the Congress, which means the revision also requires the resolution at the Congress.

*The nuclear installations in the scope of this Convention are any land-based civil nuclear power plants. They are using nuclear fuels and are regulated by both laws of safety regulations for reactors using nuclear fuel (Reactor Regulation Law) and safety regulations for commercial electric power generation (Electricity Utilities Industry Law). However, the regulations in accordance with the both laws are ruled not to overlap for the same matte*r.

The authorization is entrusted to regulatory body by in a law ,and the ordinances, in the *hierarchy under the law, describing the procedures for approval, inspection and notification, are enacted or revise, by the competent authorities, after obtaining the decision by the Cabinet. In the hierarchy under the ordinance, there are the rules which the competent authorities can establish, in accordance with the authorization by the law and the ordinance, the to define the details of application, the basis for approval, the technical requirements, the control items for radiation protection, the licensee's measures for safe operation, etc for approval, inspection and notification of various matters. And the competent authorities of the legislations and regulations can establish the technical standards (notice) and the guidelines of detailed technical requirements in accordance with these legislations and regulations.*

(3) Outlines of Major Laws and Regulations related to the Safety Regulation for Nuclear Installations

Outlines of each major law and regulation are described as follows:

1) Law on the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors (Reactor Regulation Law)

The Reactor Regulation Law, in order "to ensure that the utilization of nuclear source material, nuclear fuel material, and reactors are limited to peaceful purposes, and carried out in a planned manner, and to ensure safety of the public by preventing the hazards due to these utilization and providing physical protection of nuclear fuel material, in accordance with the spirit of the Atomic Energy Basic Law", provides prescriptions on establishment and operation etc. of reactors.

And the following matters are established for reactor facilities

- Regulations on the basic design and policies of reactor facilities for establishment of reactor facilities,
- Regulations on the detailed design for construction of reactor facilities (approval of design and construction methods),
- Inspections at the time of construction of reactor facilities (approval of the welding method, welding inspection and pre-service inspection),
- Regulations for operation of reactor facilities (approval of the Operational Safety Program and the Operational Safety Inspection) *(inspection of the observance status of the Operational Safety Program), notification of operation plan),*
- Inspection for operation of reactor facilities (Periodic Inspections of reactor facilities),
- Safe operation of reactor facilities and measures to be taken for protection of specific nuclear fuel materials,
- Regulations on transfer, succession or merger of reactor facilities, and

・ *Dismantling of reactor facilities.*

The allegation system was established so that personnel of licensees can allege violation of Reactor Regulation Law to the Minister of Economy, Trade and Industry *or* the Nuclear Safety Commission (the NSC), without unfavorable treatment. The system provides the rule to handle the allegation, such as for protection of personal data of allegers and for *appropriate implementation of the procedure for investigation and disclosure of an alleged case. By the end of March, 2007, Ministry of Economy, Trade and Industry(METI) investigated 33 alleged cases, and the NSC investigated 6 alleged cases. Licensee's violation of legislations and regulations can be discovered early by the allegation system and nuclear disaster is expected to be prevented.*

The Minister of METI is required to report quarterly the status of regulatory activities for approval of the Operational Safety Program and their change to the NSC.

For commercial power reactors, the provisions of Electricity Utilities Industry Law are applied for regulations on the detailed design for construction of reactor facilities (approval of design and construction methods), inspections at the time of construction of reactor facilities (approval of the welding method, welding inspection and pre-service inspection), and inspection after commissioning (facility Periodic Inspection), and it is prescribed in the Reactor Regulation Law that the corresponding provisions of the Reactor Regulation Law are exempted from application.

2) Electricity Utilities Industry Law

The Electricity Utilities Industry Law was established to ensure safety of public and to preserve environment by regulating construction, maintenance and operation of electric facilities, providing safety regulations for that purpose.

In view of ensuring safety of electric facilities used for electric utilities industry, the provisions on the Approval of Construction Plan, Fuel Assembly Inspection, Audit of Licensee' s Welding Check System, Pre-service Inspection, Periodic Licensee's Check, Audit of Licensee's Periodic Check System, Periodic Inspection, Operational Safety Program for Nuclear Facilities are defined in the Electric Utilities Industry Law.

The Minister of METI is required to make quarterly reports to the NSC on the status of regulatory activities, such as Approval of Construction Plan, Pre-service Inspection, Fuel Assembly Inspection, Periodic Inspection, and Audit of Licensee's Periodic Check System.

3) Basic Law for Emergency Preparedness and Special Law for Nuclear Emergency

The nuclear emergency had been addressed within the legal framework of the Basic Law for Emergency Preparedness. Taking account of the special characteristics of a nuclear emergency, the Special Law for Nuclear Emergency was established in December 1999.

The Law stipulates special measures for nuclear emergency, including licensee's obligation for preventing nuclear emergency, the Declaration of Nuclear Emergency, and establishment of the Nuclear Emergency Headquarters, as well as activation of emergency measures in nuclear emergency. It also stipulates that Senior Specialists for Nuclear Emergency be stationed in the vicinities of nuclear installations, which guides and advises licensees in preparing preventive measures for nuclear emergency, and conducts other activities necessary to prevent the occurrence and progression of a nuclear emergency.

The volume for nuclear emergency preparedness in the Basic Plan for Emergency Preparedness in accordance with the Basic Law for Emergency Preparedness, clarifies the measures to be activated at each step of the occurrence of an abnormal event, progression into a nuclear emergency, and recovery from the emergency.

4) Laws for Radiation Protection

The radiation protection at nuclear installations is regulated by the Reactor Regulation Law, the Electricity Utilities Industry Law and the Industrial Safety and Health Law.

The Reactor Regulation Law stipulates zone control for radiation protection, dose control of personnel engaged in radiation work, measurement and monitoring of radiation levels, etc. in order to protect personnel and the public. The Electricity Utilities Industry Law prescribes the radiation instrument devices to be installed in nuclear installations. The Industrial Safety and Health Law defines the dose limits of personnel engaged in radiation work, which are equivalent to the Reactor Regulation Law. In accordance with the Law for Technical Standards of Radiation Hazards Prevention, the Radiation Council was established to take a consistency among technical standards for radiation hazards.

In order to prevent hazards due to the use of radioisotopes at nuclear installations, the Law Concerning Prevention from Radiation Hazards to Radioisotopes, etc.(hereinafter referred to as "Radiation Hazard Prevention Law") stipulates zone control of radiation protection, dose control of personnel engaged in radiation work and radiation measurement in controlled area etc.

Relevant legislations were revised *incorporating* the ICRP Recommendation 1990 and enforced in April 2001.

5) Environment Impact Assessment Law

The Environmental Impact Assessment Law was enacted in June 1999, replacing the Decision of the MITI Departmental Council, July 1977, which stipulated the environmental impact assessment of nuclear installation other than safety assessment. *The environmental impact assessment is implemented in accordance with the law.*

The objective of the Environmental Impact Assessment Law is for licensees to perform proper assessment of a large business plan which may pose large impact on the environment, and to prepare proper plan to preserve the environment. The law provides a set of procedures for it. Environmental assessment on commercial power plants, including nuclear installations, is performed in accordance with the provisions of the Environment Impact Assessment Law and the corresponding provisions of the Electricity Utilities Industry Law. The environmental impact assessment is obligatory for nuclear installation regardless of its scale.

6) Law on Compensation for Nuclear Damage

The Law on Compensation for Nuclear Damage establishes the basic system on compensation for nuclear damage caused by a nuclear accident.

The Law adopts the "liability without fault" principle and imposes sole liability of compensation for nuclear damage on licensees, exempting claimants from proving licensee's fault in accordance with the general principle of the Civil Law. Also, infinite liability of compensation is imposed on the licensee.

To secure the fund of and to facilitate the compensation, the licensee is required to make the Financial Arrangement for Nuclear Damage Liability. The amount of the Arrangement is sixty billion yen for a nuclear installation in general.

The Arrangement consists of the Nuclear Damage Liability Insurance Contract with a civil insurer and the Indemnity Agreement for Compensation with the national government. The latter supplements the former in the case of large-scale accident such as caused by earthquake or volcanic eruption.

And in case the total amount arranged by the licensee is not sufficient for full compensation, the national government, on the basis of decision by the Diet, would aid to cover the licensee. In the case of enormous natural disaster or social disturbance, the national government bears the compensation, exempting licensee from liability for compensation.

(4) Provisions on Technical Requirements in the Safety Regulations

Technical requirements that will be used for review of basic design of nuclear installations are provided in the Guides for safety design and safety assessment established by the Nuclear Safety Commission. The system of the Guides is shown in Table 7-1. These Guides are used to assess the adequacy of licensee's application for the license for establishment at the safety review and assessment of the application.

For technical requirements necessary for regulations at the subsequent stage such as approval of construction plan and the pre-service inspection after approval of establishment of a nuclear installation and the Periodic Inspection after commissioning, the Nuclear and Industrial Safety Agency (NISA) has established the technical standards as performance requirements. This system of technical standards is shown in Figure 7-2.

Furthermore, NISA uses the standards of the academic societies and associations that are standards of the private sectors, as the specification requirements to realize the performance requirements, after doing the technical evaluation. Major standards of academic societies and associations are shown in Table 7-2.

NISA also defines the internal regulatory guides, such as guidelines of various evaluations and review procedures.

As revisions of guides carried out after the previous report, the revised Regulatory Guide for Reviewing Seismic Design was issued in September, 2006 aiming at further improvement of seismic safety and reliability of nuclear installations (see Article 18). In order to verify the seismic safety of existing nuclear installations based on the revision, licensees were required to implement the seismic safety evaluation referring to the revised Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities.

As revisions of technical standards carried out after the previous report, the technical standards that included specification requirements was revised to the ones that provide performance requirements only, and standards of academic societies and associations are used for providing specifications after evaluating the standards technically.

7.2 Legislative and Regulatory Framework at Each Stage

The overview of safety regulations on the basis of the Reactor Regulation Law, the Electricity Utilities Industry Law, etc. from planning stage through operation stage is shown in Figure 7-3. A summary of safety regulations for a commercial power reactor is stated in this section.

(1) Planning Stage

When selecting a site for a nuclear installation, the electric utility, on the basis of the Environmental Impact Assessment Law and the Electricity Utilities Industry Law, performs environmental impact assessment, and submits to METI the draft Environmental Impact Statement (draft EIS) explaining current status of the environment and measures to protect it. The draft EIS is sent to the related local governments to be disclosed for public comments. The utility prepares their views addressing residents' comments. Assessments on air, water, and soil pollution due to radioactive substances are performed under the Reactor Regulation Law and exempted from application of the Environmental Impact Assessment Law.

METI conducts the evaluation, soliciting experts' opinion.

In order to have opportunities to invite the opinions from the general public widely on various issues concerning establishment of the installation, the first public hearings with explanation of the electric power company are held by METI to obtain deeper understanding and cooperation of residents in the vicinity. The results of public hearings are taken into consideration in the safety examination..

(2) Establishment Stage

The license applicant, having completed the procedure of planning stage, submit application format for a license for establishment to the Minister of METI in accordance with the Reactor Regulation Law. Applicants attach documents to the application format including a description on safety design of the nuclear installation, radiation control, and accidents and failures.

NISA conducts an examination to determine the adequacy of the site, and the basic design of structure and equipment from the points of prevention of radiological hazards, focusing on the evaluation of the safety of the reactor core and the radiation exposure due to establishment of the nuclear installation. In addition, the regulatory body confirms that the nuclear installation should be used for peaceful purpose and in line with the planned development and utilization of nuclear energy, and the applicant has sufficient technical capability to ensure safety and sufficient financial basis to execute the plan.

In this examination, the regulatory guides in Table 7-1 and other documents established by the NSC are used. In the examination, site surveys, and analysis are conducted, when necessary.

The Minister of METI consults with the Atomic Energy Commission and the NSC on the results of its examination. During the review process of METI's results, the NSC reviews independently focusing on safety problems specific to the installation, and *gives its views* to the Minister of METI. The Minister of METI considers these views, asks for the consent of the Minister of MEXT, and then issues the license.

At the establishment stage, the second public hearing is held by the NSC to hear the opinions of residents in the vicinity on the safety specific to the facilities and take the opinions into consideration at the time when the NSC investigate and review the result of safety review and assessment by METI for the application of reactor establishment or alteration license applied by the electric power company. At the second public hearing, METI will explain the overview of safety review and assessment and present the view on the stated opinions. And NISA is conducting public relations activities positively to residents as provided in 8.3 (4) of this report.

In addition to these activities, Nuclear and Industrial Safety Subcommittee of the Advisory Committee for Natural Resources and Energy of METI invites opinions (Public Comments) when they formulate or alter fundamental policies, or newly introduce or alter the system which might affect the rights and duties of the people. When the NSC makes important policy decision or when Committee on Examination of Reactor Safety investigates and reviews the safety review and assessment, its executive office opens the contents of issue items for a fixed period of time to the public, and invites the opinion from the general public.

(3) Construction Stage

In accordance with the Electricity Utilities Industry Law, the licensee submits the Construction Plan for establishment of electric facilities, and obtains an approval of the

Minister of METI before starting construction works. NISA examines the Construction Plan to confirm that the detailed design of electric facilities is consistent with the basic design and design policies approved at the stage of licensing for establishment, and is in conformity with the technical standards based on the Electricity Utilities Industry Law. And the licensee must designate Chief Electrical Engineers and Chief Engineers of Boiler and Turbine and notify NISA of it.

After obtaining an approval or notification of the Construction Plan, the licensee shall undergo the Pre-Service Inspection by NISA at each process of construction and at the completion of all construction works, which confirms that construction is conducted in accordance with the construction plan and is in conformity with the technical standards. The licensee shall obtain design approval by NISA for fuel assemblies to be loaded in the reactor and undergo the Fuel Assembly Inspection conducted by NISA. The licensee shall perform Licensee’s Welding Check for welding of pressurized parts and containment vessels and shall undergo review by JNES on the organization conducting the inspection, the inspection method, schedule control, and other items provided by the ordinance of METI (Audit of Licensee's Welding Check System).

(4) Operation Stage

At the start of operation, the licensee must notify NISA of the operation plan, obtain an approval of the Operational Safety Program that prescribes procedures of operational management, operational limits and safety education of personnel, designate Chief Reactor Engineers, Chief Electrical Engineers and Chief Engineers of Boiler and Turbine, who supervise the safety of the operation, and the Persons Responsible for Operation, and notify NISA of them. The licensee is required to notify NISA of the operation plan annually.

The 17 items prescribed in the Operational Safety Program are provided in Reactor Regulation Law, which includes the Periodic Assessment, quality assurance, maintenance management, etc. of nuclear installations

The licensee must control the radiation exposure of personnel engaged in radiation work so that their doses do not exceed the dose limit, and shall report the exposure dose of personnel to NISA periodically.

Licensees must discharge gaseous and liquid radioactive waste generated during operation into the environment, in compliance with the concentration values which are lower than the concentration limits stipulated in the Reactor Regulation Law. Licensees must make efforts to reduce discharge amount as small as possible so that annual public exposure in the vicinity will be kept below 50 μ Sievert in accordance with the Regulatory Guide for the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities (hereinafter referred to, as the “Regulatory Guide for Annual Dose Target”).

After starting operation, the licensees, in accordance with the Electricity Utilities Industry Law, shall perform the Periodic Licensee’s Check to confirm that the installations conform to the technical standards. *The records of the Periodic Licensee’s Check have to be stored by the* *licensee for five years after decommissioning of the electric facilities. And fitness-for-service assessment is required to important part such as reactor pressure boundary. And also, the licensee must* undergo the Periodic Inspection by NISA on the *specified* part of structures important to safety. The Periodic Inspection and the Periodic Licensee's Check are conducted during shutdown of operation within the interval not exceeding 13 months from the date of start of operation or the date of completion of the previous inspection. Since October 2003, JNES has conducted part of the Periodic Inspection and notifies NISA of the results on the basis of the revision of the relevant laws. Also, licensees must undergo the Periodic Safety Management Review, in which JNES reviews the licensee's organization conducting the Periodic Licensee's Check, inspection method, schedule control, and other items provided in the ordinance of METI, *and report the result to NISA for the evaluation. NISA shall evaluate the result comprehensively.*

Licensees must undergo the Operational Safety Inspection by the Nuclear Safety Inspectors on the observance of the Operational Safety Program, including the licensee's organization, quality assurance, maintenance management, operation, maintenance and repair of component, surveillance, radiation control, and management of radioactive wastes, discharge control of gaseous and liquid radioactive wastes, monitoring, and safety education. NISA conducts the On-site Inspection of nuclear installation to confirm compliance with safety regulation, if necessary.

If any failure occurs in a nuclear installation, the licensee must report the failure etc. immediately to NISA in accordance with the provisions of the Reactor Regulation Law and the Electricity Utilities Industry Law, and shall report to NISA, without delay, on the situation of the failure and the measures taken. In order to improve transparency of information to the public, the reporting criteria for failures etc. were more clearly defined by amending the Reactor Regulation Law in October 2003. Licensees have established the system to collect information on events, including minor events that are outside of the reporting criteria, and disclose them to the public.

Criteria for necessity of approval or notification of the Construction Plan for any modification or repair work of electric facilities after startup of operation was clarified by the amendment of the Rules for the Electricity Utilities Industry Law in October 2003. NISA established the "Regulatory Guide on the Construction Plan" to identify the details of the amendment and notified licensees of it.

MITI (present METI) issued, in 1992, a Decision of the METI Departmental Council to request licensees to voluntarily perform the periodic safety review at a regular operating interval (approximately every ten years), including incorporation of operating experiences from commissioning to date and the latest technological knowledge, and probabilistic safety assessment. On the basis of the amendment of the Reactor Regulation Law, in October 2003, the periodic safety review at a regular interval (approximately every ten years) was incorporated into the Operational Safety Program, the observance of which the Nuclear Safety Inspector inspects at the Operational Safety Inspection. *In December 2005, NISA decided to include review with respect to enhancement of Measures for Aging Management and* *degradation of licensee's organizational climate. The implementation of probabilistic safety assessment, however, remains to be a voluntary activity of licensees as yet.*

On the basis of the amendment of the Reactor Regulation Law, in October 2003, licensees were obliged to perform technical evaluation on ageing of nuclear installation before continuous operation more than thirty years and must prepare a ten-year maintenance plan based on the technical evaluation. The subsequent evaluation should follow within ten years. *In December 2005, NISA issued the Performance Guidelines for Measures for Aging Management and the Standard Review Procedures for Measures for Aging Management to complete measures for aging management.*

7.*3 Record,* Applicable Regulations and Enforcement of Terms of License

Licensees are obliged to record and save required items for operation and use of reactor for every nuclear reactor. The concrete items and the period are provided in the Rules on Establishment and Operation of Commercial Power Reactor.

In accordance with the Reactor Regulation Law, the Minister of METI may revoke the license for establishment or issue a Shutdown Order of nuclear installation for up to one year, under circumstances such as operating a nuclear installation without a license for establishment, violating an order legally issued by NISA, failing to implement measures necessary for safety prescribed by NISA, or failing to obtain approval for the Operational Safety Program.

The Reactor Regulation Law also prescribes imprisonment and/or fines under circumstances such as establishing a nuclear installation without a license for establishment, violating a Shut-Down Order, or failing to take relevant emergency measures, which are prohibited by the Law. NISA may order changes in the Operational Safety Program whenever it is deemed necessary for preventing potential radiological hazards. Licensees failing to abide by such orders would be punished with a fine.

In accordance with the Electricity Utilities Industry Law, if it is judged for an electric facility not to conform to the technical standards, the Minister of METI may order repair, alteration, relocation, temporary suspension of usage, or limitation of usage.

The Electricity Utilities Industry Law prescribes fines if a licensee violates a technical standard order for conformity, or establishes or alters an electric facility without obtaining necessary approval for a construction plan, or uses an electric facility without undergoing or passing the Pre-Service Inspection *or* the Fuel Assembly Inspection, *or fail to receive the Audit of Licensee's Welding Check System without performing Licensee's Welding Check.* It also prescribes to revoke the business license, if an electric utility violates the law or orders based on the law causing serious damage to the public benefits.

On the basis of the amendment of the Reactor Regulation Law in October 2003, when an employee violates a law and is punished by a fine, the legal person who legally employs him or her is also punished by a fine as heavy as 100 times of the employee's fine, to prevent organizational illegal acts.

7.4 Change the Technical Standards for Nuclear Installations to Performance Requirements

The technical standard based on the Electricity Utilities Industry Law defines the technical requirements for nuclear installations. The previous technical standard included not only the performance requirements but also detailed specifications. The technical standard has been revised so that it only provides performance requirements which are necessary for safety and the detailed specifications for conforming to these performance requirements are to be defined by using standards of academic societies and associations..

The background, present status, basic policies to use, conditions as regulatory standards and methods of verification to change the technical standards including only performance requirements and the use of standards of academic societies and associations are as follows:

- *Background*

  The Japanese previous technical standards included detailed and concrete "specification requirements" on structures, materials and dimensions. Though those standards had advantages that the requirements are clear and judgment of success or failure can be made clearly and fairly, but, on the other hand, had disadvantages that the flexible responses to technological innovation or latest knowledge was not easy. This disadvantages will be solved, if the regulatory body defines "performance requirements", describing only objectives or functions of safety equipments or facilities to achieve safety levels to be considered as necessary by the regulatory body, and detail specifications conforming to the performance requirements are defined separately by using "detailed specifications" established by specialists of each field by collecting their knowledge. In recent years, it has been socially demanded that the technical standards shall be defined as "performance requirements" as much as possible in order to reflect promptly domestic and overseas operating experiences, latest knowledge etc. into "detailed specifications"

- *Present status*

  In Japan, there are standards of private sectors which have been established independently by industries such as guidelines and rules of the Japan Electric Association (JEAC, JEAG etc.), and most of them have been used by licensees. However, they were not included in the regulatory standards officially and remained as reference information. These standards have been regarded as "independent standards of private sectors" established by industries. Recently, academic societies and industrial associations, such as the Japan Society of Mechanical Engineers, the Atomic Energy Society of Japan as well as the Japan Electric Association establish the standards of private sectors with emphasis on fairness, justice and openness taking a process to be reviewed by neutral and fair members in the public. And these standards are decided to be used officially, calling "standards of academic societies and associations."

- *Basic policies to use the standards of academic societies and associations by NISA*

(1)Regulatory standards specify performance requirements, and standards of academic societies and associations are positively used to establish specifications to realize the performance requirements.

(2)When the standards of academic societies and associations used by licensees are proved to be the standards which conform to the performances required by NISA, NISA shall open the fact after technical evaluation of the Standards.

(3)When applicable standards of academic societies and associations do not exist, NISA encourages its establishment, and until establishment of the standard, the performance, that will be realized by the fulfilling the conventional specifications, is regarded as the performance standards required by regulatory standards.

- *Conditions of standards of academic societies and associations as regulatory standards and its verification*

When the standards of academic societies and associations are confirmed to fulfill the following conditions, it is regarded that the regulatory standards are satisfied;

(1)It corresponds to the items representing performances required by the regulatory standards,

(2)Concrete methods or specifications on technical matters necessary to achieve performances required by the regulatory standards are provided, and

(3)Technical adequacy has been proven for the concrete methods or specifications provided in the standards of academic societies and associations.

No matter what organization of academic societies and associations have established the standards ,they are judged to satisfy the performance required by the regulatory standards, if it can be confirmed that they have been established by the process with emphasis on fairness, justice and openness and that they fulfill the above three conditions. The regulatory body make judgment promptly, in order to make the regulation effective and efficient with respect for the technical knowledge of specialists participated in the development process.

- *Present status regarding use of standards of academic societies and associations and future policy*

An amendment of the Ordinance of Establishing Technical Standards for Nuclear Power Generation Equipment (Ordinance of METI, No. 62, 1965), which is for the purpose to execute the above mentioned policies, was promulgated on July 1, 2005 and enforced on January 1, 2006 after announcement to the World Trade Organization (WTO) and invitation of public comments.

In practicing the above ministerial order, when corresponding standards of academic *societies and associations exist, NISA clearly specifies the application of them in the "Interpretation" of the Technical Standards, Ordinance No. 62, after technical evaluation, and when corresponding standards of academic societies and associations do not exist, the items which NISA requires are shown in the "Interpretation".*

As of March 31, 2007, 21 standards of academic societies and associations were evaluated technical adequacy by NISA.

NISA will perform further technical evaluation of standards of academic societies and associations whenever they are prepared.

Table 7-1 Major Regulatory Guides *Specified by the NSC* for Power Generating Light Water Reactors

Hazards Prevention	Siting	・ Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and Application Criteria
	Design	・ Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities ・ Regulatory Guide for Reviewing Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities ・ Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities ・ Regulatory Guide for Reviewing Fire Protection of Light Water Nuclear Power Reactor Facilities ・ Regulatory Guide for Reviewing Radiation Monitoring in Accidents of Light Water Nuclear Power Reactor Facilities ・ Fundamental Policy to be Considered in Reviewing of Liquid Radioactive Waste Treatment Facilities
	Safety Evaluation	・ Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities ・ Regulatory Guide for Evaluating Core Thermal Design of Pressurized Water Cooled Nuclear Power Reactors ・ Regulatory Guide for Evaluating Emergency Core Cooling System Performance of Light Water Power Reactors ・ Regulatory Guide for Evaluating Reactivity Insertion Events of Light Water Nuclear Power Reactor Facilities ・ Regulatory Guide for Evaluating Dynamic Loads on BWR MARK-I Containment Pressure Suppression Systems ・ Regulatory Guide for Dynamic Loads on BWR MARK-II Containment Pressure Suppression Systems ・ Regulatory Guide for Meteorological Observation for Safety Analysis of Nuclear Power Reactor Facilities
	Dose Target	・ Regulatory Guide for the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities ・ Regulatory Guide for Evaluating the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities ・ Guide for Radiation Monitoring of Effluent Released from Light Water Nuclear Power Reactor Facilities
Technical Competence		・ Regulatory Guide for Reviewing Technical Competence of Nuclear Operators

Table 7-2-1 Standards etc. of Academic Societies and Associations

(Guidelines and Rules of the Japan Electric Association)

Number	Title
JEAC 4111-2003 (*)	Rules of Quality Assurance for Safety of Nuclear Power Plants
JEAC 4121-2005	*Guidelines for Application of Rules of Quality Assurance for Safety of Nuclear Power Plants (JEAC 4111-2003)- Operation Stage of Nuclear Power Plants-*
JEAC 4201-*2004*	Method of Surveillance Tests for Structural Material of Nuclear Reactors
JEAC 4202-2004	Drop-Test Method for Ferritic Steel
JEAC 4203-2004 *(*)*	Primary Reactor Containment Vessel Leakage Testing
JEAC 4205-2000	In-service Inspection of Light Water Cooled Nuclear Power Plant Components
JEAC 4206-*2004*	Methods of Verification Tests of the Fracture Toughness for Nuclear Power Plant Components
JEAC 4209-2003 (*)	Rules of Maintenance Management of Nuclear Power Plants
JEAC 4602-*2004 (*)*	Definitions of Nuclear Reactor Coolant Pressure Boundary and Reactor Containment Boundary
JEAC 4605-*2004 (*)*	Definitions of Engineered Safety Features and Related Systems of Nuclear Power Plants
JEAG 4101-2000	Guideline of Quality Assurance for Nuclear Power Plants
JEAG 4102-1996	Guideline of Emergency Measures for Nuclear Power Plants
JEAG 4204-2003	Guideline for Quality Control of Nuclear Fuel for Nuclear Power Plants
JEAG 4207-2004 (*)	Ultrasonic Examination Guideline for In-service Inspection of Light Water Cooled Nuclear Power Plant Components
JEAG 4208-*2005*	Eddy Current Test Guideline for In-service Inspections of Steam Generator Heat Transfer Tubes for Light Water Type Nuclear Power Plants
JEAG 4601-1987 *(*)*	Technical Guidelines for Aseismic Design of Nuclear Power Plants
JEAG 4601-S-1984 *(*)*	Technical Guidelines for Aseismic Design of Nuclear Power Plants: Classification and Allowable Stress
JEAG 4601-1991 *(*)*	Technical Guidelines for Aseismic Design of Nuclear Power Plants: Supplement
JEAG 4603-1992	Guideline for Design of Emergency Electric Power Supply Systems for Nuclear Power Plants
JEAG 4604-1993	Guideline for Design of Plant Protection Systems for Nuclear Power Plants
JEAG 4606-2003	Guideline for Radiation Monitoring for Nuclear Power Plants
JEAG 4607-1999 *(*)*	Guideline for Fire Protection of Nuclear Power Plants
JEAG 4608-1998	Lightning Protection Guidelines for Nuclear Power Plants
JEAG 4609-1999	Application Criteria for Programmable Digital Computer System in Safety–Related System of Nuclear Power Plants
JEAG 4610-2003	Personal Dose Monitoring for Nuclear Power Plants

JEAG 4611-1991	Guideline for Design of Instrumentation & Control Equipment with Safety Functions
JEAG 4612-1998	Guideline for Safety Grade Classification of Electrical and Mechanical Equipment with Safety Functions
JEAG 4613-1998	Technical Guidelines for Protection Design against Postulated Piping Failures in Nuclear Power Plants
JEAG 4614-2000	Technical Guidelines on Seismic Base Isolation System for Structural Safety and Design of Nuclear Power Plants
JEAG 4615-2003 *(*)*	Guideline for Design of Radiation Shielding for Nuclear Power Plants
JEAG 4616-2003	Technical Guideline for Design of Base Structures for Dry Cask Storage Buildings
JEAG 4617-2005	*Guideline for Development and Design of Computerized Human-Machine Interface in the Central Control Room*
JEAG 4618-2005	*Technical Guidelines for Aseismic Design of Steel Plate Concrete Structures – for Buildings and Structures*
JEAG 4801-1995	Guideline for Operating Manual of Nuclear Power Plants
JEAG 4802-2002	Guideline for Education and Training for Nuclear power Plant Operator
JEAG 4803-1999	Guideline for Operational Safety Preservation of Light Water Cooled Reactors

Table 7-2-2 Standards etc. of Academic Societies and Associations

(Guidelines and Rules of the Japan Society of Mechanical Engineers)

Number	Title
JSME S CA1-2005	*Standards for Nuclear Power Generation Equipment: Standards for Piping Wall Thinning Management (2005 enlarged issue)*
---	*Standards for Nuclear Power Generation Equipment: Standards for Piping Wall Thinning Management for Pressurized Water Reactor (2006)*
---	*Standards for Nuclear Power Generation Equipment: Standards for Piping Wall Thinning Management for Boiling Water Reactors (2006)*
JSME S NA1-2002 (*)	Standards for Nuclear Power Generation Equipment: Maintenance Standards (revised in 2002)
JSME S NA1-2004	*Standards for Nuclear Power Generation Equipment: Maintenance Standards (2004)*
JSME S NB1-2001 *(*)*	Standards for Nuclear Power Generation Equipment: Welding Standards
JSME S NC1-2001 (*)	Standards for Nuclear Power Generation Equipment: Design and Construction Standards
JSME S NC1-2005 (*)	*Standards for Nuclear Power Generation Equipment: Design and Construction Standards (2005)*
JSME S ND1-2002	Standards for Nuclear Power Generation Equipment: Design Standards for Prevention of Piping Break
JSME S NE1-2003 *(*)*	Standards for Nuclear Power Generation Equipment: Concrete Reactor Containment Vessel
JSME S NF1-2006	*Standards for Nuclear Power Generation Equipment: Evaluation Method of Environmental Fatigue (2006)*
JSME S FA1-2001	Standards for Spent Fuel Storage Facility: Structural Standard for Metallic Cask
JSME S FB1-2003	*Standards for Spent Fuel Storage Facility: Structure Standard for Concrete Cask, Canister Repack Device and Canister Transport Cask*
JSME S01 2 (*)	*Evaluation Guideline on Hydro-Dynamic Vibration of Piping Internal Cylindrical Structure*
JSME S014	*Standard for Validation of Active Component for Nuclear Power Plants*
JSME S016	Guideline for Prevention o f Fluid Induced Vibration of Tube and U-Tube of Steam Generator
JSME S017 *(*)*	Evaluation Guideline on High Cycle Thermal Fatigue of Piping
JSME S NA-CC-001	*Standards for Equipment, Maintenance and Case of Nuclear Power Generation Facilities: SCC Crack Propagation Velocity of Nickel*

	Rich Alloy in PWR Primary Water Environments
JSME S NA-CC-002	*Standards for Equipment, Maintenance and Case of Nuclear Power Generation Facilities: Alternative Provision for Allowable Defect Angle Restriction for the Defect of Peripheral Direction*
JSME S NA-CC-003	*Standards for Equipment, Maintenance and Case of Nuclear Power Generation Facilities: Criteria for Volumetric Examination (Eddy Current Test) of a Steam Generator Tube*
JSME S NC-CC-001(*)	*Standards for Equipment, Design and Construction and Case of Nuclear Power Generation Facilities: Standard for Protection of Over-pressurization*
JSME S NC-CC-002(*)	*Standards for Equipment, Design and Construction and Case of Nuclear Power Generation Facilities: Standard: Considerations for Inhibition of Occurrence of Stress Corrosion Cracking*

Table 7-2-3 Standards etc. of Academic Societies and Associations

(Guidelines and Rules of the Atomic Energy Society of Japan)

Number	Title
AESJ-SC-P001:2002	Procedure of Probabilistic Safety Evaluation on Shutdown Condition of Nuclear Power Station
AESJ-SC-P002:2003	Evaluation Criteria of Fuel Integrity after Transient Boiling Transition for BWR
AESJ-SC-P003:2003	Performance Criteria of Wind Tunnel Test to obtain the effective height of Discharge Source
AESJ-SC-P004:2006	*Performance Criteria of the Periodic Safety Review for Nuclear Power Plants (2006)*
AESJ-SC-P005:2007	*Performance Criteria of the Measures for Aging Management for Nuclear Power Plants(2007)*
AESJ-SC-R003:2006	*Planning and Execution of Decommissioning of reactor facilities (2006)*
AESJ-SC-F001:2000	Periodic Inspection Criteria of Cask for Spent Fuel, MOX fuel and High Level Radioactive Waste
AESJ-SC-F003:2002	Measurement method of Sorption Distribution Coefficient— Basic Procedure of Batch Method for Barrier Material of Near Face Disposal
AESJ-SC-F005:2005	*Method of Judgment on the Clearance Level (2005)*
AESJ-SC-F006:2006	*Safety Design and Inspection Criteria for Transfer Container for Spent Fuels, New Mixed Oxide Fuels, and High Level Radioactive Wastes*
AESJ-SC-F007:2006	*Method of Safety Evaluation for Disposal of Very Low Level Radioactive Wastes (2006)*
AESJ-SC-F008:2006	*Method of Measurement of Sorption Distribution Coefficient – Basic Procedures of Measurement for Barrier Material for Deep Geological Disposal (2006)*
	Performance Criteria of Probabilistic Safety Assessment of Events Caused by Earthquake on Nuclear Power Plants (2007)

Table 7-2-4 Standards etc. of Academic Societies and Associations

(Guidelines and Rules of the Thermal and Nuclear Power Engineering Society)

Number	Title
TNS-S3121-2003	Qualification Standards for Industry Product on Weld of Electric Facilities
JBWR-NCG-01-2005 (*)	*Guidelines for Accumulation Prevention of Mixed Gas (Hydrogen and Oxygen) inside BWR Piping*

Note: * Standards etc. of academic societies and associations that NISA has evaluated their technical adequacy in order to utilize as exemplification standards of specification codes.

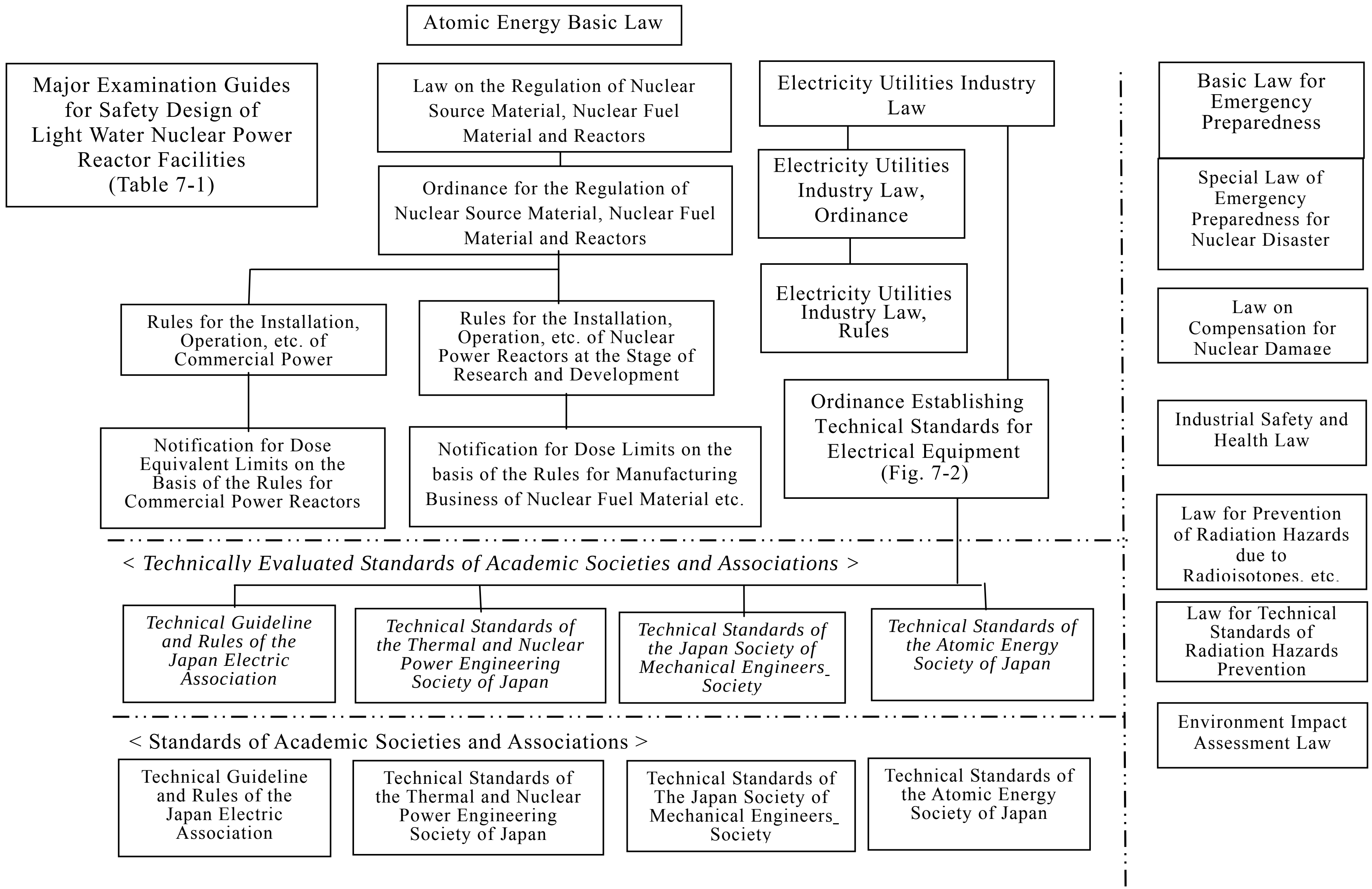

Fig. 7-1 Major Legislations Governing the Safety Regulation of Nuclear Installations

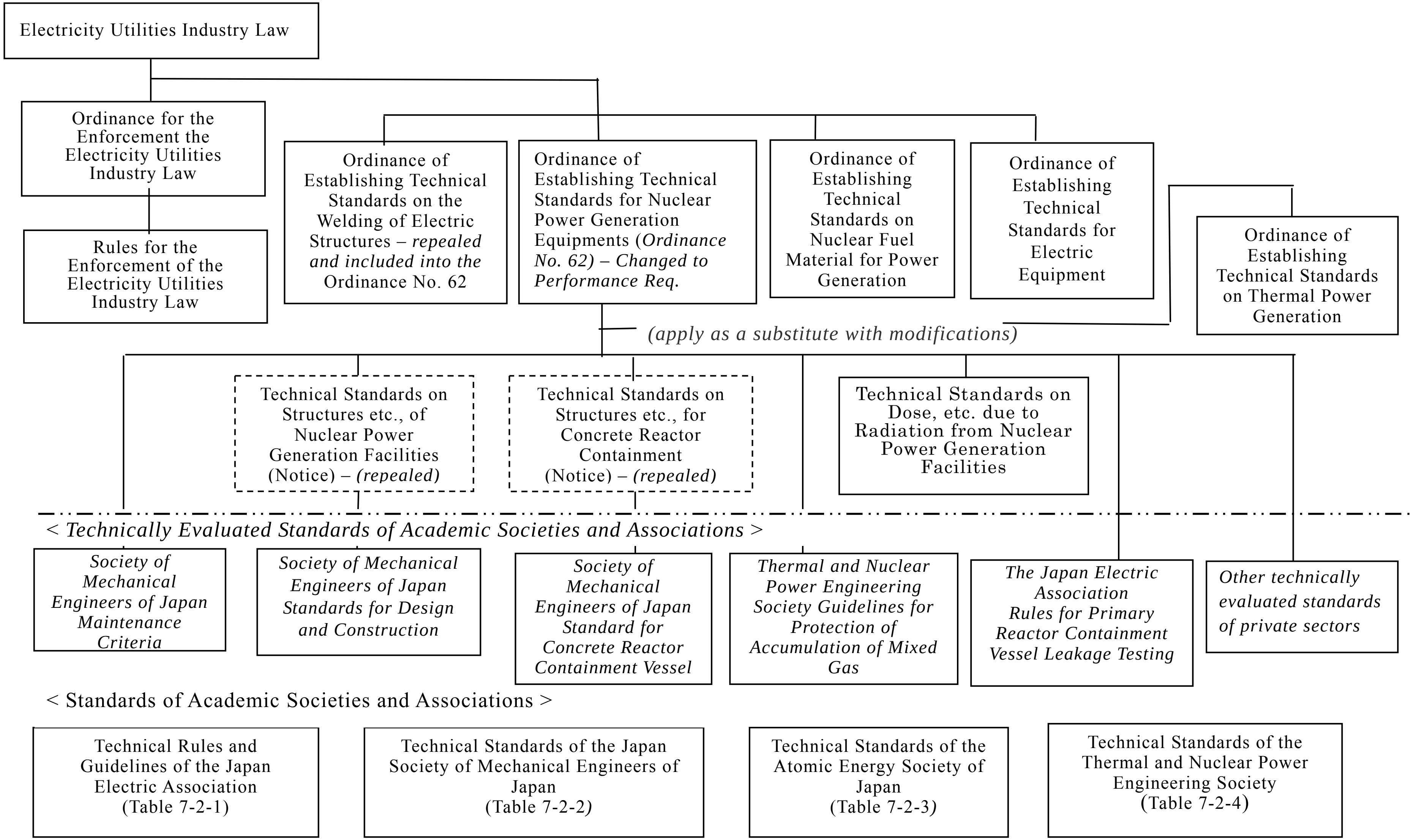

Fig. 7-2 Systems of Technical Standards

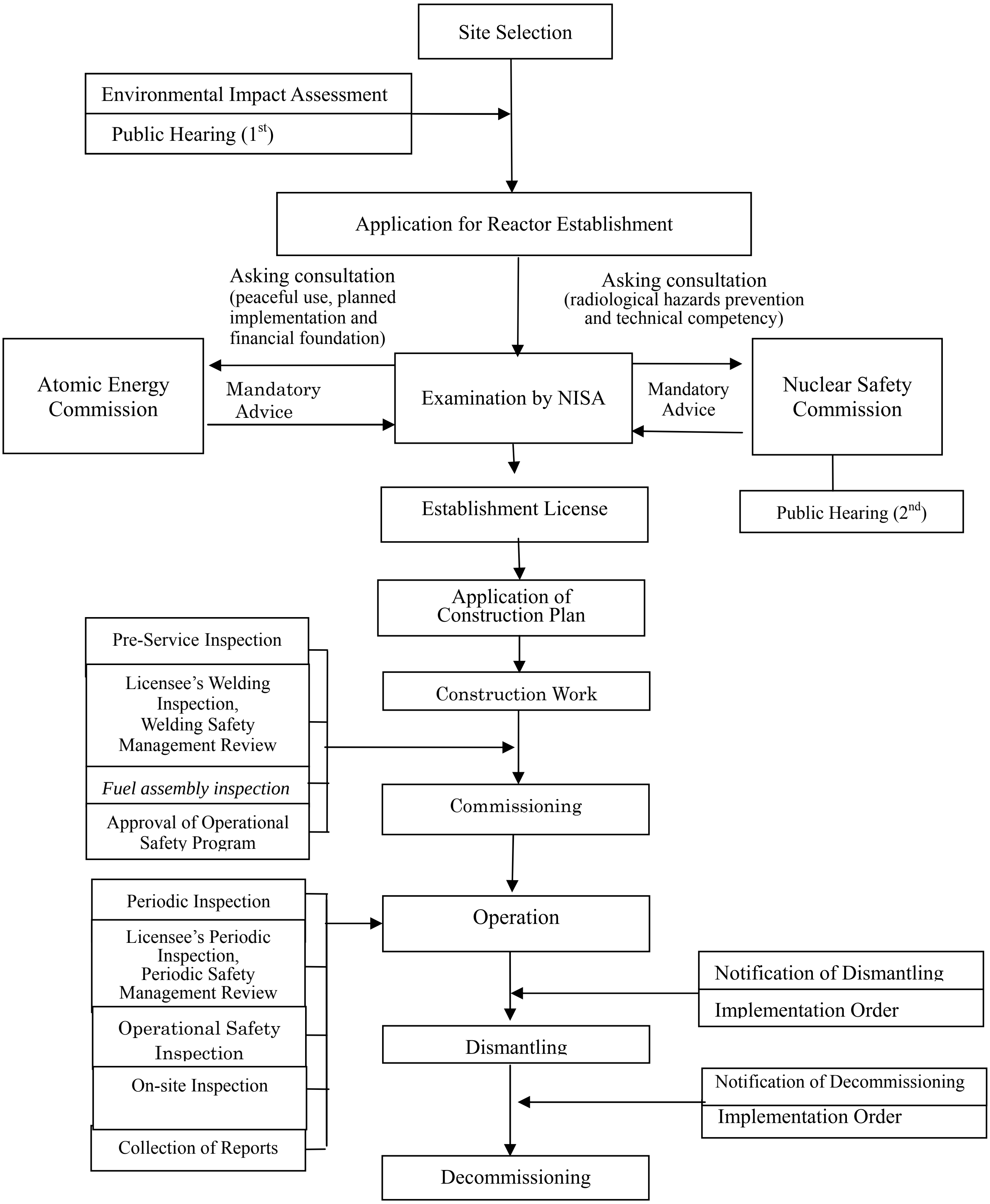

Fig. 7-3 Flow of Safety Regulations based on Legislations, etc. for Nuclear Installations

Article 8 Regulatory Body

1. Each contracting Party shall establish or designate the regulatory body entrusted with the implementation of the legislative and regulatory framework referred to in Article 7, and provided with adequate authority, competence and financial and human resources to fulfill its assigned responsibilities.
2. Each Contracting Party shall take the appropriate steps to ensure an effective separation between the functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy.

The regulatory body for ensuring the safety of facilities and activities for utilization of nuclear energy in Japan is the Nuclear and Industrial Safety Agency (hereinafter referred as "NISA"). NISA, based on The Law on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors etc. (hereinafter referred to as the "Reactor Regulation Law etc."), has definite authority and power on safety regulations. Moreover, the Nuclear Safety Commission (hereinafter referred to as the NSC") established in the Cabinet Office supervises and audits the regulatory activities of NISA. NISA is ensured to be effectively independent from other agencies or organizations that manage matters for the promotion of nuclear energy utilization.

In addition, the Japan Nuclear Energy Safety Organization (JNES) was established under the authority of METI, and is performing part of the Operational Safety Inspection for nuclear installations in accordance with the Reactor Regulation Law and the Electricity Utilities Industry Law, and is supporting NISA.

Since the previous report, NISA has conducted the Audit of Licensee's Periodic Check System etc. in order to promote firm establishment of the inspection system revised in 2003, and has reported the results of the said Audit to the NSC. Also, corresponding to the NSC's result of supervision and audit on regulatory activities of NISA, NISA is making an effort to reform the regulations further. NISA also introduced the NISA Work Management System for improving the transparency and efficiency of NISA's work.

In addition, NISA, together with the NSC, invited the Integrated Regulatory Review Services (IRRS) by the International Atomic Energy Agency (hereinafter referred to as the "IAEA") in 2007, in order to receive an international review concerning the nuclear-safety-regulatory activities. Concerning the independency of the regulatory body from promotion organizations, the IRRS showed that "NISA is effectively independent from the Agency of Natural Resources and Energy (hereinafter referred to as the "ANRE") in correspondence with the GS-R-1 (international standard of the IAEA). Moreover, this situation could be reflected in the legislation more clearly in future." [1]

[1] The IRRS Report is under the preparation by the IAEA at the time of this publication.

8.1 Mandate and Duties of the Regulatory Body

The mandate of the regulatory body is to ensure the safety of nuclear installations, and its duties are to enforce the legislative and regulatory framework described in the report of Article 7.

One of the important requirements for the regulatory body satisfying his responsibility is, as indicated in Article 8, Paragraph 2 of this Convention, to ensure effective separation between functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy. Another important function of the regulatory body is to keep communicating independently with the public of its regulatory decisions, its opinions and its basis.

On the basis of the Atomic Energy Basic Law, the regulatory body is responsible to conduct regulatory activities prescribed in the Reactor Regulation Law, the Electricity Utilities Industry Law, etc. As for legislations and regulations etc. applied to the examination of the basic design or basic design policies of nuclear installations and to the inspection of nuclear installations in the construction and operational stages, NISA and the NSC work toward improvement and enhancement of legislations and regulations based on operating experiences, trends of the latest knowledge of the technology advancement, etc., and the international consensus.

In the case of a nuclear emergency, the Basic Law on Emergency Preparedness, the Special Law of Emergency Preparedness for Nuclear Disaster and other related laws are applied. Relevant administrative bodies in such a case are described in the report of article 16.

8.2 Organizations for Enforcement of the Safety Regulation of Nuclear Installations

In Japan, the Minister of Economy, Trade and Industry (hereinafter referred to as "METI") serves as the minister in charge of safety regulation for all facilities and activities concerning the utilization of nuclear energy. NISA has been established in METI as an independent "special organization" dedicated to the administration of safety regulations.

NISA has been executing policies independently from the Agency of Natural Resources and Energy dedicated to promote nuclear energy. The incorporated administrative agency, JNES was established in October 2003. JNES, together with NISA, provides infrastructure to assure safety in the use of the nuclear energy.

The NSC and the Atomic Energy Commission (AEC) were established in the Cabinet Office. The commissioners of both of these commissions are appointed by the Prime Minister with the consent of the Diet.

Each of these two commissions' plans, deliberates, and decides policies concerning either nuclear power application or ensuring the safety, from the standpoint to regulate respectively in the country as a whole.

As described in the report of Article 7, NISA conducts a safety examination of nuclear installations, and the Minister of METI consults the NSC and the AEC on the results of the examination.

The NSC submits to the Minister of METI a report, after an independent examination and public hearings on the specific safety of the nuclear installation. The NSC establishes guides to be used for the examination.

Fig. 8-1 presents an overview of administrative organizations that are responsible for the safety regulation of nuclear installations.

8.3 Nuclear and Industrial Safety Agency (NISA)

(1) The Role of NISA

NISA administrates the safety regulations for nuclear installations. Specifically, NISA, entrusted by the Minister of METI, conducts clerical work concerning the competence of the Minister of METI as follows:

The Minister of METI, who is the competent minister stipulated in the Reactor Regulation Law, has the authority to issue licenses for establishment of nuclear installations, after conducting the examination of siting, structure, and equipment, so that a radiological hazard due to establishment of a nuclear installation is prevented. The Minister of METI has the authority to revoke a license under circumstances such as violation of the Reactor Regulation Law by the licensee.

The Minister of METI has the authority to establish ministerial orders concerning measures for the safe operation and physical protection of specific nuclear fuel material, the Operational Safety Program, Chief Engineer of Reactors, measures in emergency, etc. for the operation of reactors. The Minister of METI has the competence of authorizing the Operational Safety Program, approval of the operation plan and decommissioning plan of nuclear installations, accepting the notification concerning appointment/dismissal of Chief Engineers of Reactors, collecting reports from licensees and conducting On-site Inspection of the licensees, revocation or discontinuance of utilization of a license for establishment of a nuclear installation, ordering of measures for safe operation etc., approval of a Chief Engineer of Reactors, an order on measures for safe operation etc., dismissal order of a Chief Engineer of Reactors, implementation order concerning a decommissioning, implementation order for an emergency preparedness, etc.

The Minister of METI, and the Minister of MEXT, conducts examinations for Chief Engineers of Reactors and issues the licenses. The Minister of METI has the authority also to order to return such licenses in a case of violation of the law by the Chief Engineers.

The Minister of METI, who is the competent minister stipulated in the Electricity Utilities Industry Law, has powers to establish ministerial ordinances relating to the technical standard, pre-service inspection, the fuel assembly inspection, the Audit of Licensee’s Welding Check

System, the Periodic Inspection, and the Audit of Licensee’s Periodic Check System. The Minister of METI has powers to conduct approval of the construction plan, the pre-service inspection including verification of the safety performance of a whole power station, the fuel assembly inspection, the Periodic Inspection, and to issue an Order for Conformity to the technical standards when a case of nonconformity to the technical standards is found. The Minister of METI has powers also to hold the affairs of examinations for Chief Electrical Engineers, to issue licenses for the Chief Electrical Engineer and the Chief Engineer of Boiler and Turbine, and to order the return of such licenses in case of violation of the law by a Chief Engineer.

NISA evaluates results of the Audit of Licensee’s Periodic Check System performed by JNES. In the Audit of Licensee's Periodic Check System, JNES evaluates the organization, inspection methods, process control, and other matters of a licensee concerning implementation of the periodic licensee's check, which are determined by a METI ordinance.

NISA evaluates also the results of the Audit of Licensee’s Welding Check System performed by JNES.

(2) Organization of NISA

NISA was established as a "Special Organization" in METI, and has 11 divisions dedicated to the administration of the safety regulation of nuclear installations (including nuclear fuel cycle facilities).

They are Policy Planning and Coordination Division, Nuclear Safety Public Relations and Training Division, Nuclear Safety Regulatory Standard Division, Nuclear Safety Special Investigation Division, Nuclear Power Licensing Division, Nuclear Power Inspection Division, *Nuclear Fuel Transport and Storage Regulation Division,* Nuclear Fuel Cycle Regulation Division, Radioactive Waste Regulation Division, Nuclear Emergency Preparedness Division and Electric Power Safety Division. The assigned duties of those divisions are provided in Table 8-1.

Nuclear Safety Inspectors are assigned to each site of the nuclear installations. Fig. 8-2 shows the locations of the Nuclear Safety Inspectors Offices.

NISA has a total of approximately *350* staff engaged in the nuclear safety regulation, out of which 100 staff members are Nuclear Safety Inspectors and Senior Specialists for Nuclear Emergency stationed at nuclear installations.

(3) Quality Improvement of NISA’s Regulatory Activities

NISA provides a strong commitment to its mission, scientific and reasonable judgments, transparency, neutrality and fairness as the code of conduct for their activities. In this context, the Policy Planning and Coordination Division watches and assesses the performance of other divisions of NISA in discharging their duties, and take timely remedial actions after consulting with the senior managements. *In order to improve the quality of regulatory* *activities, the development of the NISA Work Management System started in fiscal year 2006 and implemented from fiscal year 2007. According to the NISA Work Management System, NISA's goals in the medium term and tasks in fiscal year 2007 were released in June, 2007.*

The NSC, an independent organization from NISA, supervises and audits the appropriateness of NISA's regulatory administration in the construction and operation stages after issuance of the license, from the view points of rationality, effectiveness and transparency. Thus, the framework that confirms the quality of the safety administration is maintained.

In addition, NISA makes a continuous effort to maintain the high quality of regulation through education and training of the personnel as stated in the report of Article 11, international activities and the hearing of advice from experts e.g. members of the Nuclear and Industrial Safety Subcommittee (hereinafter referred to as the "NISS").

The "Law for Evaluation of the Policies Executed by Administrative Organizations" was enforced in April 2002, and in accordance with this law, a framework, with which each administrative organization of the government evaluates and improves his own policies systematically, was built. METI has developed plans to evaluate the regulatory systems within its jurisdiction in fiscal year 2004, and NISA, according to these plans, evaluates the nuclear safety regulation system on the basis of the Reactor Regulation Law and the Electricity Utilities Industry Law.

(4) Further Approach to Information Disclosure

NISA started information service activities systematically in the form of integrating with the regulatory work process in September 2001, introducing relationship management (RM) as a new effort, which makes the feedback from the outside into a qualitative operation of regulatory activities, and is promoting positive information disclosure activities. The objectives of the RM are to make effort for improvements in recognition of NISA's responsibility, in the people's understandings about NISA's daily activities, in ensuring a steady response to the people's concerns, in establishing consent to a better regulatory system, in the preparation to an emergency, and in the activation of internal communication.

In April 2004, NISA allocated a new budget in order to enhance further activities to hear from the public, and at same time, established the Nuclear Safety Public Relations and Training Division newly formed in NISA, and appointed the Resident Public Relations Officers. The main activities of NISA from fiscal year 2004 to fiscal year 2006 were as follows, (1) NISA executive's visit and give explanation of NISA's policy and activities to the local government (133, 113 and 64 visits in fiscal year of 2004 to 2006, respectively), (2) publication of newsletters and mail magazines, (3) explanation of policies and activities of the nuclear safety regulation to the general public, (public meeting on the clearance system and amendment of the Reactor Regulation Law, Pu-thermal symposium, public meeting on seismic safety, "one-day seminars to introduce NISA" were held in major cities and site municipalities), (4) making direct dialogue group communication with site area residents (in fiscal year 2006, "dialog meetings" were held at ten places across the country, such as Tomari-mura, Hokkaido *and Genkai-cho, Saga-Prefecture), (5) activities to hear from the public at the Nuclear Safety Inspector's Offices, (6) implementation of risk-communication technical training for local-government personnel etc., (7) introduction of NISA and Nuclear Safety Inspectors Offices; editing of a video explaining about essential policies, such as the new inspection system by managements themselves and televising by site area CATVs and placing such information on a homepage.*

Also, NISA opens the Nuclear Energy Library in JNES, where the public can access documents for the reactor establishment license, reports of incidents and accidents of nuclear installations and, books and booklets on energy and nuclear power generation.

8.4 Organizations related to NISA

(1) Council etc.

On the basis of the Law for Establishment of the Ministry of Economy, Trade and Industry, the Advisory Committee for Natural Resources and Energy (hereinafter referred to as the “ACNRE”) was established, a subcommittee of which is the NISS that proposes policies on nuclear safety and safety of electric power as terms of reference. The organization of the NISS is given in Table 8-2.

The Minister of METI appoints members of the ACNRE from persons of knowledge and experience, and these members select a chairperson of the ACNRE mutually. Subcommittees are established by a resolution of the ACNRE, and the chairperson designates members of the subcommittees including the NISS. The members of the subcommittees are assigned based on their expertise and experience from the fields of nuclear and thermal-hydraulic design, nuclear fuel design, system design, equipment design, seismic design, material strength, radiation control, meteorology, geology, soil etc.

"What challenges exist in the future in order to assure safety in nuclear power generation and safety in the electric power system operation, while under rapid social and economical change" were entrusted to NISS to be discussed. The NISS and other subcommittees have deliberated on what nuclear safety regulation systems should be, and the results were reported to NISA.

NISA solicits views of experts and members of NISS.

(2) JNES

JNES, consisting of about 420 officers and staff, was established in October 2003 as an organization that establishes the infrastructures in cooperation with NISA to ensure the safety of utilization of nuclear energy.

The mission of JNES is to implement its duties with full application of its technical and engineering competence based on scientific judgments to contribute to the improvement of nuclear safety regulation and, to deliver and transmit actively the safety information to the public.

JNES is expected to ensure the nuclear safety and build the confidence of the people in nuclear safety by implementing such duties.

JNES implements the following activities:

- Inspection of nuclear installations and reactor facilities, and related work,
- Safety analysis and evaluation of designs of nuclear installations and reactor facilities;
- Work for the establishment of nuclear emergency preparedness, prevention of the escalation of a nuclear emergency (including minimization of the probability of occurrence of a nuclear emergency), and restoration from a nuclear emergency;
- Investigation, testing, research, and training to ensure safety in utilization of nuclear energy; and
- Collection, analysis and delivery of information to assure nuclear safety.

The procedures for JNES to implement activities, keeping in relation with NISA of METI, are as shown in the following:

- NISA develops a goal on each activity based on the regulatory needs, and defines the medium-term objectives in accordance with the Act on General Rules for Incorporated Administration Agencies, and the Minister of METI assign them to JNES.
- JNES prepares a scheme (medium-term scheme) to accomplish the medium-term objectives, applies for and obtains the approval of the scheme to the Minister of METI,, then JNES prepares a program in accordance with the medium-term scheme for every FY term, notifies the program to the said minister and implements it.

8.5 The NSC

The Atomic Energy Basic Law was partially revised on October 4, 1978 to establish the NSC under the Prime Minister’s Office. The NSC administers the function of safety regulation, had belonged to the AEC up until then, in order to strengthen the system of ensuring nuclear safety. (The NSC was transferred from the Prime Minister’s Office to the Cabinet Office due to central government reform in January 6, 2001.)

The NSC is responsible for planning, deliberation and decisions on matters that are related to ensuring safety of the research, development, and utilization of nuclear energy.

The NSC conducts its own review of the results of NISA’s examination on the application from the view points of the licensee’s technical capability and non hindrance to the prevention of radiological hazards. *The NSC supervises and audits the appropriateness of NISA’s regulatory administration in construction and operation stages after issuance of the license, from the viewpoint of reasonableness, effectiveness and transparency. Thus, the framework* *that confirms the quality of the safety administration is maintained.*

When the NSC deems it necessary as a part of its assigned duties, the NSC may recommend and may request reports and cooperation concerning the submission of materials, statements of viewpoint, and explanation to the heads of relevant administrative organizations, by way of the Prime Minister.

Since April 2003 (partially, from October 2003), the above functions have legally been enacted. The NSC receives from NISA the following; reports on the quarterly bases after the approval of a license to establish nuclear installations: reports concerning the conduct of the regulatory activities such as approval of the construction plan, pre-service inspection, Periodic Inspection, Audit of Licensee's Periodic Check System, Audit of Licensee's Welding Check System, Approval of Operational Safety Program, implementation states of regulations, such as the Operational Safety Inspection, report of accidents and failures of nuclear installations. The NSC also has the authority to inquire directly of the licensees, maintenance and inspection contractors in order to supervise and to audit the safety regulation implemented by regulatory body.

In the case of a violation of safety regulations in any of nuclear facilities, the employee can directly allege the fact to the NSC, and the NSC has the authority to investigate the allegation.

The Minister of METI, before issuing a license to establish nuclear installations, must receive the viewpoint of the NSC on the following matters: (1) that the applicant for the license of a nuclear installation has adequate technical capability to establish and reliably operate a nuclear reactor, and (2) that the site, structures and equipment of the nuclear installation would not cause any hindrance to the prevention of radiological hazards.

The NSC is composed of five commissioners appointed by the Prime Minister with the consent of the Diet, and these commissioners elect a chairman among them. General affairs of the NSC are performed by the NSC Secretariat of the Cabinet Office. The NSC Secretariat is composed of the Secretary-General, the General Affairs Division, the Regulatory Guides and Review Division, the Radiation Protection and Accident Management Division and the Subsequent Regulation Review Division and has about 100 personnel.

Under the NSC, two safety examination committees, eight special committees and seven others are organized as shown in Table 8-3. The Special Committees may organize working groups under them, if necessary.

The members of the Committee on Examination of Reactor Safety and the Committee on Examination of Nuclear Fuel Safety are appointed from persons of knowledge and experience by the Prime Minister in accordance with the Law for Establishment of the Atomic Energy Commission and the Nuclear Safety Commission. The Emergency Technical Advisory Body is composed of the commissioners of the NSC and the commissioners on the Emergency Technical Advisory Body who are also appointed by the Prime Minister from persons of knowledge and experience.

Results of the investigation and evaluation by each review board and special committee are reported to the NSC and are deliberated by the NSC. Reflecting the results of the discussion in the Emergency Technical Advisory Body, the NSC determine the recommendation items for an emergency.

Deliberations of all committees, including the special committees and working groups under the NSC are open to the public. The contents of the deliberations are provided for the public on a homepage (http://www.nsc.go.jp/) and at the Nuclear Energy Library.

8.6 The AEC

The AEC was established under the Prime Minister's Office on January 1, 1956, on the basis of the Atomic Energy Basic Law and the Law for Establishment of the Atomic Energy Commission and the Nuclear Safety Commission, to conduct national policy concerning research, development and utilization of nuclear energy in a planned manner and to ensure the democratic administration of the nuclear energy policy. (The AEC was transferred to the Cabinet Office in January 2001.)

The AEC has duties of planning, deliberation, and decisions concerning the research, development and utilization of nuclear energy (excluding matters relating to regulations on ensuring safety).

If the AEC deems it necessary as part of its assigned duties, it may advise by way of the Prime Minister, and request reports and cooperation including the submission of materials, statements of viewpoint, and explanation from the heads of relevant administrative organizations.

The Minister of METI, before issuing a license to establish nuclear installations, shall receive views of the AEC with regard to the following items: (1) the nuclear installations will not be used for any purposes other than peaceful purposes, (2) the license will cause no hindrance to the planned development or utilization of nuclear energy, and (3) the applicant has an adequate financial basis to construct and maintain the nuclear installations.

The AEC is composed of a chairman and four other commissioners appointed by the Prime Minister with the consent of the Diet.

8.7 Other Administrative Bodies

Establishment of nuclear installations necessitates the compliance with other laws such as the Fire Protection Law and the Port Regulation Law. Therefore, the relevant safety regulations are conducted by the relevant government offices e.g. the Fire Protection Agency and the Ministry of Land, Infrastructure and Transport.

Table 8-1 Assigned Duties of the Divisions Related to Safety Regulation of Nuclear Installations (including nuclear fuel cycle facilities), NISA, METI

Policy Planning and Coordination Division	・ Planning and coordination the general policy of the NISA
Nuclear Safety Public Relations and Training Division	・ Activities for public hearing and public relations concerning the nuclear safety ・ Administration of the Nuclear Safety Inspectors and Senior Specialists for Nuclear Emergency Preparedness ・ Training and education of personnel to gain and to improve their competency
Nuclear Safety Regulatory Standard Division	・ Planning and coordination concerning technology and the system to ensure the nuclear safety ・ Regulation of nuclear power reactors in the stage of research and development ・ Research and development, etc.
Nuclear Safety Special Investigation Division	・ Management of allegation and litigation concerning nuclear safety
Nuclear Power Licensing Division	・ Regulation of commercial power reactors in the design and construction stage
Nuclear Power Inspection Division	・ Regulation of commercial power reactors in the operation stage
Nuclear Fuel Transport and Storage Regulation Division	・ Regulation of spent nuclear fuel storage business ・ Regulation concerning transportation of nuclear fuel ・ materials from sites
Nuclear Fuel Cycle Regulation Division	・ Regulation concerning businesses of refining, processing, ・ fabrication, spent-fuel storage, and reprocessing
Radioactive Waste Regulation Division	・ Regulation of radioactive waste business, dismantling and decommissioning of nuclear installations including nuclear fuel cycle facilities
Nuclear Emergency Preparedness Division	・ Planning of nuclear emergency preparedness ・ Prevention and investigation of incidents and accidents in nuclear businesses ・ Administration of activities in a nuclear emergency ・ Matters concerning physical protection
Electric Power Safety Division	・ Regulation of welding for electric structures ・ Environmental impact assessment

Table 8-2 Organization of the Nuclear and Industrial Safety Subcommittee, ACNRE

Basic Safety Policy Subcommittee	・ General matters securing safety
Nuclear Reactor Safety Subcommittee	・ Technical matters on commercial power reactors and power reactors at the stage of research and development
Nuclear Fuel Cycle Safety Subcommittee	・ Fabrication and reprocessing of nuclear fuel, storage of spent fuel, transportation of nuclear fuel material, and the technical standards
Decommissioning Safety Subcommittee	・ Decommissioning of nuclear installations*
Radioactive Wastes Safety Subcommittee	・ Securing safety of disposal and storage of radioactive wastes
Seismic and Structural Design Subcommittee	・ Technical matters on seismic safety and structural integrity of nuclear installations
Nuclear Emergency Preparedness Subcommittee	・ - Measures for incidents and failure, and general crisis management for emergencies of nuclear installations* and physical protection of nuclear material
INES Evaluation Subcommittee	・ INES Evaluation on incidents and accidents of nuclear installations*
Subcommittee for the Convention on Nuclear Safety	・ Matters related to the Convention on Nuclear Safety and ・ international standards on nuclear safety
Electrical Power Safety Subcommittee	・ Securing safety of electrical power
Study Group on the Way of Inspection	・ Matters concerning the inspection system of nuclear power generation facilities and nuclear fuel cycle facilities
Subcommittee for the Joint Convention on Radioactive Waste and Spent Fuel Safety	・ Matters related to the Convention on Joint Convention Radioactive Waste and Spent Fuel Safety
Subcommittee for the Institution of Nuclear Safety Regulation	・ Study of the legal system for the prevention of falsification of the self-controlled inspection record based on the investigation of the background of the falsification
Subcommittee for Fitness-for-Service Assessment etc. of nuclear power system	・ Study of the following, in the cases where a plant has cracks in a core shroud or reactor coolant re-circulation system piping: (1) Verification of validity in the check methods for core shroud etc. (2) Technical fitness-for-service assessment judgment method (3) Fitness-for-service verification etc. of individual plants based on check results specifically
Nuclear Safety Infrastructure Subcommittee	・ *Study of the current state and issues etc. of safety infrastructure (safety infrastructure study, codes and standards, human-resources infrastructure, study of facility infrastructure, knowledge base)*
Aging Countermeasure Examination Committee	・ Clarification of the requirements used as the rationale of measures for aging management, guidelines, etc. and study about the way of the reasonable safety inspection by the government

*: Including nuclear fuel cycle facilities

Table 8-3 Examination and advisory bodies under the NSC

Committee on Examination of Reactor Safety	• Matters concerning the safety of nuclear reactor facilities
Committee on Examination of Nuclear Fuel Safety	• Matters concerning the safety of nuclear fuel material
Emergency Technical Advisory Body	• Technical advice in emergency measures in case of occurrence of an accident or a failure that meet the given standard level in nuclear installation etc.
Emergency Technical Advisory Body for Disaster Prevention of Nuclear Carriers	• Technical advices in emergencies/ disasters of nuclear carriers
Emergency Technical Advisory Body for Nuclear Carrier Nuclear Disaster Prevention due to armed attack	• Technical advices in nuclear emergencies/ disasters due to armed attacks
Special Committee for Nuclear Safety Standards and Guides	• Matters concerning safety standards and guides of nuclear reactors, nuclear fuel facilities, and other nuclear installations
Special Committee on Radioactive Waste and Decommissioning	• Matters concerning safety assurance in radioactive waste disposal • Matters concerning the safety assurance in decommissioning nuclear installation
Special Committee on Safety Goals	• Establishment of safety goals
Special Committee on Radiation Protection	• Matters concerning the radiation protection considering domestic and international trends
Special Committee on Safety Transport of Radioactive Materials	• Matters concerning the safety assurance in transportation of radioactive materials considering domestic and international trends
Special Committee on Analysis and Evaluation of Nuclear Accidents and Failures	• Analysis and evaluation of domestic and international nuclear accidents and failures
Special Committee on Nuclear Safety Research	• Planning of nuclear safety research programs • Monitoring of the nuclear safety research programs • Evaluation of the nuclear safety research programs
Special Committee on Nuclear Disaster	• Emergency preparedness in the vicinity of nuclear installations, etc.
Task Force for introduction of Safety Regulations Using Risk Information	• Review and analyses of the issues in the introduction of safety regulation using risk information
Project Team on Safety Survey of Reprocessing Facilities	• Survey and analysis of matters relevant to the safety regulation activities during the test operation of the Rokkasho reprocessing facility
Safety Investigation on Disposal of Specialized Radioactive Wastes	• Technical matters concerning the safety assurance in the final disposal of high-level radioactive wastes
Investigation Project Team on Seismic Safety of Nuclear Facilities	• *The review of Results of Seismic Safety Re-evaluation of Existing Plants* • *Matters concerning the newest knowledge on seismic safety*

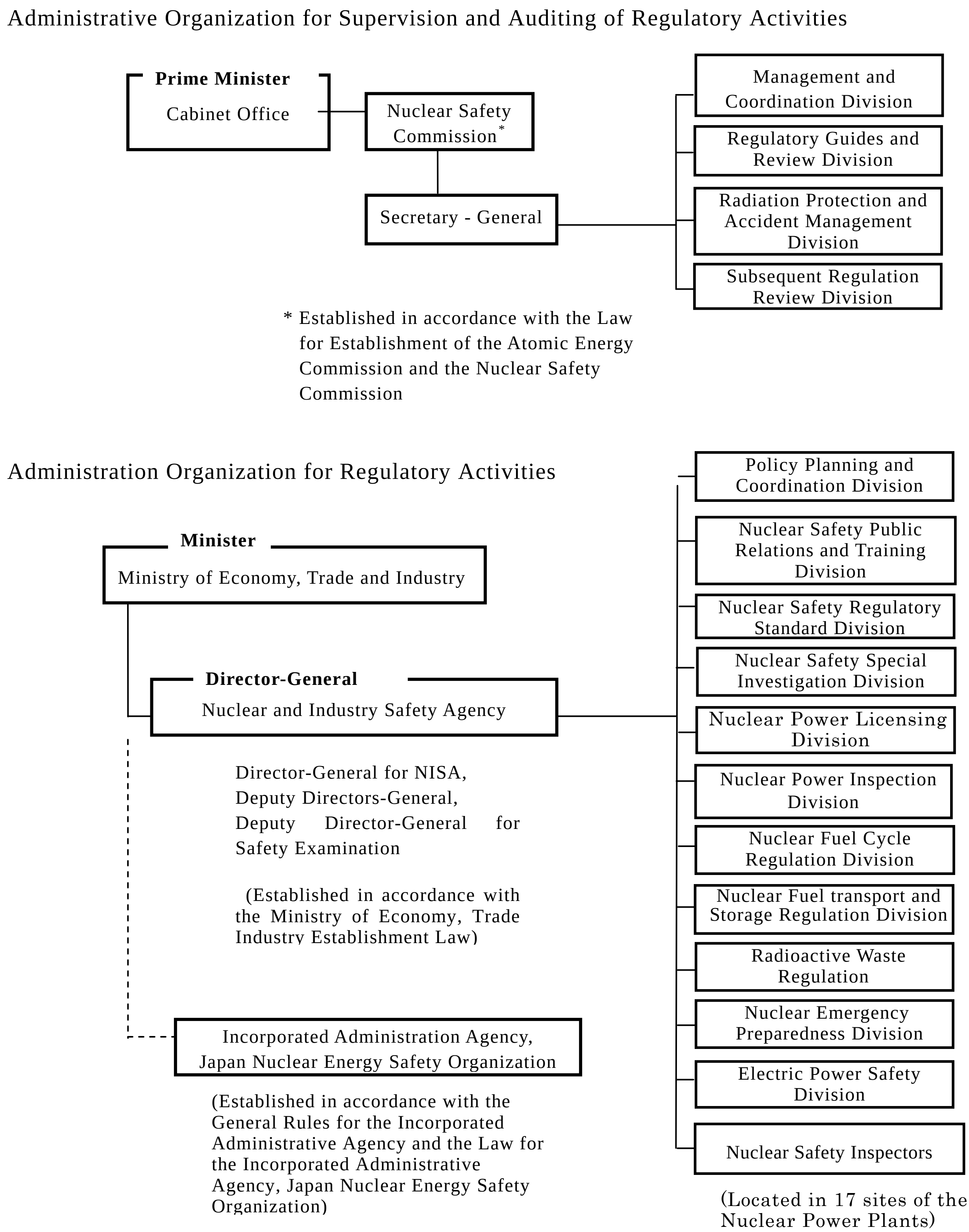

Fig. 8-1 Outline of the Safety Administrative Organization for Nuclear Installations (including the Nuclear Fuel Cycle)

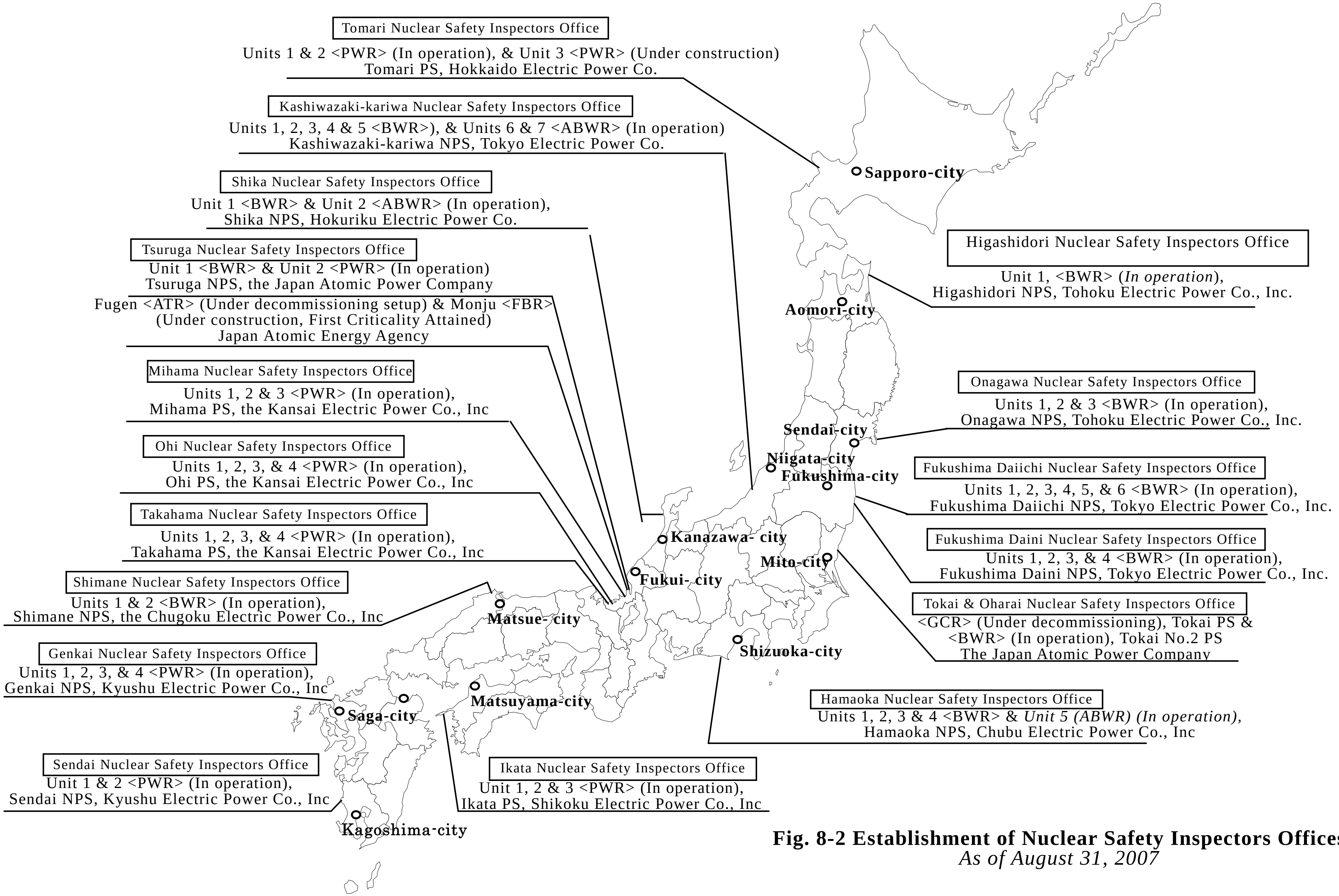

Fig. 8-2 Establishment of Nuclear Safety Inspectors Offices
As of August 31, 2007

Article 9 Responsibility of Licensee

Each Contracting Party shall ensure that prime responsibility for the safety of a nuclear installation rests with the holder of the relevant licence and shall take the appropriate steps to ensure that each such licensee meets its responsibility.

The prime responsibility for the safety of a nuclear installation rests with the licensee, while the regulatory body establishes relevant regulation to ensure the public safety and supervises that the licensee complies with the regulation.

The rules for the Reactor Regulation Law clarifies the licensee's responsibility on quality assurance and maintenance management, while the Electricity Utilities Industry Law clarifies the licensee's responsibility with the Periodic Licensee's Check and the Audit of Licensee's Periodic Check System

9.1 Regulatory Measures for the Licensee to take the Prime Responsibility

The prime responsibility for the safety of a nuclear installation rests with the licensee and the licensee shall comply with the regulatory requirements in each stage from planning through operation, which are stipulated in the Reactor Regulation Law, the Electricity Utilities Industry Law, etc. Those regulatory requirements are described in the Article 7 of this report.

The following activities of the licensee are described in the respective articles of this report;

- Education and Training of Operational Personnel etc. (Article 11),
- Quality Assurance Activities (Article 13)
- Periodic Safety Review (Article 14),
- Aging Management Review (Article 14),
- Emergency Preparedness (Article 16),
- Design and Construction (Article 18) and
- Operation (Article 19)

In addition, in order to ensure safety, not only meeting with these regulatory requirements, *the licensees are continuously to make effort for improving the safety and reliability of their nuclear installation by taking the following measures;*

- Education and training of operators and maintenance personnel and preparation of effective operation manual,
- Collection, examination and exchange of the information related to the operating experiences, ,
- Study of the state-of-the-art technical insight and implementation of safety research,
- Adoption of operating experience, etc. to design, operation and maintenance,
- Implementation of quality assurance activities and
- Preparation of the accident management, etc.

9.2 Supervision of Licensees by Regulatory Body

The basic mechanism to ensure the safety of nuclear installations is that NISA issues licenses, orders the licensee to bear the prime responsibility for safety and supervises it within the legislative and administrative framework.

The following is an overview of the above mentioned mechanism.

(1) Licensing

The Minister of METI issues a license for the establishment of a nuclear installation after examining that the nuclear installation will not be used except for the peaceful purposes, that there is no potential obstacle for accomplishing the planned development of atomic energy, that technical capability and financial foundations of licensees are sufficient, and that the site, the structure and the equipment of the nuclear installation may not cause any hindrance to the prevention of nuclear emergency. The regulation under the Reactor Regulation Law and the Electricity Utilities Industry Law in each stage from planning through operation is described in section 7.3.

(2) Periodic Licensee's Check and Audit of Licensee's Periodic Check System

In addition to the confirmation by NISA at the Periodic Inspection, licensee's self-controlled inspection that was carried out voluntarily by licensees has been upgraded to the mandatory "Periodic Licensee's Check" through the amendment of the Electricity Utilities Industry Law in 2003. Accordingly licensees shall inspect the nuclear installations subject to the Technical Standards and confirm the conformity and keep the records of the results.. The implementing system of the inspections above is audited by JNES as the "Audit of Licensee's Periodic Check System" and the audit results are reported to NISA. NISA evaluates the audit results and publicizes its conclusion.

(3) Operational Safety Inspection and Nuclear Safety Inspector

NISA conducts "Operational Safety Inspection" periodically based on the Reactor Regulation Law to confirm whether the compliance to the Operational Safety Program is assured. In accordance with the Reactor Regulation Law, NISA stations the Nuclear Safety Inspectors at each nuclear installation, who conducts the Operational Safety Inspection four times a year to confirm the licensee's compliance with the Operational Safety Program, and addresses incidents if they occur.

(4) Quality Assurance and Maintenance Management Activities

In accordance with the ordinance based on the Reactor Regulation Law, the licensee shall establish quality assurance system and maintenance management system and include them in the Operational Safety Program. NISA confirms the compliance with the Operational Safety

Program through the Operational Safety Inspection.

(5) The Senior Specialist for Nuclear Emergency

In accordance with the Special Law for Nuclear Emergency, NISA stations Senior Specialist for Nuclear Emergency at each site of nuclear installations, who guides and advises the licensee in preparing the Licensee's Plan for Emergency Preparedness, and conducts duties necessary to prevent nuclear emergency and mitigate the consequence should it occur.

(6) Periodic Safety Review

In accordance with the ministerial order based on the Reactor Regulation Law, licensees shall conduct the Periodic Safety Review for the operating reactor facilities every 10 years.

(7) Aging Management Review

In accordance with the ordinance based on the Reactor Regulation Law, licensees shall perform technical review on aging for the safety-related equipment and structures of nuclear installations and to establish and implement the Ten-Year Maintenance Program in no later than thirty years after the start of commercial operation. NISA reviews the technical evaluation and maintenance program prepared by the licensees.

(8) Accident Management

The licensee prepares an accident management program according to the "Accident Management of Severe Accidents at Power Generating Light Water Reactor Facilities", a decision by the NSC, 1992 (partly revised by the NSC in 1997), and submits it to NISA for review. NISA reviews and evaluates the technical adequacy of it.

(9) Reports on accidents and failures

In accordance with the Reactor Regulation Law or the Electricity Utilities Industry Law, the licensee shall report to NISA on accidents or failures.

(10) On-site Inspection

NISA conducts on-site inspection, if necessary, at the plants, offices, etc. of licensee or its contractor (welders) in accordance with the Reactor Regulation Law or the Electricity Utilities Industry Law

(11) Revocation

Judging that the licensee violates regulation, the Minister of METI may take measures of enforcement such as revocation of the license, suspension of operation, fine, etc., in accordance with the Reactor Regulation Law or the Electricity Utilities Industry Law.

In accordance with the provisions of Article 35 of the Reactor Regulation Law, licensees shall take necessary measures for 1) ensuring safe conditions of the nuclear facilities, 2) maintaining safe operation and 3) the safe transportation, storage and disposal of nuclear fuel materials or materials contaminated with nuclear fuel materials. When the above is violated, the Minister of Economy, Trade and Industry can order suspension of operation of the nuclear facilities. Moreover, in accordance with the provision of Article 39 of the "Electricity Utilities Industry Law", licensees are obliged to meet the technical standards as for electric facilities and when it is violated, the Minister of Economy, Trade and Industry can order the suspension of operation of the electric facilities.

9.3 Communication with Licensees

NISA, for mutually promoting the understanding between NISA and licensees on the policy of regulations, is trying to facilitate the opportunities to exchange opinions based on the transparency.

- *In order to build mutual trust with licensees and to make a smooth communication at inspection sites, the "Handbook for Inspectors", which describes the inspector's rules, is distributed to all of NISA inspectors to be carried always with them.*

- *Licensees, JNES and NISA forms the "Project Team for Operational Improvement of the Inspection System" and is making efforts to make efficient and steady use of the new inspection system.*

- *The Director General of NISA, presidents of licensees, etc. have opportunities to exchange opinions freely and openly on the safety situation of licensees' nuclear installation and their future tasks from the viewpoint of quality assurance. Moreover NISA staff visits the nuclear power station, and explains the current trend of safety regulation and the concept behind to the field operators and exchange opinions with them in order to promote the morale among the organizations and people concerned.*

- *In order to share nuclear related safety information concerning accidents, failures etc. of nuclear installations, "Regular Meeting on Safety Management of Nuclear Power Stations" is held about once every two months as an opportunity for opinion exchange on the safety management among NISA, JNES, electric utilities, the Federation of Electric Power Companies, and Japan Nuclear Technology Institute.*

C. General Safety Considerations

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Article 10 Priority to Safety

Each Contracting Party shall take the appropriate steps to ensure that all organizations engaged in activities directly related to nuclear facilities shall establish policies that give due priority to nuclear safety.

The development and utilization of nuclear energy has been promoted giving due priority to safety in accordance with the Atomic Energy Basic Law.

The JCO criticality accident in 1999, etc. showed the importance of moral and *education of the employee* in the organization, and resulted in the introduction of the Operational Safety Inspection system and the allegation system by the employee.

However, TEPCO falsification issue revealed in August, 2002, showed, again, that negligence of priority to safety among personnel gave rise to organizational falsification, and this led to the renovation of the safety regulation by NISA. *In the renovated regulation for organization and management, it has introduced the quality assurance system into the licensee and have established it in the regulation. The inappropriate quality assurance system and maintenance management activities were causes of the Secondary Pipe Rupture Accident at Unit 3 of the Mihama Power Station that occurred in August, 2004, and it was pointed out that the deterioration of the "safety culture" lurked in it's background. As the quality assurance and maintenance management activity are positioned in the center of the corporate culture and organizational climate concerning the nuclear safety (safety culture), the regulatory body comprehends licensees' efforts for preventing deterioration of the corporate culture and organizational climate at the periodic safety review, and is promoting their efforts by positively encouraging good practices etc. On the other hand, licensees etc. reported to NISA about the measures to prevent recurrence of about one hundred (100) cases of information falsifications and procedural deficiencies in April 2007, and aiming at ensuring nuclear safety, they strengthened their efforts for systematic improvement in the safety culture and started its enhancement. NISA, receiving the reports, is taking measures relevant to the priority to safety at the licensees.*

10.1 Basic Policy for Priority to Safety

Priority to safety is a basic policy in all nuclear energy development and utilization in Japan. Article 2 of the Atomic Energy Basic Law states that priority should be given to ensure safety in all related activities.

Also, Article 1 of the Nuclear Regulation Law states that "this law, in accordance with the Atomic Energy Basic Law, is enacted for the purposes of providing necessary regulations on the establishment and operation of reactors, in order to ensure that the use of atomic energy is limited to peaceful purposes and carried out in a planned manner, and at the same time, to ensure the public safety by preventing the hazards due to these materials and reactors."

10.2 Efforts for Improvement in Safety Culture

Safety culture in the organization is so vital to ensure the safety of the nuclear installation, that the lack of safety culture there may result in serious consequences.

Each regulatory body and licensee makes diverse efforts to establish safety culture.

(1) Efforts by Nuclear Industry

1) Policies of the whole nuclear industry

Japan Atomic Industrial Forum Inc., consisting of about *480* business operators (electricity utilities, reactor manufacturers etc.) who are engaged in the nuclear business, *established a "Charter for Safety by Nuclear Power Industry" in October 2006. The objectives of establishing the "Charter", and the main text of the "Charter" are described below. The top managements of all organizations are obligated to take necessary measures so that the "Charter" penetrates to the all fronts of each organization, and is practiced positively as a voluntary and continuous effort, and is aimed to the long-term continuation of safety achievement.*

Objectives to Establish the "Charter"

Technologies for peaceful use of nuclear energy in Japan is on a globally high level, but accidents and troubles which have occurred in the nuclear industry have affected the social confidence of the nuclear power industry. In order to be trusted by society with public confidence it is required of every person engaged in the nuclear industry to have a sense of pride and a sense of responsibility, to raise the consciousness of "not causing any accident by any means", and to establish the safety by taking action. The "Charter" is established as an action agenda accordingly.

Main Text of the "Charter"

Article 1
We have a sense of responsibility and an awareness of its duty, we give the priority to ensuring safety over all, no matter what it may be in what status,

Article 2
We aim at thoroughness of safety measures by learning modesty from past faults and sharing safety information.

Article 3
We make effort to develop good working environments, where matters perceived to be unsafe can be discussed at any time, are produced.

Article 4
We always keep a "questioning attitude", without being self-conceited with good safety

achievements.

Article 5
We positively release error information as well as we listen sincerely to the voice of society.

In addition, although the Charter is conducted by each member's independent effort, the Japan Atomic Industrial Forum Inc. has performed activities to promote the establishment of the Charter by the visiting local governments by the President, visiting members' offices to purport explanation, the presentation of each member's independent efforts at the member's liaison councils, etc.

On the other hand, in April 2005, the nuclear industry (including nuclear power operators, nuclear fuel fabrication facility operators, plant manufacturers, etc.) established the Japan Nuclear Technology Institute(JANTI), who inherits and enhances the functions of the network of organizations for sharing and in the improvement of the safety culture, "Nuclear Safety Network (NS Net)" and the Nuclear Information Center of Central Research Institute of Electric Power Industry and it has new additional functions to study and develop standards and codes, aiming at further improvement in the self-controlled operational safety activity of the nuclear industry. The activities concerning the improvement in the safety culture of this association is as follows;

a. Safety-culture dissemination activities

- *Holding seminars concerning safety and lecture meetings, opinion exchange meetings concerning safety for persons at licensees' sites. Moreover, the activities are opened to the public for transparency.*
- *Investigating and studying trends of safety culture in and outside Japan, and planning to support the licensees' self-controlled activities for safety culture.*

b. Peer-review activities

- *Cooperating with the Institute of Nuclear Power Operation (INPO) who has abundant peer-review achievements in the U.S. and the World Association of Nuclear Operators (WANO) who is developing the international peer review. It has been enhanced from the conventional review of just confirming documents to the review of focusing on field activities. Moreover, making effort to obtain good foreign practices and making an international contribution, by dispatching personnel to the peer reviews of WANO and the IAEA.*

- *Disclosing peer-review results, aiming at the formation of social consensus.*

c. Effective utilization of information

Utilizing the open library "NUCIA" on nuclear power generation and overseas information, JANTI analyses and evaluates the information and provides the results to *the licensees at the periodically held "Study Group on Operational Information" and at peer reviews. Licensees are supporting INPO, WANO etc. in exchange of operational experience information with overseas as well.*

d. Safety-culture assessment activities

Based on the efforts of fostering the safety-culture by the former NS Net, the status of the safety culture of member's sites will be, on a basis of questionnaires, assessed by JANTI as the third party, to support the self-controlled activities for fostering member's safety-culture. These assessment activities will start in the 2007 fiscal year as a trial and will be applied in a practical way in the 2008 fiscal year.

2) Policies of Licensees

All licensees *have declared their principles to give due priority to nuclear safety at nuclear installations, and have tried hard to improve not only in the safety culture but also the corporate ethics or quality assurance. Under the policy to give priority to safety, each licensee started system development so that the top management (president) participates in ensuring safety under his direct responsibility.*

But in July, 2006, NISA judged that the quality assurance system might not be functioning sufficiently at Tohoku Electric Power Co., Inc., and directed the company to perform the integrated check of the quality assurance system (refer to Section 13.2). Moreover, after discovering data falsification at a hydroelectric power plant of the Chugoku Electric Power Co., Inc. in October 2006, investigation at sections including thermal power and nuclear power was conducted, and of 316 cases in total, 98 cases of falsifications and procedural deficiencies at the nuclear power sections, were discovered. (Refer to Section 6.2)

The causes were judged by the companies that efforts till then by the electric power companies, who are licensees, were not pervading thoroughly to the job sites, and the support by top managements and managers were insufficient to lighten a burden of site stuff. After discussing about the prevention of recurrence at the "Reliability Recovery Committee" of the Federation of Electric Power Companies in March 2007, the electric power companies, reexamined the action agenda of the Federation, and in May of the same year, presented an action plan for prevention of recurrence to NISA, which includes participation of the top management, thoroughness of training and education of personnel, enhancement of sharing safety information, and this started the reconstruction and fixing of the safety culture.

(2) Effort of the National Government

Although the safety culture should be developed in organizations of licensees, who take the primary responsibility in the safety operation of nuclear installations, the national government has appropriate attention on licensees' fostering safety culture and promoting it.

1) Efforts of NISA

As it is important for a manager of an organization to pervade a sense of value by giving top priority to safety among site staff, regulatory bodies are required to look at the nature of the licensee's management and promotion of safety culture. The regulatory bodies are encouraging the licensee's managements so as to make them incorporate the quality management system and safety culture, for the time being, and they are requiring licensees to follow their quality management systems strictly. As part of this activity, NISA is designing institutional arrangements to clarify regulatory requirements concerning the quality assurance so that the licensee's quality assurance system is established firmly. The establishment of the quality assurance system is described in Article 13.

NISA together with JNES is, in order to promote the licensee's safety culture, taking the following measures;

- *Development of the quality assurance system*

  From October 2003, the licensee's quality assurance system is provided in the Operational Safety Program, and it is verified during the Operational Safety Inspection etc. that the quality assurance activities are functioning properly. Refer to the Article 13

- *Deterioration prevention of the corporate culture and organizational climate*

  Since the corporate culture and organization culture are fundamentals of various activities for ensuring safety, licensees are taking measures for deterioration prevention and the performing of the self-assessment at the periodic safety review. NISA perceives the licensees' efforts during the periodic safety review and is promoting their efforts by positively encouraging good practices as follows.

  a) Viewpoints

  NISA perceives the licensee's efforts from the following viewpoints.

  (i) Effectiveness of the efforts for detection and prevention of organization climate deterioration
  (ii) Self or external assessment on the effectiveness of the efforts for detection and prevention of organization climate deterioration

  b) Encouragement

  (i) Especially effective efforts for detection and prevention of organization climate deterioration
  (ii) Especially effective efforts to enrich measures for aging management in the organizational climate. The viewpoints to be used to perceive efforts developed

by JNES are provided in Table 10-1.

- *Evaluation of safety culture*

  NISA together with JNES is developing a guideline to assess the licensee's safety culture at the Operational Safety Inspection. In the development of this guideline, IAEA's publication (INSAG-4 "Safety Culture", "ASCOT Guideline", etc.), ISO 9001 (2000) and foreign examples are referred to.
  Note: ASCOT: Assessment of Safety Culture in Organizations Team

- *Root cause analysis of accidents and failures*

  When an accident or a failure occurs, it is necessary to clarify not only the direct causes, but also the root causes taking account of their organizational factors. Licensees are required to implement systematic and lasting measures. From the viewpoint of making licensee's root cause analyses effectual, a guideline for root cause analysis is being prepared in the system of the quality assurance standards of Japan Electric Association. NISA together with JNES is developing a guideline to assess effectiveness of the licensee's root cause analysis.

- *Comprehensive check of power generation facilities*

  Based on the comprehensive check (refer to Section 6.2) of power generation facilities, NISA decided to take measures for licensees to give due priority to safety furthermore. These matters will be reflected in related legislations, etc.

  (i) Clarification of a compliance system in the Operational Safety Program

  The following will be added as matters to be provided in the Operational Safety Program.
  - *Matters relating to the compliance system,*
  - *Matters relating to the system for fostering safety culture,*
  - *Matters relating to the root cause analysis, and*
  - *Matters relating to the information reporting important-to-safety.*

  (ii) Addition of measures for Safe Operation

  - *To develop work procedures properly and to perform operational safety activities following the procedures, and*
  - *To take actions for the required procurement control in order to be able to share manufacturer's information about safety technologies among licensees.*

  (iii) Development of an organizational system in which independency of the Chief Engineer of Reactors is secured

2) Efforts of the NSC

- *First-Series Roundtable Discussions on Safety Culture*

The NSC, as one of measures taken after the JCO nuclear criticality accident occurred in September 1999, held the " First-Series Roundtable Discussions on Safety Culture " with unit managers and shift supervisors of twenty one (21) nuclear facilities in Japan from July 2001 to December 2003. The contents were compiled and published in a document "Site interviews about Safety Culture –Discussions on sites where the safety should be assured -" (January 2004). The summary is as follows;

(a) Findings brought from the sites for developing safety culture

a. Educate the staff to have their own pride and responsibility without hiding in an organization.

b. Educate the staff to develop common sense and morals so that they can recognize the necessity of safety by their own values.

c. Mistakes are not for learning about people's faults. Mistakes are lessons for preventing problems from occurring again and for assuring safety. It is necessary to recommend staff members to report mistakes and managers should not punish them for it.

d. Learning does not always consist of sitting and learning. There is actual experience and learning by simulation.

e. You can find opportunities everywhere on-site for systematic training, lessons for preventing problems from reoccurring, and for other things.

f. Successful experiences do expire. Do not adopt everything at once. Be sure to confirm and consider them since some successful experiences may be out of dat.

g. Introduce knowledge in quality control and other areas to business management.

h. Be sure to cope with all circumstances, imagine every possible situation and make it clear who is responsible for making the decisions.

i. There is no purpose in collecting information alone. It is more important who collects it and how it is used.

j. Learn to see yourself and your organization from a broad point of view.

(b) Important topics brought from the sites regarding safety culture

a. Educate every staff member to comply with the corporate morals.

b. Use adverse circumstances and pressures (You can keep tension by the strict gaze of the surrounding society).

c. Communicate more with the media than usual and proceed with transparency of information.

d. Educate every staff member to be accustomed to being conscious of the complete *flow as a whole.*

e. Let them realize that reporting and consulting are the best steps for assuring safety.

f. Introduce engineer morals and morals for work accomplishment into education and training programs.

g. Educate all staff members to have a questioning attitude about the safety of their work, actions, and activities.

h. Make the most use of knowledge in the general analysis system that supports the passing on technology and the high technologies that explain vagueness.

i. Always appreciate the appropriate allocation of human resources. (Top management)

j. Reinforce the quality assurance system and concentrate on software quality assurance.

k. Maintain and improve the ability of the staff in accordance with the increase rate of direct constructions.

l. Sharing the values of safety with cooperating companies is a major key for developing safety culture.

m. Use the latest knowledge about risk communication to communicate with outside stake-holders.

n. It is necessary to observe our actions both from the subjective and objective viewpoints and to ask ourselves if they are appropriate.

- *Second-Series Roundtable Discussions on Safety Culture*

The NSC held a series of "Roundtable Discussions on Safety Culture", from October 2004 to April 2005, as one of the measures after the secondary pipe rupture accident at Unit 3 of the Mihama Power Station that occurred in August 2004, in order to exchange opinions with top managers of nuclear power companies and major contractors. The contents were compiled and published in a document "Fostering a Culture of Safety in Japan's Nuclear Industry – Exchange of Views with Top Management" (June 2005), The summary is as follows.

(a) Management Safety Alertness and Activities

To conduct activities of nuclear energy, everybody in the organization must share and implement the practices of valuing the concept of "safety first", by constantly questioning whether the current practices of activities are appropriate from the viewpoint of ensuring safety. To this end, the top management should take leadership in such areas as organizational composition, resource allocation, quality assurance system, technology, human resources and training.

(b) Productive Communications between Site Staff and Upper Management

To give safety-ensuring activities substantial significance, it is crucial that safety-related information smoothly permeates throughout the organization, and that a system and means be provided to enable it. With full awareness of the difficulties involved in creating smooth communications, management must constantly and intentionally motivate their staff to improve the situation by ensuring the bi-directional information channels, and appropriate and timely remedial actions.

(c) Workplace Environment

Regulators and licensees must continue to make bilateral efforts to improve the effectiveness of regulatory activities for productively improving safety assurance, without being content with the formalities in meeting regulatory standards in effect. It is necessary that the management of licensees and contractors will maintain and promote a thoroughgoing cooperative relationship, while sharing a strong perception that ensuring safety is the prerequisite to everything else in nuclear activities, and that it be the most efficient means of cost optimization.

Table 10-1 Viewpoints to Comprehend the Efforts to Prevent Deterioration of Licensees' Organizational Climate

Points to confirm symptoms of the organizational climate deterioration	***Viewpoints of Comprehension***
1. Top management's commitment	*Pervasion of messages for priority to safety to the tips of organization.*
2. Senior management's clear policies and behavior	*Presentation and behavior of the policies of ensuring safety*
3. Improvement and fixing of the quality management system (QMS)	*Feedback of the knowledge obtained from the nonconformity management to QMS improvement*
4. Reporting culture	*System, encouragement and utilization of reporting*
5. Learning organization	*Operating experience feedback, efforts of maintaining and improving technical capabilities, communication in the operational safety activities*
6. Workplace with good communication	*Efforts to improve in-company communication, communication with contractors*
7. Exclusion of decision-making by a misjudgment	*Preventive measures for eliminating a decision-making by a misjudgment*
8. Compliance with rules	*Maintenance and management of rules, fixing of routine work*
9. Accountability and transparency	*Timely information service, promotion of mutual understanding among local residents and regulatory bodies, improvement in transparency*
10. Self (or 3rd party) assessment	*The self-assessment (the 3rd party) method for preventing activities to be just a formality*

Article 11 Financial and Human Resources

1. Each Contracting Party shall take the appropriate steps to ensure that adequate financial resources are available to support the safety of each nuclear facility throughout its life.
2. Each Contracting Party shall take the appropriate steps to ensure that sufficient numbers of qualified staff with appropriate education, training and retraining are available for all safety-related activities in or for each nuclear facility, throughout its life.

The financial basis of electricity utility rests on the understanding and recognition that nuclear energy is the environmentally clean energy and reliable source for base load power, under the pressure of reduction of power rate due to the deregulation of electricity utility industry.

In Japan the sufficient numbers of qualified staff are available through imposing requirements of appointment of chief reactor engineers, persons responsible for operation, chief electrical engineers, etc. upon licensees *and assessing the technical competence of the licensees. After more than 30 years of operational experiences, many experienced employees are now retiring. The licensees* face the challenge of succession of technology and ensuring human resources. The recruitment and training of personnel in various fields are now in progress.

11.1 Financial Resources of the Licensee

(1) Confirmation at Issuing License

Before issuing license of a nuclear installation, the Minister of METI, in accordance with Article 24 (Criteria for the license) of the Reactor Regulation Law, confirms that the applicant for the license possesses necessary financial basis by requiring the applicant to submit "Amount of Funds Required for Construction and Finance Procurement Plan", and also consults with the AEC. (Refer to *section 7.3* and Fig. 7-3)

(2) Applicant for the License of Nuclear Installations

Applicants for license of commercial power reactors are the general electric utilities, that is, 9 electric power companies and 2 wholesale electric power companies. The Minister of METI issues license for *electricity utility business* only to those meeting certain criteria of financial basis, technical capability, etc.

The nuclear power generation is recognized as a superior power source with characteristics of reliable supply and useful energy to cope with global warming, and measures, such as performing a preferential load dispatching, are taken. Such utilization of the nuclear energy ensures stable flow of income for electric utilities.

On the one hand, since back end businesses, such as reprocessing of spent fuel generated by nuclear power generation, requires large amount of expense over extremely long period of time, in accordance with the "Law for Reserving and Management of Reserve Funds for Reprocessing Spent Fuel in Nuclear Power Generation, etc." enacted in 2005, the electric utilities reserve funds for the expense in advance. As for final disposal of high level radioactive wastes, the Nuclear Waste Management Organization of Japan, an implementing organization for disposal, will perform geological disposal in accordance with the Law concerning Final Disposal of Specific Radioactive Waste enacted in 2000, using funds reserved by electric utilities etc. Furthermore, METI enacted the Ministerial order of Reserve Fund for Dismantling Nuclear Power Facilities in accordance with provision of Article 35 of the Electricity Utilities Industry Law, and the electric utilities deposit reserves for decommissioning on the basis of this order.

On the other hand, the Japan Atomic Energy Agency, who owns R&D reactors of Monju and Fugen, is established by a law, and financial basis necessary for its business operation is provided by the national budget.

11.2 Human Resources *concerning the Nuclear Installation of the Licensee*

(1) Confirmation of Technical Competence

Before issuing license of a nuclear installation, the Minister of METI confirms that the applicant possesses technical competence necessary to establish a nuclear installation and operate it adequately, and consults with the Nuclear Safety Commission (the NSC). The NSC had established the “Regulatory Gude for Reviewing Technical Competence of Nuclear Operators” on May, 2004, *and based on this examination guide, the applicant's technical competence for the following items is examined.*

The examination items of technical competence;

1. *Organization for design and construction,*
2. *Ensuring engineers for design and construction,*
3. *Experience related to design and construction,*
4. *Quality assurance activities concerning design and construction,*
5. *Organization for operation and maintenance,*
6. *Ensuring engineers for operation and maintenance,*
7. *Experience related to operation and maintenance,*
8. *Quality assurance activities concerning operation and maintenance,*
9. *Education and training for engineers,*
10. *Designation and staffing of qualified personnel etc.*

The licensees are responsible for safety of the decommissioning and for preparing personnel for it. *In fact, the licensees have trained and prepared human resources and implemented technological development programs through the decommissioning and verification test on a research reactor (JPDR) in cooperation with national organizations, manufacturers and construction companies, and implemented decommissioning for the Tokai Power Station of the*

Japan Atomic Power Company.

(2) Qualification, Training and Retraining of Personnel Engaged in Safety Activities

1) Staff Qualification

The licensees shall appoint a Chief Reactor Engineer to supervise safety operation of nuclear installation, a Chief Electrical Engineer and a Chief Engineer of Boiler and Turbine to supervise safety during construction, operation and maintenance of electric facilities. The licensee assigns the Persons Responsible for Operation from those who have knowledge, skills, and experience required for operation of reactors, and who satisfy requirements provided by the Minister of METI. *As a method to judge conformity to requirements of a person responsible for operation, it is under study to incorporate an accreditation system utilizing an independent organization. The person responsible for operation observes operation in general and supervises operators. The numbers of Chief Engineers of Reactors and Persons Responsible for Operation are 1254 and 421 respectively at the end of June, 2007.*

Their duties are explained in Section 19.3 in detail.

As for the qualification of staff engaged in the *dimensional measurement* of defect in the fitness-for-service assessment of equipment, *PD (performance demonstration) qualification system for the ultrasonic flaw detection system (including the qualification of staff) is employed by the Japanese Society for Non-Destructive Inspection, and 14 PD technicians are certified (as of November 2006).*

2) Staff Training and Retraining and Resources for Training

Licensees shall integrate education on operational safety of personnel in charge of operation and management of a nuclear installation into the Operational Safety Program, and prepare and carry out long-term and short-term staff training programs to maintain and improve their skills and capabilities. Licensees, in addition to in-house operator training course using simulators (Table 11-1); periodically send their operators to external operation training centers for retraining. There are two centers: the BWR Operation Training Center (BTC) for BWRs and the Nuclear Power Training Center (NTC) for PWRs. A curriculum suitable for the ability/skill of each operator is prepared in these training centers.

Each licensee has established maintenance training centers (Table 11-2) for education and training of maintenance personnel. Various mock-up devices, inspection devices and training devices, etc, simulating plant facilities for training purposes, have been used to maintain and improve the knowledge, skills and work management capabilities of personnel involved in maintenance and inspection.

11.3 Efforts for Ensuring Infrastructure of Human Resources in the Regulatory Bodies

(1) Training of Experts in NISA

Staff members, who are in charge of nuclear regulation in the Nuclear and Industrial Safety Agency (NISA), are the Senior Specialist for Nuclear Emergency, the Nuclear Safety Inspector, the Nuclear Facility Inspector, the Electric Facilities Inspector, and the Safety Examiner. These are called "Nuclear Regulatory Staff" as shown below.

A Senior Specialist for Nuclear Emergency is stationed at each nuclear installation, guides and advises the licensees in preparing its Plan for Emergency Preparedness, and conducts duties necessary to prevent progression of nuclear emergency should it occur.

A Nuclear Safety Inspector is stationed at each nuclear installation, conducts the Nuclear Safety Inspection to confirm licensee’s compliance with the Operational Safety Program, address incidents if they occur, and supervises operation management of a nuclear installation.

A Nuclear Facility Inspector and /or an Electric Facilities Inspector is dispatched from NISA head office, and conducts inspection activities, such as the Pre-Service Inspection and the Periodic Inspection of a nuclear installation, and the Fuel Assembly Inspection, on the basis of the Reactor Regulation Law or the Electricity Utilities Industry Law, respectively.

Safety Examiners conduct the Safety Examination of a nuclear installation.

A Nuclear Regulatory Staff is required to have expertise in nuclear technology. The system of long term and multistage education and training programs necessary for improvement of his/her expertise is developed, taking account of his/her experience and of the nature of the facility to which he/she is assigned. Moreover, NISA started a Special Training Course on Quality Assurance of Nuclear Installation in 2002.

In order to increase effectiveness of the training, the contents of the training being implemented are also reviewed and improved suitably. The disposition of the personnel engaged in securing safety of nuclear installations is improved through these training. Summary of training for nuclear safety regulation is shown in Fig. 11-1.

NISA has appointed six Special Inspection Instructors in December 2003. They advise inspectors for the Nuclear Safety Inspection, the Periodic Inspection, etc. in each power station, instruct them to equalize the levels of inspections, and collect opinions and proposals from inspectors and *licensees for the purpose of opinion exchanges at the site.*

Furthermore, NISA maintains and develops its regulatory capability, as well as contributes to international safety regulation, through exchange of technical experts and information on safety regulation and safety technology, under bilateral arrangements with foreign regulatory bodies and in the framework of multilateral cooperation (the IAEA and the OECD/NEA).

Moreover, besides training professional human resources as mentioned above, NISA recruits *professional human resources for nuclear safety from industries or other ministries and government offices, and is making efforts to ensure infrastructure of human resources.*

In the future, the training curriculum that makes staff be able to acquire the required regulatory skills effectively will be established, and a training management system that keeps and manages every personnel's training status and offers appropriate training at suitable time will be developed. Moreover, measures, such as preparation of training facilities for training of practical inspection skills etc. through with mock-up will be taken.

(2) Training of Experts in JNES

Japan Nuclear Energy Safety Organization (JNES), as well as NISA, develops training courses for its personnel, putting emphasis on inspection activities.

JNES's inspection activities include the Electric Facilities Inspection, the Nuclear Facility Inspection, the Welding Inspection, the Audit of Licensee's Periodic Check System, the Audit of Licensee's Welding Check System, the Safety Confirmation of Disposal Facility, the Safety Confirmation of Radioactive Waste Package, the Confirmation of Transportation Packaging, and the Confirmation of Transportation Method. The Electricity Utilities Industry Law or the Reactor Regulation Law stipulates that each of these activities be conducted by qualified personnel. JNES prepares various training courses for staff members to get appropriate qualification in their respective activities. President of JNES assigns inspectors from those qualified persons.

JNES encourages inspectors and examiners to be qualified in the disciplines related to their duties. Moreover, JNES encourages inspectors and examiners to participate in school of external bodies, scientific seminars etc. to enhance their expertise.

11.4 Maintaining Human Resources in Nuclear Fields in Japan

In order to ensure safety of nuclear power generation, highly capable human resources should be maintained and ensured. In Japan, it has been an issue to keep human resources in appropriate condition, because of low birthrate and increase of elderly people, decrease in population, retirement of skillful engineers, and decrease of construction opportunities of nuclear installation.

(1) Efforts by National Government

NISA has been studied strategic measures for training and ensuring human resources of nuclear specialists at the Nuclear Safety Infrastructure Subcommittee under the Nuclear and Industrial Safety Subcommittee established in 2006. Based on the availability of external advisory specialists and the emphasis on ensuring safety in nuclear facilities and on appropriate safety regulation, the subcommittee studies the strategy for training and ensuring of human resources with the understanding that it is necessary to clarify the technical fields (specific and basic) in which human resources should be allocated.

In parallel with this study, the Ministry of Education, Culture, Sports, Science and Technology and the Ministry of Economy, Trade and Industry will implement the nuclear human resource training programs focusing on the following items from 2007 fiscal year;

i) Support of educational activities, such as basic nuclear education and research, internship, and preparation of core curriculum for the nuclear power,

ii) Implementation of nuclear human resource training programs for research activities focusing on basic and infrastructure technology fields supporting the nuclear power, for training of human resources to ensure research successors.

(2) Efforts by the Nuclear Industry

The nuclear industry has grave concerns in the succession of technology, expertise and experiences between the generations. The first generation experts who had many experiences in commissioning test, operation, maintenance, and trouble shooting in abnormal events are in the age of retirement.

Main activities currently performed for human resource training and technology succession in the nuclear industry are shown in the following;

1) Training of on-site technicians and succession of skills

In the area where the nuclear installation is established, the training for qualification, training of practical skills for maintenance and repair, OJT training at the power station, etc. are implemented beyond the frame of an individual firm.

2) Study on qualification and certification system for private sectors

For the purposes of improved skills of maintenance-and-repair workers, appropriate staffing, and ensuring future human-resources, the common standards and qualification / certification procedures for objective evaluation of skill level are being studied. These standards etc. will be made to harmonize with the licensee's in-house qualification system.

3) Acquisition of advanced expertise

Licensees' engineers acquire the education in the graduate school related to the nuclear power, thus engineering specialists with advanced expertise are fostered.

Moreover, the Japan Atomic Industrial Forum, Inc., consisting of enterprises related to the nuclear power has proposed policies as a private sector, in order to improve the effectiveness of "the nuclear human resource training program. It has investigated and compiled the current state, issues of schools, such as universities, graduate schools and research organizations, for improvement for human resource training. The study is continued.

(3) Efforts by Universities and Research Institutes

Recent years, it has been a trend to reorganize and to unify faculties or graduate schools. In this trend, conventional nuclear department and majoring were unified with other ones, and their names were changed. And they were reorganized to departments and majoring that address wider fields, including energy, environment etc. However, in recognition of an importance of nuclear education, for the purpose of training for engineering specialists with practical capabilities and engineering theories in the nuclear field, the faculty, graduate school and professional school have been established in nuclear research and development complex, such as Fukui Prefecture and Ibaraki Prefecture.

Moreover, some research organizations and graduate schools incorporate a cooperation system of graduate schools. Through the system, the facilities/equipment and human resources in the research organizations are shared, the contents of education/study at the graduate school become more in details, the communication between researchers is promoted, and the education at graduate schools is activated. The Atomic Energy Society of Japan has founded a senior network (SNW) whose members are retired seniors from nuclear related organizations. Senior people plays a central role in dialogues with college students and in other activities at SNW to make succession of nuclear technologies to young people and students who are responsible for next generation and to spread correct understanding of the nuclear power.

(4) Establishment of Professional Engineers System for Nuclear and Radiation Technologies

The Ministry of Education, Culture, Sports, Science and Technology who has jurisdiction over the Professional Engineers System established a nuclear and radiation technology department for professional engineers in *2004 fiscal year*. *The qualification examination has been implemented every year, and a total of 153 people were qualified as the Professional Engineer by the end of 2006 fiscal year. The purposes of the Professional Engineers System are such to enhance nuclear engineering capabilities, to utilize the capability in the nuclear safety regulation, to strengthen the safety management system in each corporation.*

Table 11-1　Operator Training Facilities of Nuclear Installations

Organization	Location	Simulator
BWR Operator Training Center Corp.	Okuma-machi, Futaba-gun, Fukushima Prefecture Kariwa Village, Kariwa-gun, Niigata Prefecture	Full scale; 3 units Full scale; 2 units
Nuclear Power Training Center Ltd	Tsuruga, Fukui Prefecture	Full scale; 3 units
The Japan Atomic Power Co.	The Japan Atomic Power Company Training Center (Tokai Village) On site of Tsuruga Power Station	Compact; 1 unit Compact; 2 units
Hokkaido Electric Power Co., Inc.	On site of Tomari Power Station	Full scale; 1 *unit*
Tohoku Electric Power Co., Inc.	Nuclear Power Engineering Training Center (on site of Onagawa Nuclear Power Station) Nuclear Power Engineering Training *Building*(on site of Higashidori Nuclear Power Station)	Full scale; 1 unit Full scale; 1 unit
Tokyo Electric Power Co., Inc.	On site of Fukushima Daiichi Nuclear Power Station On site of Fukushima Daini Nuclear Power Station On site of Kashiwazaki Kariwa Nuclear Power Station	Full scale; 1 unit Full scale; 1 unit Full scale; 1 unit
Chubu Electric Power Co., Inc.	Nuclear Power Training Center (on site of Hamaoka Nuclear Power Station)	Full scale; 2 units
Hokuriku Electric Power Co.	Nuclear Power Engineering Training Center (on site of Shika Nuclear Power Station)	Full scale; 1 unit
The Kansai Electric Power Co., Inc.	On site of Mihama Power Station On site of Takahama Power Station On site of Ohi Power Station	Compact; 1 unit Compact; 1 unit Compact; 1 unit
The Chugoku Electric Power Co., Inc.	Ohno Training Center (Ohno-machi)	Full scale; 1 unit
Shikoku Electric Power Co., Inc.	Nuclear Engineering Training Center (Matsuyama) On site of Ikata Power Station	Full scale; 1 unit
Kyushu Electric Power Co., Inc.	Nuclear Power Training Center (on site of Genkai Nuclear Power Station) Nuclear Power Training Center (on site of Sendai Nuclear Power Station)	Full scale; *2* unit Full scale; 1 unit
Japan Atomic Energy Agency	On site of Fugen Power station On site of Monju Construction Office	Compact; 1 unit Full scale; 1 unit

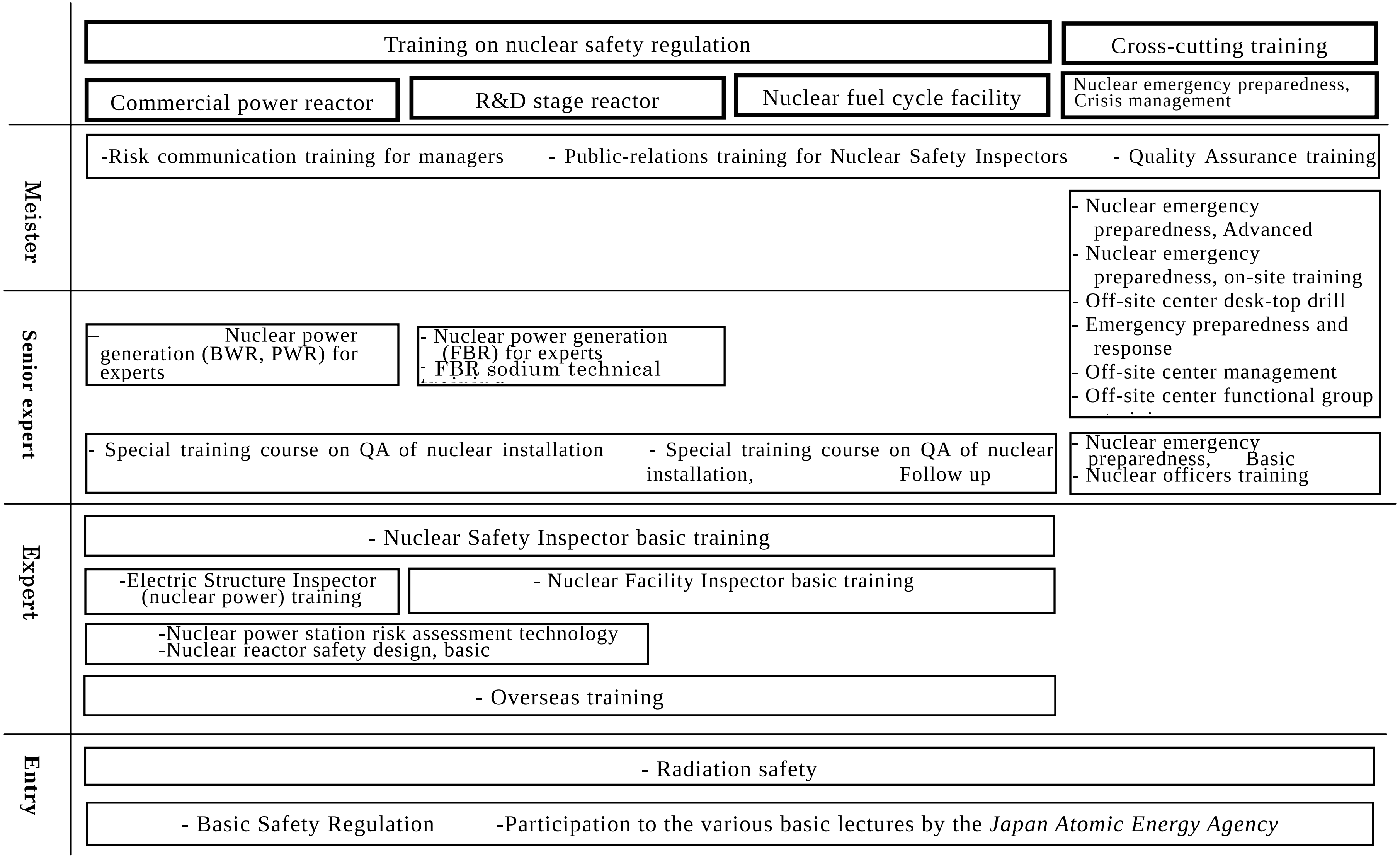
(As of the end of March, 2007)

Table 11-2 Maintenance and Repair Training Centers of Licensees

Organization	Name	Location
The Japan Atomic Power Co.	The Japan Atomic Power Company Training Center	Tokai Village, Naka-gun, Ibaraki Prefecture
Hokkaido Electric Power Co., Inc.	Nuclear Power Training Center	On site of Tomari Power Station
Tohoku Electric Power Co., Inc.	Nuclear Power Engineering Training Center	On site of Onagawa Nuclear Power Station
Tokyo Electric Power Co., Inc.	Fukushima Nuclear Power Plant Training Center	On site of Fukushima Daiichi Nuclear Power Station
	Kashiwazaki Kariwa Nuclear Power plant Training Center	On site of Kashiwazaki Kariwa Nuclear Power Station
Chubu Electric Power Co., Inc.	Nuclear Power Training Center	On site of Hamaoka Nuclear Power Station
Hokuriku Electric Power Co., Inc.	Nuclear Power Engineering Training Center	On site of Shika Nuclear Power Station
The Kansai Electric Power Co., Inc.	Nuclear Power Maintenance Training Center	Takahama-cho, Ohi-gun, Fukui Prefecture
The Chugoku Electric Power Co., Inc.	Shimane Nuclear Power Station Engineering Training Center	On site of Shimane Nuclear Power Station
Shikoku Electric Power Co., Inc.	Nuclear Engineering Training Center	Matsuyama City, Ehime Prefecture
Kyushu Electric Power Co., Inc.	Genkai Nuclear Power Station Nuclear Power Training Center	On site of Genkai Nuclear Power Station
	Sendai Nuclear Power Station Nuclear Power Training Center	On site of Sendai Nuclear Power Station
Japan Atomic Energy Agency	General Training Facility for FBR Cycle	On site of International Nuclear Information/Training center

(As of the end of March, 2007)

Fig. 11-1 Training on Nuclear-Safety Regulation



Article 12 Human Factors

Each Contracting Party shall take the appropriate steps to ensure that the capabilities and limitations of human performance are taken into account throughout the life of a nuclear installation.

The licensee takes human factors into consideration at the design stage of nuclear installation, and, at the operation stage, prepares operational procedures, education and training course for its personnel and the management system for operation and maintenance. The regulatory body also takes various steps for prevention and correction of human errors at design and operation stage.

Since the last report, the requirements for prevention of misoperation in the main control room are clarified and came to be reflected in the facility design.

12.1 Efforts by Regulatory Body

(1) Design Stage

1) The Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities requires that "the nuclear installation be designed to reflect appropriate preventive considerations against operators' misoperation", and its explanatory document requires that "In designing, attention should be given in consideration of ergonomics-oriented factors, to panel layout, operability of operating devices, valves, etc., instrument and alarm indication for accurate and quick recognition of reactor status and prevention of errors during maintenance and inspection." and that "in designing, measures should be taken so that necessary safety function is maintained without operator's actions for a certain length after the occurrence of an abnormal condition." The guide also requires that "control room be designed that the situation of operations and principal parameters of reactor and principal related facilities can be monitored and that prompt manual control can be performed, whenever required, to maintain safety." In conformity to these requirements, the Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities requires that "safety analysis be performed in consideration of the following: in case that operator actions are expected at the occurrence of abnormal situations, sufficient time and adequate information be available so that operator may be able to properly judge the situations and take necessary acts with a high degree of confidence." JNES prepared a manual for evaluation of human factors in the main control room, to confirm that these requirements are reflected in the design.

2) At the Approval of Construction Plan, the Technical Standards as *a performance code* under the Electric Utilities Industry Law request that the main equipment necessary for safe operation of nuclear installation can be monitored at a glance and necessary actions can be taken in the control room *without any misopration*.

3) The Nuclear and Industrial Safety Agency (NISA) and the Japan Nuclear Energy Safety Organization (JNES) clarified the requirements for prevention of misopration, so that the *above mentioned performance code is reflected on the specification code developed as an academic society and association standard. These requirements are shown on table 12-1.*

(2) Operation Stage

1) The Reactor Regulation Law provides that the licensee prepare the Operational Safety Program, obtain approval of NISA on it and comply with it. The program includes preparation of operation management system, education on operational safety, operational procedures etc. NISA confirms and approves the Operational Safety Program, and the resident Nuclear Safety Inspectors confirm the compliance with it by the licensee in the Nuclear Safety Inspection.

2) The licensee reports failures of the installation to NISA in accordance with the laws. Especially, in the case that the failures are identified to be caused by human errors, the licensee reports to NISA measures addressing failures including improvement of equipment. *Licensees are required to correct any nonconformance of direct cause, such as a human error, and NISA is preparing guidelines to evaluate licensees' self-supporting efforts for corrective actions and to identify viewpoints to promote them in cooperation with JNES.*

NISA also consults on the failure with experts when necessary and urges licensees to apply lessons learned to other installations. JNES analyses human error-related cases in detail, and selects items to be reflected in the safety regulations. JNES prepares a summary report on lessons learned, and accumulates them in the database.

Table 12-1 Requirements to prevent misoperation in a main control room

Item	*Requirements*
1. *Environmental conditions of a main control room*	*Main control room should be in comfortable environmental conditions taking into consideration temperature, lighting and noise so that operators can operate appropriately.*
2. *Arrangement and working space of a main control room*	*(1) Consider that the following does not become too much burden at an operator in any plant operating conditions.* (a) *Duty allocation of human and machines shall be decided.* (b) *Items that should be intensively supervised and operated in a main control room shall be defined, and the duty allocation with local spots (including the panels installed at the back of main control panels) shall be decided.* (c) *The equipment arrangement shall be designed so that information sharing among operators to be effective.* *(2) When manual operation is required for safety in a plant abnormal condition, the operators' working area shall be limited to the area possible for surveillance and operation.*
3. Arrangement of devices on control panels	*Alarm, display and control devices which are installed on control panels shall be arranged with unified displaying manner so that an operator's misoperation or erroneous recognition can be prevented.*
4. Display system (including alarm system)	*(1) Information function* *(a) Information that is used for condition display of plant system and equipment that is necessary for ensuring safety shall be sufficient and be provided to operators being easy to understand at suitable position.* *(b) It should be considered that communication defect or judgment error should not arise, which is an important function of communication and cooperation with on site emergency station.* *(c) Safety significant information should be displayed at the position where operators in a control room can share among them such information.* *(2) Alarm function* *When an anomaly arises in the plant equipment or process, it shall be notified to operators so that suitable action can be made by operators.* *(3) Operation support* *When an operation support system is provided, even when the system function is lost, plant facilities should be operable safely.*
5. Control function	*(1) The control equipment shall be easy to operate so that misoperation becomes as small as possible.* *(2) Systems and equipment controlled from a control room should be designed so that they cannot be operated unsafely not to impair plant safety.* *(3) During an automatic operation, operators should be able to check the progress of the automatic operation.*

12.2 Efforts by Licensee

(1) Considerations in Design

Licensees take following considerations on human factors in designing a central control room.

The central control room is designed so that operating conditions of the reactor and other important equipment and principal plant parameters can be monitored at a glance and necessary actions can be taken in the room during normal operation and abnormal transients, and in an accident of a nuclear installation.

For example, *advanced BWRs (ABWRs) and present PWRs under design / construction are designed re-examining the plant layout and applying computer technologies, and also adopt "advanced central control panels", which is improved in operability and parameter monitoring capability. For example, by adopting a large display screen, which is easy to overlook power station conditions at a glance and to share information among operators, the opportunity of preventing and taking corrective action for an error is increased.* Moreover, centralized supervisory operation panels, which can make operators concentrate on observation and operation just sitting on chairs, are adopted, and automation scopes are expanded more than those of existing nuclear installations, which reduced operator's work loads for routine operation following a reactor scram. (See Fig. 12-1 and Fig. 12-2)

When remodeling control panels in the central control rooms of existing nuclear installations, extensive use of CRT has improved monitoring capability and operability of control panels.

Fig. 12-1 Main Control Panel of ABWR

Fig. 12-2 Main Control Panel of Latest PWR (proto type)

A guide for digitalized main control panels was established as one of the academic society and association standards, *JEAG 4615-2005*, "*Development and Design* Guide for Human Machine Interface of Computerized Central Control Rooms for Nuclear Power Plants" by the Japan Electric Association. *This guide refers to related international standards and overseas* *requirements, and reflects past fruits of design development etc. incorporating the trend of Japanese regulation and latest technical progresses. Specifically, requirements on functions and designs of the central control room (information display, control and operation equipment, alarm device etc.) and standard development and design processes of the human machine interface are defined.*

(2) Considerations in Operation Management

Licensees perform appropriate operation management during normal operation and in accidents.

1) Operational management

a. Organizations for operation

The manager of power generation division, responsible for the operation of a nuclear installation, controls operating shifts in charge of the operation and their supporting groups.

The shift supervisors have authority and responsibilities to take measures required in an accident, and are selected from those who conform to the criterion specified by the Minister of METI *and have suitable experiences and suitable competence.*

b. Shift of operators

Operators work in shifts. There are shifts devoted to education and training, in addition to operating shifts, to maintain and improve operator’s capability. The education and training of operators is one of the important elements of human factors. Details are described in Section 11.2.

When turning over shift duties, the shift supervisor makes sure to pass on the logbook, the supervisor logbook, keys, and precise description of operations to the succeeding supervisor. Each operating staff also transfers information of plant operation to the succeeding operating staff.

2) Preparation and amendment of operation procedures

Operation procedures are prepared for normal operation, failures and accidents and are constantly amended by lessons learned from incidents and accidents or by alteration of facilities.

Symptom-based procedures for multiple failures are prepared in addition to scenario-based procedures for design basis events. The symptom-based procedures enable prevention of accident progression without identifying the cause of an accident. Also prepared *were* the procedures addressing severe accidents exceeding design basis events, and accident management guidelines for the staff group supporting shift operators. The effectiveness of these procedures is verified by comparison with the results of the analysis of plant transient by the analysis code used in the application for licensing for establishment, and probabilistic safety assessment. Training course using simulator, based on a symptom-based procedure, is conducted at the operator training facility. Preparation of the procedures for emergency situation is expected to be effective for mitigation of operators' stress in an emergency.

3) Maintenance Management System

The maintenance department of a licensee controls the work of periodic inspection, modification works, etc. of a nuclear installation carried out by the plant manufacturer and many affiliated companies. A majority of human errors in the past occurred in the works associated with maintenance and repair, which means that the maintenance management by the licensee is very important.

The plant manager of a nuclear installation manages modification works, clarifying scope of work, scope of responsibility and authority. Maintenance of important equipment is carried out with a prior mock-up test.

Chief engineers (Chief Engineer of Reactors, Chief Electrical Engineer, Chief Engineer of Boiler and Turbine) perform verification and assessment of regulatory inspections by attending the regulatory inspections or confirming inspection records. They also perform verification and assessment, as appropriate, of the plans and results of regular inspections or modification works to prevent human errors in maintenance and management works.

Article 13 Quality Assurance

Each Contracting Party shall take the appropriate steps to ensure that quality assurance program are established and implemented with a view to providing confidence that specified requirements for all activities important to nuclear safety are satisfied throughout the operating life of a nuclear installation.

The regulatory body (Nuclear and Industrial Safety Agency, NISA), licensees, plant manufacturers and equipment suppliers (hereinafter referred to as “manufacturers”), conduct quality assurance (hereinafter referred to as “QA”) activities for nuclear installations in a coordinated way at each stage from design through operation and maintenance.

The basic concept of regulatory inspection has been continually renovated, seeking for more effective and efficient inspection activities, and in line with international trend in regulatory inspection. It has moved from a concept of system and component inspection to a concept where NISA encourages licensees to improve their QA activities and confirms the adequacy of them.

A series of wrongdoings by licensees and manufacturers, such as the TEPCO falsification issue, falsification of fuel assembly inspection results, improper construction work at a spent fuel reprocessing facility, revealed the importance of transparency of licensee’s QA activities and of regulatory supervision on them. *In view of the circumstances,* NISA *accelerated study on regulatory requirements concerning QA* and the NSC reviewed the regulation concerning QA operational safety activities and presented its views and opinions to NISA

Thus, in Japan, NISA established regulatory requirements for QA. NISA encouraged establishment of academic society and association standards in conformity to international standards and reviewed their technical adequacy and licensees apply newly established academic society and association standards to their QA activities. *NISA, when an inadequate quality assurance system is discovered at the Operational Safety Inspection to licensees, etc., directs the integrated check.*

Moreover, licensees and relevant parties are performing the self-controlled operational safety activities based on the academic society and association standards.

13.1 *Legislative Framework* for QA of Nuclear Installation

NISA conducts regulatory activities from the design stage to the operation stage, such as a licensing for establishment, an approval of construction plan, and an equipment inspection, on the basis of the Reactor Regulation Law and the Electricity Utility Industry Law.

In the in-service operational safety activities, the legislative requirements stipulate that the quality assurance system should be established and be included in the Operational Safety Program, based on the Reactor Regulation Law, and NICS supposes to check the state of implementation for the licensee through the Operational Safety Inspection.

The aim of the mechanism mentioned above is for the licensee to establish comprehensive and systematic QA programs, to be enabled to have firm belief in their operational safety activities and to implement them and to carry out their accountability to the public to obtain confidence of the public.

The key points of QA activities are; a) to involve top management, b) to be based on international standards on QA (ISO9001: 2000), c) to improve the activities continuously by Plan-Do-Check-Act cycle for planning, performing, and evaluating the operational safety activities, and d) to implement the full audit by the independent audit organization.

The Reactor Regulation Law stipulates that licensee's QA program should include 1) organization governing the performance of QA, 2) plan of activities for operational safety activities. 3) implementation of activities for operational safety activities., 4) evaluation of activities for operational safety activities,. and 5) improvement of operational safety activities.

Note) The term, activities for operational safety activities., means activities necessary to maintain safety, in maintenance work of facilities, operation of reactors, and transportation, storage and management of nuclear fuel materials or materials contaminated by nuclear fuel materials.

Licensee prepare their QA program of the nuclear facilities and implement them for the operational safety activities, according to JEAC 4111-2003, "Rules of Quality Assurance for Safety of Nuclear Power Plants" (hereinafter referred to as JEAC 4111-2003) established by the Japan Electric Association (JEA) in autumn of 2003 based on the ISO9001: 2000, and NISA evaluated the rules and accepted them as the standards to meet the regulatory requirements.

The contents of JEAC4111-2003 are shown in Table 13-1.

13.2 Verification of Quality Assurance by Nuclear and Industrial Safety Agency

NISA requires applicant for license at the each stage of *safety regulation* to submit appropriate QA program, and verifies implementation of QA program as follows:

(1) Review of Basic Policy for QA activities at Reactor Establishment Stage

NISA requires the applicant to submit the "Policy for Quality Assurance" attached to an establishment license application document in the licensing for establishment of nuclear installation.

(2) Confirmation of QA Program in Construction Stage

In application of the construction plan of nuclear installation, NISA requires the licensee to submit the "Description on QA Assurance" which the licensee of the reactor establishment should implement through each stage, such as the design, manufacturing, installation and functional test. NISA confirms that the licensee has prepared appropriate procedures to audit principal contractor's procedures such as the procurement quality control, the material control as well as the audit of the principal contractor's QA program and their process control, also on the licensee's responsibility.

(3) Verification of QA of Fuel Assembly

NISA requests the licensee of fuel assembly fabrication to submit the application form of the approval of fuel assembly design which describes such as the performance, the strength and the flow sheet of fabrication process for fuel assemblies, and "Description on Quality Assurance" attached at the time. When conducting the Fuel Assembly Inspection, the inspector of NISA verifies not only the licensee's test results but also the adequacy of licensee's test procedures by checking the extracted test processes without prior notice.

NISA requests the licensee to submit such as the description of QA program for the application for inspection of imported fuel assemblies. *In addition, the fuel assembly inspection is described in Section 14.2 in detail.*

(4) Verification of QA Activities throughout Operating Life

NISA verifies the licensee's QA activities throughout in-service for the nuclear installation as follows:

- Description of provisions relating to the quality assurance in the Operational Safety Program
- Verification of the Operational Safety Program through the Nuclear Safety Inspection

During outage of nuclear facilities, it is assessed that the periodic licensee's check is implemented appropriately as Audit of Licensee's Periodic Check System.

NISA is performing inspection on licensee's quality assurance since the 2004 fiscal year through the Operational Safety Inspection based on the Reactor Regulation Law. Specifically, NISA confirms licensee's situation of implementing the quality management system in a timely manner. In addition, in conducting the Operational Safety Inspection, nationwide inspectors from Nuclear Safety Inspectors Offices meet together at "Nuclear Inspectors Meeting", and participate in the effort aiming at equalization of nuclear-safety-inspection methods and information sharing by Nuclear Safety Inspectors, by performing "Operational Safety Inspection model". The national government and JEA are improving the guidelines in order to adequately conduct root cause analysis focusing organizational factors. The details are provided in Section 10.2.

In these Operational Safety Inspections and Audit of Licensee's Periodic Check Systems etc., following examples, which were caused because of inadequate preparation or functioning of the licensee's quality management system, were found out, and NISA instructed to implement the check for fitness of the quality assurance system and to take required measures to the licensees concerned as shown in the following table;

Licensee	*Date*	*Background and instruction*
The Kansai Electric Power Co., Inc.	*September 27, 2004*	*Background* *Inadequate preparation of the quality assurance system for ensuring systematic "nuclear safety" (Direct causes of the secondary system piping failure accident at the Mihama Power Station Unit 3)* *Instruction;* ・ *Verification of the quality assurance system* ・ *Establishment of effective measures to prevent recurrence*
Tohoku Electric Power Co., Inc.	*July 7, 2006*	*Background* ・ *Inadequate piping thickness control (prevention of recurrence of the secondary system piping failure accident of the Mihama Power Station Unit 3)* ・ *Inadequate "nonconformity management" and "procurement control"* *Instruction;* ・ *Integrated check of the quality assurance system* ・ *Establishment of effectual measures to prevent recurrence*

13.3 Implementation and Evaluation of QA Program by Licensee

Outlines of QA activities of licensee as follows;

(1) Establishment of QA Program

Licensee prepares QA program in accordance with JEAC4111-2003, and implement QA activities based on it. These programs cover the procedures of document control, design control, procurement control, management of inspection and testing, nonconformity management, and audit, etc. The licensee submits "basic Policy for QA" and "Description on QA" to NISA based on this quality assurance program as described in section 13.2 (1) through (4).

(2) QA activities at each stage of Design, Construction, Commissioning, and Operation (Maintenance)

QA activities are carried out by many organizations. The licensee clarifies the scope and responsibility of the manufacturer in QA activities, and entrust it with QA activities in its scope. In the same way, the manufacturer entrust its subcontractors with QA activities in their scope.

(3) QA audit, *observation and measurement*

As activities relevant to an evaluation of the licensee's quality management system, the top management reviews the quality management system of the company including nuclear installation periodically.

Moreover, as the observation and measurement (equivalent to Check of the Plan-Do-Check-Act cycle), the internal audit, observation of processes and measurement, and nuclear safety inspection and test are conducted.

As the internal audit, the independent audit that directly reports to the management is implemented by sections other than the nuclear power division.

And for procurement, following activities are conducted for suppliers;
(a) In selecting suppliers, their supply capabilities of procuring articles are evaluated.
(b) At the confirmation of procuring articles, in order to ensure that the procuring articles conform to procurement requirements, required inspections and other activities are conducted.

Moreover, the *supplier* audits subcontractors' activities in addition to its own internal audit.

Table 13-1 Contents of JEAC4111-2003,

"Rules of Quality Assurance for Safety of Nuclear Power Plants"

0. Introduction
1. Objective
2. Scope of Application
3. Definition
4. Quality Management System
5. Responsibility of Management
6. Management Control of Resources
7. Planning and Implementation of Job
8. Evaluation and Improvement

Article 14 Assessment and Verification of Safety

Each Contracting Party shall take the appropriate steps to ensure that:

(i) Comprehensive and systematic safety assessments are carried out before the construction and commissioning of a nuclear installation and throughout its life. Such assessments shall be well documented, subsequently updated in the light of operating experience and significant new safety information, and reviewed under the authority of the regulatory body;

(ii) Verification by analysis, surveillance, testing and inspection is carried out to ensure that the physical state and the operation of a nuclear installation continue to be in accordance with its design, applicable national safety requirements, and operational limits and conditions.

The national government (governmental offices responsible for safety regulations) and licensee perform and record the assessment and verification of the safety of nuclear installations in accordance with the legislative framework provided in Article 7, at each stage of planning, establishment, construction and operation of nuclear installations. The national government enacts and utilizes necessary regulatory guides for assessment and verification of each stage. When the regulatory framework is newly enacted or is updated, licensees carry out necessary assessments for the safety of nuclear installations and receive review by the national government.

In the safety assessment at the stage before construction, the Nuclear Safety Commission (hereinafter referred to as the NSC) deliberates on the safety review and assessment results by the Nuclear and Industrial Safety Agency (hereinafter referred to as NISA) from the view point of the licensee's technical capability and non-hindrance in the prevention of radiological hazards (implementation of double checking).

NISA confirms with the Periodic Inspection, the Operational Safety Inspection, the Audit of Licensee's Periodic Check System, and the periodic safety assessment (the periodic safety review and measures for aging management), that licensee's activities for operational safety are continuously performed appropriately to satisfy the safety design requirements, limiting conditions of operation for the facility. In addition, when guidelines and/or technical standards etc. are revised reflecting new knowledge etc., NISA directs the licensees to confirm the safety of the operating facilities, as necessary. The NSC supervises and audits the appropriateness of NISA's regulatory administration in the construction and operation stages after issuance of the license, from the view points of rationality, effectiveness and transparency.

In recent years, the probabilistic safety assessment methodology has been developed as a useful tool in supporting the conventional deterministic safety evaluation and the resultant risk information are used in regulatory activities such as development of accident management and the Periodic Safety Review (PSR) in Japan. Also introduction of safety goals *and performance goals (proposal)* that define the acceptable risk levels are under development.

New reporting items since the last report are as follows:

NISA developed the "Standard Review Procedures for Aging Management of Commercial Power Reactors" in order to review the technical evaluation report of aging and the long-term maintenance program submitted by licensees in accordance with the regulatory requirements.

NISA published the "Basic guidelines for use or risk -information in safety regulation of nuclear installations (trial use)", and "The quality guidelines for probabilistic safety assessment (PSA) of nuclear installations (trial use)", and is actively making efforts for utilizing such "risk information" in regulation.

In response to the revision of the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities by the NSC in September 2006, NISA required licensees to implement the seismic safety re-evaluation for the existing nuclear installations in accordance with the revised Guide.

14.1 Safety Assessment prior to Construction

A person who intends to install a nuclear installation submits license application documents *including the results assessed for the safety of basic design of the nuclear installation* to NISA pursuant to the Reactor Regulation Law.

NISA examines whether the application conforms to the licensing criteria prescribed in the Reactor Regulation Law.

The Minister of Economy, Trade and Industry consults with the NSC in order to hear opinions about the results of examination. The NSC deliberates on NISA's examination results from the view point of the licensee's technical capability and non-hindrance in the prevention of radiological hazards, with taking into consideration the opinions received at the public hearing.

The outlines of the safety assessment for Establishment License submitted and regulatory criteria are provided below, with a commercial power reactor (light water nuclear power reactor facility) used as an example.

(1) Documents for Establishment License Application

An application for establishing a commercial power reactor consists of a main text and attached documents detailing safety design, safety analysis, siting assessment, etc. in accordance with the provisions of the Reactor Regulation Law and the related legislation. The application document describes basic design with sufficient information for examining the safety.

(2) Method and Criteria of Assessment

1) Siting Assessment

The siting assessment of nuclear installations is conducted pursuant to the Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and application Criteria (hereinafter referred to as the "Guide" in this paragraph). The Guide requires that design, construction, operation and maintenance shall be implemented so that an accident may not occur irrespective of the place in which a nuclear reactor is established. In addition, in order to ensure the public safety in case of an accident, the following siting conditions are necessary in principle; a) there has been no event (natural disaster) in the past to induce a large accident and no such event is expected to occur in the future, and there are few events that escalates a disaster, b) nuclear reactors shall be, in relation to its engineered safety features, located at a sufficient distance from the public, and c) the environment of the nuclear reactor site including its immediate proximity shall be such that appropriate measures for the public can be implemented as required.

The conditions for judging the suitability for siting conditions are provided in "the guides for siting examination" of the Guide.

2) Assessment of Safety Design

The basic design and/or design concept of nuclear installations are confirmed that it conforms to the Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities (hereinafter referred to as the "Regulatory Guide for Safety Design") at the safety review and assessment conducted by NISA, and the safety is discussed and evaluated as a whole in accordance with the provisions of the Regulatory Guide for Safety Assessment. The evaluation method and the criteria using the Regulatory Guide for Safety Design and the Regulatory Guide for Safety Assessment are provided in Section 18.1 through Section 18.4.

14.2 Assessment and Verification of Safety Prior to the Commissioning

The licensee shall develop a construction plan for the installation of electric structures, and shall obtain the approval of NISA before starting construction in accordance with the Electricity Utilities Industry Law. After obtaining the approval of construction plan, the licensee shall undergo the pre-service inspection by NISA at every stage and completion of construction. For fuel assemblies to be loaded into the nuclear reactor, the licensee shall obtain the design approval and undergo the fuel assembly inspection by NISA. For the welding of pressurized parts, containments, etc., the licensee shall conduct the Licensee's Welding Check, and shall undergo review of the implementation system of the Licensee's Welding Check (Audit of Licensee's Welding Check System) performed by the incorporated administrative agency, Japan Nuclear Energy Safety Organization (hereinafter referred to as "JNES"). *Moreover, before starting an operation of reactors, in order to operate the nuclear installation safely, the licensee must obtain the approval of Operational Safety Program that describes the activity, commitments etc. (refer to Section*

19.3 of Article 19 for the contents).

Approval of construction plan and safety verification at the pre-service inspection, approval of fuel design and safety verification by the fuel assembly inspection and safety verification by the Audit of Licensee's Welding Check System are described in the following.

(1) Approval of Construction Plan and Safety Assessment and Verification at the Pre-Service Inspection

The licensee shall develop a construction plan for installation of electric structures and shall obtain the approval by NISA after obtaining the establishment license and before starting construction in accordance with the Electricity Utilities Industry Law. NISA reviews it to confirm that the detailed design of electric structures is not contradictory in the basic design or fundamental design policies at the establishment licensing stage, and that it is conforming to the technical standards in accordance with the Electricity Utilities Industry Law for the approval of construction plan concerned.

Licensees, after obtaining the approval of construction plan, undergo pre-service inspection by NISA at each construction stage and at the completion of all construction work, to verify that the construction is completed in accordance with the approval of the construction plan and that it is conforming to the technical standards. The pre-service inspection includes inspections on structure, strength and leak-tightness of each component and inspections on function and performance of the overall system of a nuclear installation. Details are shown in Table 14-1. The inspection at the time of criticality and completion of construction work in the table are so-called commissioning tests. Since October 2003, JNES conducts part of the above mentioned pre-service inspection.

(2) Approval of Fuel Design and Safety Verification by the Fuel Assembly Inspection

A person who intends to use fuel assemblies undergoes the Fuel Assembly Inspection, pursuant to the Electricity Utilities Industry Law, after obtaining the approval of fuel assembly design. NISA, in issuing the approval, verifies that the proposed fuel design takes into consideration the thermal characteristic, performance in radiation and corrosion resistance corresponding to operating conditions, and that it maintains sufficient strength through the years in service. NISA confirms this at inspections for each stage of fabrication that fuel assemblies are fabricated in accordance with the approved design and technical standards. The fuel assembly inspection is also required for reload fuel, regardless of whether or not there have been design changes. Since October 2003, JNES conducts part of the above mentioned fuel assembly inspection.

Imported fuel assemblies are also required to undergo and pass the fuel assembly inspection by NISA.

(3) Audit of Licensee's Welding Check System

Licensees perform Licensee’s Welding Check on welds of pressurized parts and containment, and the management system of Licensee’s Welding Check which undergoes review by JNES.

In addition, the Working Group for Welding Control and Inspection of the Electric Power Safety Subcommittee under the Nuclear and Industrial Safety Subcommittee prepared a report "Application Improvement in an Audit of Licensee's Welding Check System" concerning appropriate verification procedures of the Licensee’s Welding Check system, welding specialists and welding method in November 2006.

14.3 Assessment and Verification of Safety during Operating Life Time

The licensee performs a comprehensive safety verification of nuclear installations by the periodic safety assessment, Periodic Licensee's Check, surveillance pursuant to the Operational Safety Program, evaluation and investigation at the time of an accident or failure and measures to prevent recurrence, and undergoes the spot entry inspection at any time in addition to the Operational Safety Inspection, Periodic Inspection, and the Audit of Licensee's Periodic Check System during the operating lifetime.

Safety verification by inspection and periodic safety assessment are described in the following:

(1) Verification of the Safety by Inspection

NISA performs the Operational Safety Inspection for verification to confirm the appropriateness for activities of the licensee to ensure operation safety and the Periodic Inspection to confirm activities to ensure the integrity of equipment of nuclear installations. Furthermore NISA performs the comprehensive assessment of the system, method etc. of the Periodic Licensee's Check in response to the notification of the review results of the Audit of Licensee's Periodic Check System performed by JNES. These inspections are performed in accordance with related legislations and regulations as follows.

NISA has set-up resident Nuclear Safety Inspectors at nuclear installations and performs four inspections per year (Operational Safety Inspection) on the observance of Operational Safety Program in accordance with the Reactor Regulation Law to determine the status of compliance to various regulations for the safety and the status of the activities for safe operation performed by the licensee.

In accordance with the Electricity Utilities Industry Law, NISA and JNES perform the Periodic Inspection of structures and components important to safety within a time interval that does not exceed 13 months after the day of commissioning and the final day of the last Periodic Inspection.

The inspections which used to be conducted by the licensee as a self-controlled inspection was redefined in October 2003, by the amendment of the Electricity Utilities Industry Law as a Licensee’s Periodic Check, and JNES performs the audit type inspection (Audit of Licensee's Periodic Check System) to review the implementing system, planning and management of

Licensee’s Periodic Check.

After two years of experience, the firm establishment of these inspection systems has been promoted considerably. However, an increase is expected in the number of nuclear installations that have been operating for many years, and it is necessary to improve measures for aging management further, the Task Force on Inspection System had resumed in November 2005. Section 19.3 provides the status of the Task Force activities.

(2) Periodic Safety Assessment

Pursuant to the request of the Ministry of International Trade and Industry (at present, Ministry of Economy, Trade and Industry), licensees had performed the periodic safety review voluntarily since 1992. However NISA has decided that it is necessary to define the position of the periodic safety review as a part of reconsideration of nuclear safety regulations.

In accordance with the Reactor Regulation Law, NISA decided in October 2003 to obligate licensees to implement the "Periodic Safety Review (PSR)"after every 10 year interval since the first review at the time not exceeding ten years after commissioning and the "Aging Management Review" within 30 year limit after commissioning.

1) Periodic Safety Review (PSR)

PSR is a licensee's effort to evaluate his operational safety activities performed since the commissioning of the nuclear installation about every ten years and to obtain the prospect of capability for the nuclear installation to continue safe operation thereafter with keeping a higher level than or equivalent to the newest nuclear installation.

In December 2005, NISA revised the Rules for the Installation, Operation, etc. of Commercial Power Reactors under the Reactor Regulation Law. And the details of the licensee’s activities that should be performed at the periodic safety review were clarified in "Evaluation on the Status of Implementing the Operational Safety Activities at Reactor Facilities" and "Evaluation on the Reflection Situation of the Latest Technical Knowledge for Operational Safety Activities Performed at the Reactor Facilities".

In order to understand the licensee’s status on the improvement in aging management and on the prevention of deterioration in the organization culture, both in short term and in long term, NISA decided to verify the licensee’s maintenance management activities and the status of the organization culture at a usual Operational Safety Inspection (for short term understanding) from January 2006. Furthermore NISA also decided to verify that the licensee understand the status appropriately and is taking measures, such as intensification of surveillance, at the periodic safety review (for long term understanding) implemented every ten years after the start date of the operation of reactor.

2) Aging Management Review

In October 2003, NISA added the Aging Management Review in the provisions of

"Periodic Assessment of Reactor Facilities" to the "Rules for the Installation, Operation, etc. of Commercial Power Reactors" and provided it as one of the requirements in the Operational Safety Program to implement Measures for Aging Management.

Matters to be implemented as the Aging Management Review are: a) to analyze the impacts of technically conceivable aging phenomena on components and structures of nuclear power stations with safety functions at a time within 30 years after commissioning, and to technically evaluate the possibility for prevention of the loss of function of the components and structures due to aging phenomena under the present maintenance activities provided to them, b) to extract new maintenance measures from the technical evaluation results to develop the ten-year maintenance program. c) to re-evaluate the ten-year maintenance program with a ten-year interval. And NISA decided to confirm the status of the implementation of these items which shall be implemented as a part of the quality assurance systems of the Operational Safety Program.

The Aging Countermeasure Examination Committee under the Nuclear and Industrial Safety Subcommittee, issued the "Enhancement of Measures for Aging Management at Commercial Nuclear Installations" in August 2005. In response to the report, NISA prepared the "Guidelines in Implementing Measures for Aging Management at Commercial Nuclear Installations" and the "Standard Review Procedures on Measures for Aging Management of Commercial Power Reactors" in December 2005. And, JNES issued the "The compilation of Technical Information concerning Measures for Aging Management", which disclosed the standards, view points and evaluation points when NISA and JNES assess and review licensee's technical evaluation reports and long-term maintenance programs.

The aging technical evaluation reports for Unit 3 of the Fukushima Daiichi Nuclear Power Station, Tokyo Electric Power Co., Inc., Unit 1 of the Hamaoka Nuclear Power Station, Chubu Electric Power Co., Inc., and Unit 3 of the Mihama Power Station, the Kansai Electric Power Co., Inc. are submitted at present.

NISA assessed the technical evaluation report and the long-term maintenance program for Unit 1 of the Hamaoka Nuclear Power Station, Chubu Electric Power Co., Inc. on January 31 2006, and the result was reported to the NSC in May 2006 as follows.

Summaries of the assessment results; 1) the system for implementing the technical evaluation is appropriate, 2) the technical evaluation implemented for aging, the technical evaluation for ensuring seismic safety, and conservation measures are appropriate, and 3) the long-term maintenance program based on the technical evaluation is appropriate.

The same assessments were carried out for the other two nuclear reactors mentioned above. NISA will study issues for improvement extracted from the actual evaluation results of the three nuclear installations so that effective improvement is achieved.

Also, in response to the "Completion of Measures for Aging Management at Commercial Nuclear Installations" issued in August 2005, licensees reported to the Aging Countermeasure Examination Committee, the status of efforts for the measures for aging management for the following matters; (1) ensuring transparency and effectiveness, (2) preparing technical information infrastructure, (3) preventing deterioration in corporate culture and organization culture, and keeping and improving technical capabilities, and (4) steadily achieving the accountability on measures for aging management. From now on, the licensees will carry out confirmation of the situation of measures for aging management at the Periodic Licensee's Check, and NISA decided to verify the licensees' implementing situation through the Periodic Inspection, Audit of Licensee's Periodic Check System, Operational Safety Inspection, etc.

NISA launched the Coordination Committee on Technical Information in JNES, in order to reinforce the measures for increasing aged plants, and in order to share domestic and overseas technical information for effective utilization among the industrial world, academic and governmental institutions.

Furthermore, an ad-hoc committee that consists of NISA, JNES, universities, research organizations, electric utilities, nuclear plant manufacturers, plant engineering companies, etc. was established under the Atomic Energy Society of Japan. From July 2004 through March 2005, the ad-hoc committee had prepared a load map on measures for aging management and long life-time safe operation of light water reactors.

In addition, NISA regards the aging of nuclear installations as one of the more important research subjects on safety. Clarification of the aging phenomenon and prediction of the aging process, development of early detection and detailed measurement methods of cracks and deteriorations, and development of the structural integrity evaluation method are the subjects of concern. The research on irradiation assisted stress corrosion cracking (IASCC) using the material testing reactor (JMTR) of the Japan Atomic Energy Agency, the probabilistic fracture mechanics research, and JNES's data-base and technology development for flaw detection and sizing, etc. are currently underway.

(3) Assessment at the Occurrence of an Accident or a Failure, Survey on Accidents and Failures and Measures to Prevent Recurrence

Activities of licensees and regulatory bodies for assessment at the occurrence of an accident or a failure, survey on accidents and failures and measures to prevent recurrence are provided in Section 19.6 and 19.7. Furthermore, Section 6.2 provides actions taken for accidents and failures at existing nuclear installations.

14.4 Utilization of Probabilistic Safety Assessment in Regulation

(1) Utilization of Probabilistic Assessment in the Actual Regulation

1) Judgment on the Necessity of the Protection for External Events

In the safety review and assessment for an application for reactor establishment license, the protection is required for external events (including natural events and human induced events which may occur inside or outside a facility), when the possibility of occurrence of events that affect safety related facilities and equipment exceeding a certain definite value, and the adequacy of the protection design is a subject at the safety review and assessment..

For example, at a turbine missile event affecting the reactor facility, the probability of damage to the reactor facility (coolant pressure boundaries of the containment and the spent fuel pool) has thus been so far evaluated. If the result exceeds the criterion, it is required to consider it in the design as "a missile assumable to generate inside the reactor facility."

As for an aircraft drop to the reactor facility, the probability of an aircraft drop to the reactor facility has been evaluated. And if the result exceeds the criterion, it is required to consider it in the design as "a human induced external event." In addition, NISA enacted the "Criterion for Evaluation of the Aircraft Drop Probability to a Commercial Nuclear Installation (by-law)" in July 2002.

2) Evaluation on Effectiveness of Accident Management Measures

The licensees have implemented the PSA for nuclear power reactors under operation or construction to evaluate the soundness of the core and the containment in the case of a severe accident, and utilized the result for effectiveness assessment of their accident management *(AM)* measures. Internal events during operation were subject to their analysis, and the results were utilized to discover AM measures and to evaluate their effectiveness.

3) Utilization of the Probabilistic Safety Assessment (PSA) at the Periodic Safety Review

As provided in 14.3 (2), licensees have implemented the PSA for internal events during power operation using the latest data for the periodic safety review for commercial nuclear installations since 1992. And the results were utilized for safety assessments, understanding the features on the safety of nuclear installations concerned, and verification for the effectiveness of accident management measures etc. Furthermore, at the periodic safety review after 2001, the PSA for internal events during the shutdown condition has been additionally implemented. *The position of the periodic safety review in legislations was identified in October 2003, but implementation of the PSA was not obliged by the law. However licensees are requested to implement the PSA as part of their independent activities since then.*

4) Assessment of Impacts and Measures of Operating Experience Etc.

In the study on the measures to prevent recurrence of the pipe rupture accident which occurred in November 2001 on the steam condensing line (SCL) of the residual heat removal system (RHRS) of the Unit 1 of Hamaoka Nuclear Power Station, the PSA, with considerations of the pipe break accident specifically to the SCL of the RHRS, was implemented on the core integrity, and it resulted that the pipe rupture of the accident concerned does not significantly increase the risk. It was also evaluated that several proposed measures were effective in reducing the risk.

In addition, the effectiveness of provisional measures for functional impediment of the emergency core cooling system strainer of BWRs and the screen of containment-recirculation sump of PWRs was confirmed by the PSA.

(2) Activities for Introduction of the Safety Regulation Utilizing Risk Information

In order to establish a more effective and efficient regulatory system, many countries are making efforts to utilize risk information for safety regulations, though the extent of such efforts is different.

Also in Japan, the NSC decided the "Basic Policies on Introduction of Nuclear Safety Regulation Utilizing Risk Information" in November 2003.

NISA together with JNES, receiving the decision, published the "Fundamental Concepts of Utilizing the "Risk Information" at Nuclear Safety Regulations" in May 2005, and provided the way of thinking for utilizing the "risk information" at nuclear safety regulations. In addition, the "Present Implementation Plan of Utilizing the "Risk Information" at Nuclear Safety Regulations" was published, and the study on utilizing the "risk information" was promoted in accordance with the implementation plan. The implementation plan was revised based on the progress since then in January 2007.

NISA published the "Guideline for Utilizing the "Risk Information" at the Safety Regulations of Nuclear Power Stations" (trial use)", and provided the guidelines for utilizing "risk information" at the safety regulations of nuclear power stations in April 2006. Furthermore as a guideline for ensuring quality assurance of the PSA in utilizing the "risk information", the "Guidelines for the Quality of the PSA for Nuclear Power Stations (trial use)" was published.

In addition, the NSC established "Task Force for Introduction of Safety Regulations Using Risk Information (RIR)", in this Task Force, the current status of relevant organizations' approaches to the application of RIR and the issues to be solved for the wider utilization of risk information were compiled.

(3) Introduction of Safety Goals and Performance Goals

The NSC issued the "Interim Report on the Investigation and Review on Safety Goals" in December 2003, and the performance goals were established for nuclear installations in March 2006. The outlines of the goals are as follows.

1) Safety goals

The safety goal should be established for all activities in the utilization of nuclear *energy that may have an adverse influence of radiation exposure on the public.*

The objectives to establish the safety goals are as follows;

- *To make it possible to assess regulatory activities in utilizing nuclear energy at various fields with same standards for reasonable and consistent evaluation among them,*

- *To make it possible to exchange opinions on the way of nuclear regulatory activities of national governments, such as establishment of guidelines and standards, among the national government and people more effectively and efficiently, and*

- *Make it possible for licensees to implement their independent risk management activities more effectively and efficiently to meet the expectation of the regulatory authority.*

Along with these objectives, first of all, the safety goals are applied as reference to make judgment on the whole regulatory activity, in terms of rationality and consistency, and it is considered as appropriate to start with more general applications, on a specific facility, after abundant experience for the safety goals are accumulated.

The safety goal is of two fold. One is the qualitative goal, which is a controllable level of risk due to an accident that licensees must observe under the nuclear safety regulations. The other is the quantitative goal that specifies the numerical value corresponding to the acceptable level of the risk. In this context, the risk during the normal operation of nuclear power reactor facilities is excluded. And as the indices for quantitative goals, the death risk of the average individual of the public who lives in a certain range is used.

The proposal on safety goals are made of the following configurations.

a. Proposal to Qualitative Goal

The possibility of health damage to the public by emission of radiation or release of radioactive materials accompanied with activities for utilization of nuclear energy should not meaningfully increase the risk of damage to the public's health in daily life.

b. Proposal to Quantitative Goal

The mean value of acute fatality risk by radiation exposure resultant from an accident of a nuclear installation to individuals of the public, who live in the vicinity of the site boundary of the nuclear installation, should not exceed the probability of about $1x10^{-6}$ per year. And, the mean value of fatality risk by cancer caused by radiation exposure resulting from an accident of a nuclear installation of individuals of the public, who live in the area but some distance from the nuclear installation, should not exceed the probability of approximately $1x10^{-6}$ per year.

2) *Performance Goal*

It is reasonable to review and indicate the level that will be understood as the performance goal to conform with the safety goal, according to the characteristics of each accident that could occur at nuclear installations.

The preparation of the PSA method is advancing and the following performance goals are proposed for nuclear power generation facilities that have experience in utilizing the risk information.

a. *Indices for the Performance Goal*

The following indices are also used, because they well represent the facility performance on the integrity of a reactor core and the integrity of the confinement function of a containment unit, and are clearly defined and appropriately quantified.

Index 1: Frequency of core damage (CDF)

Index 2: Containment loss-of-function frequency (CFF)

b. *Proposal to Indices Value*

The knowledge obtained by the PSA for domestic nuclear installations implemented by the national government, research organizations, licensees etc. and the PSA results in the U.S. etc. were studied as references, and the following indices values were proposed as a proposal to the performance goals corresponding to the proposal to the safety goals.

Index value 1: CDF: 10^{-4} */ reactor-year approximately.*

Index value 2: CFF: 10^{-5} */ reactor-year approximately.*

These indices values shall be satisfied concurrently.

From now, studies for preparing a framework for use of performance goals in safety regulations, application to nuclear installations other than commercial nuclear installations, and a high safety level in future reactors are required.

14.5 Assessment of Seismic Safety

(1) Assessment of the Seismic Safety of Existing Nuclear Installations Following the Revision of the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities

The NSC revised the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities, etc. in September 2006 (refer to the report for Article 18).

Following the revision, NISA required licensees to assess the seismic safety for existing *nuclear installations etc. in accordance with the revised Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities, and to report the results.*

Each licensee etc. submitted his implementation plan for the seismic safety evaluation to NISA in October 2006.

Each licensee submitted the process to implement the safety assessment within two to three years, and the implementation plan describes that the assessment will be implemented in the order of geological and active faults investigation, determination of the design basis earthquake ground motion, and evaluation for seismic safety.

NISA decided to verify the adequacy of the contents of the licensees' reports for the seismic safety assessment, and then to report the verified results to the Aseismatic and Structural Design Subcommittee of the Nuclear and Industrial Safety Subcommittee of the Advisory Committee for Natural Resources and Energy.

As of the end of June 2007, the assessment results for Hamaoka Unit 3 and 4 were submitted from the licensee etc. (February 21 and January 25, 2007, respectively). NISA will strictly verify the adequacy of the reported results by means of the study at the Aseismatic and Structural Design Subcommittee of the Nuclear and Industrial Safety Subcommittee, etc.

The NSC established "Investigation Project Team on Seismic Safety of Nuclear Facilities" in July 2007. The mandate of this Project Team is to review the NISA's review results on the Seismic Safety Re-evaluation of the existing nuclear facilities that will be done in near future.

(2) Assessment of Seismic Safety for the Nuclear Installation Experienced a Major Earthquake

1) Assessment of Seismic Safety for the Onagawa Nuclear Power Station at the Earthquake Occurred at Miyagiken-oki in August 2005

NISA directed Tohoku Electric Power Co., Inc. to analyze the factors of the earthquake ground motion that exceeded the design basis earthquake ground motion at the Onagawa Nuclear Power Station that was confirmed at the earthquake of Miyagiken-oki on August 16 2005, and to assess the seismic safety of the equipment important to safety. The seismic safety assessments (reports) for Onagawa Unit 2 and 3 were submitted on November 25, 2005 and January 20 2006, respectively. NISA reported these seismic safety assessments (reports) to the experts of the Aseismatic and Structural Design Subcommittee of the Nuclear and Industrial Safety Subcommittee. And NISA asked JNES to make a crosscheck assessment and study the seismic safety. NISA determined that the licensee's study results on the seismic safety was appropriate, and informed the conclusion to Tohoku Electric Power Co., Inc.

Furthermore, the seismic safety assessment (report) for the Onagawa Unit 1 was submitted by Tohoku Electric Power Co., Inc. on May 19, 2006. The additional report for the seismic safety for the reactor building foundation ground based on an investigation result of additional boring was submitted on June 12. The amendment report including a *piping wall-thinning survey implemented after the report for Onagawa Unit 1 and the analytical evaluation results were submitted on August 22.*

NISA held the Aseismatic and Structural Design Subcommittee of the Nuclear and Industrial Safety Subcommittee, and received the expert's opinions on the reports, and NISA asked JNES to make a crosscheck assessment and study the seismic safety for Unit 1. NISA determined that the licensee's study results for seismic safety was appropriate, and reported the conclusion to Tohoku Electric Power Co., Inc. on September 13, 2006.

2) Assessment of the Seismic Safety for the Shika Nuclear Power Station at Noto-Hanto Earthquake in March 2007

Unit 1 and 2 of the Shiga power station were in shutdown condition at the time of the Noto-Hanto earthquake on March 25, 2007. As the ground motion during the Noto-Hanto earthquake exceeded some of the design basis earthquake ground motions, the licensee re-evaluated the seismic safety of Unit 1 and 2, and reported the results to NISA on April 19, and additional report and amendment were also submitted on June 1 and August 20, 2007respectively. NISA decided to confirm the reports that were submitted by the licensee, and NISA held the Aseismatic and Structural Design Subcommittee of the Nuclear and Industrial Safety Subcommittee, and received the expert's opinions on the reports. NISA determined that the licensee's study results for seismic safety was appropriate, and reported the conclusion to Hokuriku Electric Power Co., Inc. on August 27, 2007 receiving experts' opinions at the Aseismatic and Structural Design Subcommittee of the Nuclear and Industrial Safety Subcommittee of the Advisory Committee for Natural Resources and Energy.

14.6 Promotion of Nuclear Safety Research

(1) Prioritized Nuclear Safety Research Program by the NSC

The NSC proposed nuclear safety research (prioritized nuclear safety researches) that should be performed selectively for about five years from the 2005 fiscal year. In the proposal, the following important research areas were proposed as safety researches for nuclear installations; 1) the regulatory system area (example: use of risk information, assessment of root-causes of failures and accidents), 2) light water reactor area (example: safety analysis, material degradation and aging management, seismic safety technologies) and nuclear reactor disaster prevention technologies.

In order to reflect the latest technical knowledge obtained from the results of safety researches upon safety regulations and improvement of safety regulations, regulatory authorities and research organizations should come to a mutual understanding, and the regulatory authorities should make efforts to identify the outcome of safety research required for safety regulations and to show clearly how the results of safety researches are to be utilized. Furthermore, research organizations are required to arrange and present the issues and results for safety research that meet the requirements of regulatory authorities appropriately. Therefore, the NSC understands and evaluates the status of research in related research organizations.

Furthermore, the NSC holds a debrief session on the results of safety research every year. From the 2006 fiscal year, in order to support safety research to be implemented smoothly, the NSC started Nuclear Safety Research Forum. And the relevant persons coming from industries, academia and the government meet together, and they deliberate on regulatory required safety research and utilization, the promoting system, and international contributions and they exchange information.

(2) Safety research by NISA

Since ensuring safety of nuclear installations is one of licensee's responsibilities, the licensee must implement all safety research necessary for ensuring safety. On the other hand, the regulatory body is required to promote appropriate safety research in order to make a more scientific judgment.

NISA has performed extensive safety research and has implemented the following activities during the report period.

1) Review of Plans and Results of Research by the "Fundamental Research Subcommittee for Ensuring Nuclear Safety"

NISA established the Fundamental Research Subcommittee for Ensuring Nuclear Safety under the Nuclear and Industrial Safety Subcommittee in September 2006, and decided to plan, implement and assess research for the nuclear safety infrastructure with a load map to be developed for building a framework systematically and efficiently aiming at ensuring safety by industrial societies and regulatory authorities etc. in the fields that should be focused on from now on. Then, NISA will plan, implement and assess the fundamental research for ensuring nuclear safety.

2) Continuation of Research Facilities and Preparation for Strategic Research Organization

Recent years, research facilities for nuclear safety are under the threat of closure internationally. Therefore, NISA reviewed the continuation of research facilities at the Fundamental Research Subcommittee for Ensuring Nuclear Safety. The subcommittee proposed that the Japan Material Testing Reactor (JMTR) of the Japan Atomic Energy Agency should be positioned as the strategic fundamental safety research facility.

In addition, after the pipe rupture accident at Unit 3 of the Mihama Power Station, it was decided to prepare a strategic organization for nuclear safety research in Fukui Prefecture, where the Mihama Power Station is located. NISA defined the safety research to be performed, and contributed to the strategic research plan.

3) *Promotion for International Joint Studies*

NISA and JNES have been promoting the international cooperation research actively. Especially in the reporting period, the cooperation with OECD Nuclear Energy Agency (NEA) was promoted. Specifically, NISA supported the OECD/ROSA plan in the field of *thermal hydraulic safety research that the Japan Atomic Energy Agency will perform as the first Japanese host organization.*

In the fuel material research, NISA participated to the OECD/CABRI plan and the Halden plan for some time. NISA started the SCAP plan that is the project specially funded by Japan. In the project, stress corrosion cracking (SCC) and cable aging are being studied in the NEA since measures for aging management are very important. In addition, NISA has been continuously participating in many of the OECD projects performed abroad and participating in the IAEA's CRP (Coordinated Research Activities), etc.

(3) Safety Research by JNES

JNES, as a specialized agency supporting NISA for technical fundamentals, promotes safety research necessary for providing the scientific knowledge that should be reflected in the safety regulations, such as preparation of safety standards and criteria necessary for the safety regulation of nuclear installations etc.,.

(4) Safety Research by the Japan Atomic Energy Agency

The Japan Atomic Energy Research Institute and the Japan Nuclear Cycle Development Institute unified in October 2005, and the Japan Atomic Energy Agency was established as an incorporated administrative agency, which performs comprehensive research and developments for nuclear energy. The Agency possesses a large number of facilities necessary for implementing the safety research, and has human resources over broad area of expertise. Therefore, the Agency plays the central role in implementing safety research, and is required to play the role to support the safety regulation technically utilizing integrated nuclear energy research facilities.

(5) Safety Research by Licensees etc.

In order to cope with longer life and sophistication in use of light water reactors, licensees implement research that is needed for improvement in safety, reliability and economical efficiency. The outcome of the research is reflected in preparation and sophistication of the private sectors' standards as needed, which contributes in ensuring safety.

Table 14 - 1 Outline of Pre-Service Inspections

Time of Inspection	Contents of Inspection
(1) At the time of installation of each structure and component	Test of structure, strength and/or leak tightness of reactor, reactor cooling system, instrumentation and control system, fuel handling system, radiation management system, waste processing system or reactor containment structure is performed, when each item is installed and ready to be tested. Specifically, material inspection, structure inspection, pressurized leak test, inspection on foundation and support structure are performed
(2) At the time of installation of steam turbine and auxiliary boilers	Test of steam turbine structure is performed when installation of bottom half part of turbine casing is completed. Test of structure, strength and/or leakage on auxiliary boiler is performed when its main part is completely assembled.
(3) At the time of fuel loading	When the reactor is ready for fuels to be loaded, inspections of systems around reactor, items required ensuring safety before fuel loading, and items for which inspection would be difficult after fuel loading are performed. In the case of BWR, inspection of main steam bypass valves, inspection of function and performance of those systems as control rod drive system, core spray system, residual heat removal system, etc. and functional inspections of safety protection system, etc. are performed.
(4) At the time of criticality	When the reactor attained criticality, inspections are performed on nuclear characteristics of reactor core, and overall function and performance of nuclear installations which can be performed only after fuel loading. In the case of BWR, an inspection to confirm shutdown margin at full fuel loading, inspections of control rod full stroke test, effective multiplication factor at the first criticality and moderator temperature coefficient tests are performed.
(5) At the time of completion of construction	When all construction work under the Construction Plan has been completed, inspections are performed on performance of systems around reactor, overall functions and performances of nuclear installations that can be confirmed after fuel loading, and functions and performance of systems other than those around reactor. In the case of BWR, inspections are performed on one control rod scram test, loss of external power-supply test, generator load interception inspection, plant trip inspections, and load inspections.

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Article 15 Radiation Protection

Each Contracting Party shall take the appropriate steps to ensure that in all operational states the radiation exposure to the workers and the public caused by a nuclear installation shall be kept as low as reasonably achievable and that no individual shall be exposed to radiation doses which exceed prescribed national dose limits.

The standards of radiation protection for the general public and personnel engaged in radiation work in Japan are prescribed in the laws and legislations, such as the Reactor Regulation Law, the Electricity Utilities Industry Law and the Industrial Safety and Health Law, etc. The 1990 recommendations of the ICRP are incorporated into their provisions of radiation protection with due considerations. Consequently, licensees have kept the radiation exposure doses of personnel engaged in radiation work below the dose limit, as a matter of course, and have attempted to reduce the exposure based on the ALARA policy.

15.1 Summary of Laws and Requirements on Radiation Protection

The national standards of radiation protection for a nuclear installation are provided in the Reactor Regulation Law, the Electricity Utilities Industry Law and the Industrial Safety and Health Law, etc. and related ordinances, ministerial orders and notifications based on these laws, and guidelines. The 1990 recommendations of the ICRP are given due consideration and have been incorporated into legislation and regulation. *At present the revision of the 1990 recommendations of ICPR is in progress at ICRP, Japan will incorporate the future revision, if necessary, based on discussions at IAEA etc.*

As the clauses on radiation protection, a ministerial ordinance, ‘The Rules for Commercial Nuclear Power Reactors concerning the Installation, Operation, etc.’ under the Reactor Regulation Law is established, which prescribe area control for radiation protection, radiation control of workers in the radiation controlled areas, measurement and surveillance of radiation levels, monitoring of discharged radioactive materials, and maintenance of radiation control equipment. Also the Dose Limits Notification are enacted within the said ministerial order, which prescribe dose limits and concentration limits of radioactive materials both inside the controlled area and outside the peripheral monitoring area, dose limits and concentration limits of radioactive materials for radiation workers, and dose limits for workers in emergency activities.

In order to ensure those rules are complied with, each licensee, is required to prescribe in the Operational Safety Program, 1) radiation control area, access controlled areas, and peripheral monitoring area and access control to these areas, 2) monitoring equipment at air ventilation and water discharge, 3) monitoring of the dose, the dose equivalent, the concentration of radioactive materials and the density of the surface radioactive materials of objects contaminated by radioactive materials, and the decontamination, 4) maintenance of radiation monitoring equipment.

The ministerial order “Ordinance of Establishing Technical Standards for Nuclear Power Generation Equipment” based on the Electricity Utilities Industry Law, provides technical standards for radiation control equipment (biological shielding walls, ventilation facilities, instrumentation devices, alarm devices, and waste processing equipment, etc.) at nuclear installations. NISA confirms that such radiation control equipment conforms to the ministerial order at issuing approval of the construction plan and when they conduct an inspection of the equipment.

The Industrial Safety and Health Law provides that licensees (employers of laborers) take measures to prevent damage to the health of radiation workers, including radiation exposure, throughout their period of employment, and it requires that they be educated on issues of health and safety, work environment monitoring and medical examination of workers. On the basis of the law, the Ministry of Health, Welfare and Labor has enacted a ministerial order, ‘the Rules for Prevention of Hazards from Ionizing Radiation’, which prescribes controlled areas, dose limits and measurement, protection from external radiation, and prevention of radioactive contamination.

Radioisotopes etc. used in nuclear installations are also regulated in accordance with ‘the Law Concerning Prevention from Radiation Hazards due to Radioisotopes, etc.’ in the same manner as regulated by the Reactor Regulation Law.

In examining the license to establish a nuclear installation, it is confirmed that the application conforms to the Regulatory Guides established by the NSC in addition to the legislation and technical standards mentioned above. The Regulatory Guide for the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities gives dose target guide to reduce the discharge of radioactive materials from a nuclear installation into environment and the dose of the public as low as reasonably achievable (ALARA).

Each licensee has defined the release control value of liquid wastes and gaseous wastes in the Operational Safety Program based on this Regulatory Guide for the Annual Dose Target for the Public in the Vicinity of LWR.

The 1990 Recommendation of the ICRP(Publication 60) has been, after examination by the Radiation Council, incorporated into national legislations and regulations on radiation protection, by revision of related ministerial orders and notifications in April 2001 with the following additional considerations. First, the radiation controlled area is defined where the dose may exceed 1.3 mSv / 3 months, corresponding to 5 mSv/year which is a special dose limit to the public. Second, the occupational dose limit for female workers is set at 5 mSv / 3 months, an allocated value for a shorter period, reducing possible dose of a potential embryo. The dose limits in emergency work remain 100 mSv/ year as before, considering the IAEA BSS.

The Radiation Council is an organization established under MEXT for the purpose of coordinating technical standards on prevention of radiation hazards. The Radiation Council submits reports related to inquiries from related administrative organizations, or advises them as necessary.

15.2 Laws and Requirements and Response of Licensees

(1) Dose Limits

1) Dose Limits for Controlled Areas

The Rules for Commercial Power Reactors and the Dose Limits Notification requires licensees to establish a radiation controlled area including the reactor room, spent fuel storage facilities and radioactive waste disposal facilities, where the dose of external radiation may exceed 1.3 mSv for three months, or where the concentration of radioactive materials in the air or the surface density of radioactive materials may exceed the values specified in the Notification, respectively, and to establish necessary measures to be taken in these areas.

2) Dose Limits for Occupational Exposure

The dose limits for radiation workers are specified in the Dose Limits Notification as listed in Table 15-1

Table 15-1 Dose limits for radiation workers

Item	Limit
1. Effective dose limits	
a) Radiation workers	100 mSv / 5 years, but do not exceed 50 mSv for any year
b) Female	100 mSv / 5 years, but do not exceed 5 mSv for any 3 months
c) Pregnant female	100 mSv / 5 years, but do not exceed 5 mSv for any 3 months and do not exceed 1 mSv from internal exposure during pregnancy
2. Equivalent dose limits	
a) Eye lens	150 mSv/ year
b) Skin	500 mSv/ year
c) Female abdominal region	2 mSv from notification of pregnancy to delivery
3. Dose limits in emergency	
a) Effective dose	100 mSv/ incident
b) Equivalent dose for eye lens	300 mSv/ incident
c) Equivalent dose for skin	1 Sv/ incident

Licensees have paid much attention not only to comply with the dose limits but also to reduce doses in line with ALARA concept by incorporating the following activities:

- reducing the radiation source in systems and components of a nuclear installation,
- keeping distance from or setting shields against radiation sources,
- reducing working time in a radiation environment.

Consequently, the exposure doses of the radiation worker, etc. have been successfully reduced to the level as shown in Annex 2.

Exposure doses of radiation worker in a commercial nuclear installation during the reporting period are summarized below;

a. Individual dose at commercial nuclear installations

The average of individual dose at commercial nuclear installations for the reporting period was *1.0* to *1.4* mSv / year, and it shows *the gradual decrease trend* for the past several years. Thus, the doses are well within the dose limit prescribed in the notification.

In fiscal year *2006*, the average of annual dose of radiation workers was *1.0* mSv and the maximum annual individual dose experienced per nuclear installation was *19.7* mSv, these numbers were within the dose limit of the notification, which are *similar to the value of 1.0 mSv and slightly lower than the value of 19.8 mSv for the previous year of 2005, respectively. No worker who had worked in multiple nuclear installations* exceeded 20 mSv, which is well below the dose limits. The number of radiation workers who had been exposed of 15 to 20 mSv was *243, and this number showed a slight* in*crease from 216 of the previous year.*

Table Collective Dose, Average Individual Dose and Number of Workers at Commercial Nuclear Installations

Fiscal year	2000	2001	2002	2003	*2004*	*2005*	*2006*
Collective Dose (man-Sv)	78.83	78.05	84.03	96.41	*77.86*	*66.91*	*67.43*
Average annual individual dose (mSv)	1.2	1.2	1.3	1.4	*1.2*	*1.0*	*1.0*
Total number of workers	65,900	67,800	63,800	66,600	*66,700*	*66,300*	*66,900*

b. Performance of Collective Dose at commercial nuclear installation

In Japan at the end of June, 2007, total of 55 units, namely 32 BWRs and 23 PWRs were operating.

The collective doses per reactor year have been slightly decreasing in recent years. The data for operating BWRs were 1.58 man-Sv in 2004 FY, 1.39 man-Sv in 2005 FY, and 1.33 man-Sv in 2006 FY respectively. The data for operating PWRs were 1.25 man-Sv in 2004 FY, 0.97 man-Sv in 2005 FY and 1.08 man-Sv in 2006 FY respectively. The reasons for the decrease were mainly due to decreases of periodic inspection duration and amount of improvement and modification work.

Table Collective Dose per Unit / Reactor-Year

Fiscal year	2000	2001	2002	2003	*2004*	*2005*	*2006*
BWR (man-Sv)	1.96	1.68	2.10	2.40	*1.58*	1.39	*1.33*
PWR (man-Sv)	1.03	1.27	1.00	1.07	*1.25*	0.97	*1.08*

Trend of Collective Dose per Unit / Reactor-Year is shown below.

Figure Collective Dose per Unit / Reactor-Year

3) Dose Limits for the Public

The dose limits for the public are also given in the Dose Limit Notification as listed in Table 15-2.

Table 15-2 Dose limits for the public

Items	Limit
Dose limits outside the peripheral monitoring area	
Effective dose	1 mSv/ year
Equivalent dose for eye lens	15 mSv/year
Equivalent dose for skin	50 mSv/ year

(2) Conditions for Discharge of Radioactive Materials

1) Dose Target and Discharge Control to Reduce Dose of the Public in the Vicinity (ALARA)

In the Regulatory Guide for the Annual Dose Target for the Public in the Vicinity of Light Water Nuclear Power Reactor Facilities, the NSC has prescribed a numerical guide of 0.05 mSv, one twentieth of the dose limit to the public, in order to reduce the dose for the public due to discharge of radioactive material to the environment during normal operation of a nuclear installation as low as reasonably achievable.

The licensee, in order to achieve the target, establishes an annual numerical discharge control guide, which corresponds to the annual discharge amount evaluated at the safety review and assessment, and makes the effort to keep the discharge of radioactive effluents below the numerical discharge control guide. NISA acknowledges the numerical discharge control guide and receives the report from the licensee.

2) The Discharge Data and the Measures Taken to Reduce the Amount of the Discharge

The discharge records of radioactive gaseous and liquid waste from the nuclear installations (BWR and PWR) over the past seven years are shown in Tables 15-3 to 15-5. The tables clearly show that the discharge quantities are substantially below the numerical discharge control guide, the noble gas discharge from the PWR being only one 1000th of the dose target. This is due to the fact that the licensees have carried out the radiation management of the nuclear installation in line with the ALARA principle, including the following measures.

Gaseous waste is discharged from the ventilation port, while being measured and monitored, after particles are removed by a high efficiency filter, noble gas and iodine are decayed in a holdup tank or activated carbon type noble gas hold-up device.

All liquid waste is collected in a disposal facility, and the equipment drain is recovered after being processed in an equipment filter or demineralizer. The floor drain is recovered after being processed in a concentrator and demineralizer. Floor drain is reused in general, though part of it may be discharged through the discharge outlet after the concentration is measured. The recovered liquid waste from the resin is reused after being treated in a concentrator and demineralizer. Concentrated liquid waste generated in this process is treated as solid waste. Low-level laundry wastewater, etc. are usually drained into the environment after being treated through a filter and then it is monitored.

In addition to the measures shown in the paragraphs before, a very low level of gaseous discharge and liquid radioactive waste were the results of the following efforts, the substantial reduction of the possibility of a fuel leak by the improvement of fuels, (so only three cases with four fuel assemblies of fuel leak arose during the period of reporting), filtering ventilation during periodic inspections through local high efficiency filter.

(3) Environmental Radiation Monitoring

The licensee is required to install environmental radiation monitoring equipment during the normal operation of a nuclear installation. This equipment includes monitoring devices of the dose inside the radiation control area and outside the peripheral monitoring area and automatic devices to alarm any abnormal increase of concentration of radioactive materials or dose rates.

The licensee conducts radiation monitoring at the site vicinity during normal operation, assesses the impact upon the environment of the discharge of radioactive materials from the nuclear installation, and feedbacks the results in improving discharge control and facility management. Local governments (prefectures where nuclear installations are located) also monitor the radiation level independently at the site vicinity to protect public health and safety.

The NSC decided the fundamentals of planning and implementation of the monitoring and the evaluation of radiation dose in the Guide for Environmental Radiation Monitoring, in order to improve and to standardize the monitoring technology. Local governments and licensees implement monitoring in accordance with this guide.

15.3 Regulatory Control Activities

(1) Discharge Control of Radioactive Materials

By the Rules for Commercial Power Reactors, the licensee is required to report immediately to NISA when a concentration of radioactive materials in the air outside the peripheral monitoring area exceeds the allowable limit in discharging gaseous radioactive waste, or when the concentration of radioactive materials in the water at the outer boundary of the peripheral monitoring area exceeds the allowable limit in discharging liquid radioactive waste, and report the status of the event and measures taken against it within ten days.

(2) Control of Personal Exposure

The Rules for Prevention of Hazards from Ionizing Radiation require the licensee to measure the dose due to external and internal exposure of workers who are engaged in radiation work or in emergency work, or enter temporal access into the radiation control area. The rules require that the licensee monitor and check daily the dose due to external exposure, if it is expected to exceed the specified value of 1 mSv at 1 cm dose equivalent, and to calculate, without delay, the dose of the personnel engaged in radiation work using the method prescribed by the Minister of Health and Labor, and to keep these records for a period of thirty years.

The Radiation Workers' Registration Center of the Association of Radiation Impact was established in November 1977, to address the difficulty of controlling the personal dose of each worker who works in more than one radiation environment. The Center unitarily collects and controls such personal radiation control data of each worker who works under the Reactor Regulation Law, with personal identification control, a personal radiation control booklet, periodical dose registration and transfer and custody of personal radiation dose record.

(3) Control of Collective Dose

The trend of collective dose in Japan after 1990s has generally continued to be flat or a gradual increase, while that of Western countries has gradually decreased. Though the decreasing trend is observed after 2004 with the decrease of modification work, the trend through the whole period still remains at a higher level as compared with that of Western countries.

In view of the recent situation, a study of radiation exposure at nuclear power plants in both Japan and Western countries was carried out and the effort for dose reduction based on the ALARA principle were also investigated in order to clarify the reasons of difference between Japan and Western countries and also to identify the issues for dose reduction (from 2004 Fy to 2007 Fy).

As the results and reasons for differences of the collective dose between Japan and Western countries were clarified, differences in workload during plant outage (amount of construction/modification work and the number of workers), length of operation cycle and maintenance criteria, e.g. a long term operation of 18 to 24 months is permitted at some plants in Western countries and 13 months operation is permitted as the maximum in Japan, and the plant outage period is about 2 to 3 months in Japan, which is about two times longer than that of Western countries. And, the results of investigation on the efforts for dose reduction based on ALARA principle identified the subjects for the optimization of collective dose management, such as medium and long term dose reduction strategy, information sharing between licensees and for ALARA, way of regulatory intervention to the efforts of licensees etc..

Though at present the collective dose level itself (about 1.1 man-Sv) is not a matter of concern, at present the individual dose (average annual dose of recent years: about 1 mSv, maximum individual dose: about 20 mSv) is below the dose limits (100 mSv / 5 years and 50 mSv / year), it is important to promote activities for collective dose reduction continuously based on the ALARA principle.

The regulatory body will precede the study of solid control measures of collective dose, such as development of the diagnostic evaluation method focused on the dose reduction processes in order to stimulate the licensees' activities.

Table 15-3 Annual discharge of radioactive noble gas in gaseous waste

(unit: Bq / year)

Year / Station	1997	1998	1999	2000	2001	2002	2003	*2004*	*2005*	*2006*	*Numerical Discharge Control Guides*
Station - A	N.D.*	N.D.	N.D.	N.D.	N.D.	N.D.	N.D.	*N.D*	*N.D*	*N.D*	6.7×10^{15}
Station - B	4.3×10^{11}	6.1×10^{11}	1.2×10^{11}	5.7×10^{10}	1.5×10^{10}	2.8×10^{10}	1.8×10^{11}	4.1×10^{11}	6.2×10^{9}	2.9×10^{9}	3.7×10^{15}

* N.D. indicates a value below the detection limit concentration of 2 x 10^{-2} Bq/cm^3.

Table 15-4 Annual discharge of radioactive iodine (I-131) in gaseous waste

(unit: Bq / year)

Year / Station	1997	1998	1999	2000	2001	2002	2003	*2004*	*2005*	*2006*	*Numerical Discharge Control Guides*
Station - A	N.D.*	N.D.	N.D.	N.D.	N.D.	N.D.	N.D.	*N.D*	*N.D*	*N.D*	2.3×10^{11}
Station - B	8.6×10^{5}	1.2×10^{5}	1.6×10^{5}	1.1×10^{6}	2.7×10^{5}	N.D	N.D	1.9×10^{8}	*N.D*	*N.D*	1.0×10^{11}

*: N.D. indicates a value below the detection limit concentration of 7 x 10^{-9} Bq/cm^3.

Table 15-5 Annual discharge of radioactive materials (excluding ^{3}H) in liquid waste

(unit: Bq / year)

Year / Station	1997	1998	1999	2000	2001	2002	2003	*2004*	*2005*	*2006*	*Numerical Discharge Control Guides*
Station - A	N.D.*	N.D.	N.D.	N.D.	N.D.	N.D.	N.D.	*N.D.*	*N.D.*	*N.D*	2.5×10^{11}
Station - B	N.D.	N.D.	N.D.	N.D.	N.D.	N.D.	N.D.	*N.D.*	*N.D.*	*N.D*	1.4×10^{11}

*: N.D. indicates a value below the detection limit concentration of 2 x 10^{-2} Bq/cm^3.

(Represented by ^{60}Co)

(Note) Station - A: Kashiwazaki-Kariwa NPS (BWR), Station - B: Ohi PS (PWR)

Article 16 Emergency Preparedness

1. Each Contracting Party shall take the appropriate steps to ensure that there are on-site and off-site emergency plans that are routinely tested for nuclear installations and cover the activities to be carried out in the event of an emergency. For any new nuclear installation, such plans shall be prepared and tested before it commences operation above a low power level agreed by the regulatory body.

2. Each Contracting Party shall take the appropriate steps to ensure that, insofar as they are likely to be affected by a radiological emergency, its own population and the competent authorities of the States in the vicinity of the nuclear installation are provided with appropriate information for emergency planning and response.

3. Contracting Parties which do not have a nuclear installation in their territory, insofar as they are likely to be affected in the event of a radiological emergency at a nuclear installation in the vicinity, shall take the appropriate steps for the preparation and testing of emergency plans for their territory that cover the activities to be carried out in the event of such an emergency.

On emergency preparedness, the Special Law of Emergency Preparedness for Nuclear Disaster (hereinafter referred to as the “Special Law for Nuclear Emergency”) was enacted in December 1999, incorporating the lessons learned from the JCO criticality accident.

Considering the special characteristics of a nuclear emergency, measures for nuclear emergency preparedness have been defined to cope within the existing legal framework established by the Basic Law on Emergency Preparedness, which has defined such preparedness for disasters as earthquakes, typhoons, and conflagrations etc.

In May 2007, the Nuclear Safety Commission (hereinafter referred to as the NSC) revised the “Emergency Preparedness for Nuclear Installations” (hereinafter referred to as the “Emergency Preparedness Guides”), which specified technical and special matters of nuclear emergency measures, based on international trends such as safety requirements GS-R-2 published by the IAEA.

In addition, as the Special Law for Nuclear Emergency provides that its enforcement situation is subject to review five years after its enforcement, the enforcement situation was investigated by MEXT and METI. The investigation results were reported to the Special Committee on Nuclear Disaster, the NSC.

Moreover, during the reporting period, the so called "Civil Protection Law" to protect the people in armed attack situations etc. was enforced in September 2004. Since nuclear installations are also included in the target-of-attack facilities, some training incorporating the provisions of the law were conducted during some nuclear emergency exercises.

16.1. Development of Laws and Rules for Nuclear Emergency Preparedness

For Japan who was promoting the utilization of nuclear energy under the basic premise of securing safety, the JCO criticality accident which occurred in September 1999 was the first serous accident of its kind, and it was so serious that local residents were instructed for sheltering or evacuation. Lessons learned from this accident clarified the special characteristics of a nuclear emergency, which would require quick initial responses, close coordinated cooperation between the national government and local governments, strengthening of the national emergency system and the clarification of licensee's responsibilities. The Special Law for Nuclear Emergency was enacted in December 1999 and it was enforced in June 2000, addressing the special characteristics of nuclear emergencies as mentioned above. The Special Law for Nuclear Emergency was enacted so as to harmonize with the existing legal framework established by the Basic Law on Emergency Preparedness, which had defined the roles of the national government, local governments, etc. in emergencies such as earthquakes, typhoons, and conflagrations.

The “Nuclear Emergency Preparedness” in the Basic Plan for Emergency Preparedness based on the Basic Law on Emergency Preparedness, was extensively revised in accordance with the Special Law for Nuclear Emergency, clarifying roles and responsibilities of the national government, local governments, and licensees etc. The NSC, in May 2000, also taking into consideration of the Special Law for Nuclear Emergency and the lessons learned from the JCO criticality accident, revised the "Emergency Preparedness Guides" on technical and special matters of nuclear emergency measures, to include:

- Research reactors and nuclear fuel cycle facilities in addition to commercial power reactors; and,
- Accidental release of nuclear fuel material during transport of nuclear fuel, etc. in addition to release of noble gas and iodine from NPS etc.

After that, the Emergency Preparedness Guides have been enhanced through the following multiple revisions by the NSC:

- In March 2001, the dose coefficient (Sv/Bq) for internal exposure was changed along with the term, in response to the amendment of the relevant legislations such as the Reactor Regulation Law etc. based on the adoption of the 1990 Recommendation of the ICRP;
- In June 2001, provisions of the emergency exposure medical treatment for exposed patients was revised to be more effective and responsibilities of the national and local governments and nuclear licensees were clarified based on the experience of the criticality accident;
- In April 2002, protective measures concerning the taking of stable iodine tablets as a prevention were established based on the scientific knowledge acquired from the long-term follow-up survey to atomic bomb sufferers and the investigation results of the Chernobyl Power Station accident, etc;
- In November 2002, measures for mental health care in a nuclear emergency were established based on the experience of JCO criticality accident, experiences of natural disasters such as seismic disasters, etc; and,
- In July 2003, the designation of a regional emergency exposure medical treatment system was established.

In May 2007, the emergency measures were reviewed with reference to the IAEA Safety Requirement GS-R-2, "Preparedness and Response for a Nuclear or Radiological Emergency", Safety Guide GS-G-2.1, "Arrangements for Preparedness for a Nuclear or Radiological Emergency" (hereinafter referred to as the "IAEA documents"), etc., and the following six items were revised:

- *Characterization of the Emergency Preparedness Guides was clarified as "guides on technical and special matters specified by the NSC for the national and local governments and nuclear licensees in preparing plans related to nuclear emergencies and in taking protective measures during an emergency;"*

- *Situations for which the Emergency Preparedness Guides are applied were clarified as "nuclear emergencies at reactor facilities except the reactors for nuclear ship, fuel fabrication facilities, processing facilities, utilization facilities (limited to facilities that use nuclear fuel material equal to or exceeding the critical mass), waste disposal facilities and waste storage facilities, and transportation of nuclear fuel materials etc;"*

- *The following four goals of protective measures were clarified referring to the IAEA documents, and it was also clarified that in taking these protective measures it is important to assess them with the principles of "Justification of intervention" and "Optimization of intervention";*

  1) *To prevent the occurrence of deterministic health effects in residents in the vicinity, to nuclear-installation workers, in those relevant in emergency preparedness, etc.,*

  2) *To render first aid and to manage the treatment of radiation injuries,*

  3) *To prevent, to the extent practicable, the occurrence of stochastic health effects in the population, and,*

  4) *Reducing anxiety on the health of residents in the vicinity, workers, and to those relevant in emergency preparedness.*

- *The IAEA documents specify the precautionary action zone (PAZ) and the urgent protective action planning zone (UPZ) as off-site emergency zones for which arrangements shall be made for taking urgent protective action. In the results of the specialists' study, it was clearly written that this was also effective, in the emergency measures of Japan, to implement precautionary/protective measures before or immediately after the release of radioactive materials instead of setting up a new specific zone as PAZ. It was confirmed that the setting up of UPZ was for the same purpose as EPZ which had already been set up in the Emergency Preparedness Guides of Japan as an area where protective measures should be focused on in the implementation*;

- *The IAEA documents provide guidelines for the protective measures corresponding to avertable doses. In Japan, projected doses are used when implementing protective measures. It was confirmed that making judgment using projected doses rather than avertable doses as guidelines for the protective measures served as a response on the safe side; and,*

- *It was clearly written that the effect of taking stable iodine tablets as prevention and a protective measure is appropriate only for the internal exposure by radioactive iodine and the measure will complement protective measures, such as sheltering and evacuation.*

As the Special Law for Nuclear Emergency provides that its enforcement situation is subject to review five years after its enforcement, the enforcement situation was investigated by MEXT and METI. Results of the investigation were reported to the Special Committee on Nuclear Disaster, the NSC in March 2006.

The Nuclear and Industrial Safety Agency (NISA) checked the enforcement situation concerning four issues that were presupposed to respect when the Special Law for Nuclear Emergency was enacted, and reported the following:

- *Concerning the speeding up of the initial response, non-scenario-based training should be carried out, and the effort should be continued;*

- *Concerning enhancing the cooperation among the national government and local governments, the "Integrated Nuclear Emergency Preparedness Network", which is a large-scale system and preparation of a fast unified network of communication among them, should be made;*

- *Concerning enhancing the emergency response system of the national government, necessary renewal of materials and equipment of the Emergency Preparedness Center should be promoted; and,*

- *In relation to clarification of the licensees' duties, the effectiveness should be verified and improved so that nuclear emergency specialists may achieve their required functions in an emergency.*

16.2. Nuclear Emergency Preparedness and the Measures

The Special Law for Nuclear Emergency has prescribed measures in a nuclear emergency at power reactors, research reactors, nuclear fuel cycle facilities, etc. Emergency measures of commercial nuclear installations are given below.

(1) Responsibilities of Related organizations concerning Nuclear Emergency Preparedness

1) Responsibility of the National Government

The national government prepares the necessary emergency preparedness and is ready to take measures in an emergency:

- METI stations a Senior Specialist for Nuclear Emergency in the vicinity of each nuclear installation, who guides and advises the licensee in preparing his emergency action plan and, in an emergency, takes the necessary measures in preventing expansion of the emergency;

- The NSC is mandated to give technical advice to the chief of the Nuclear Emergency

Response Headquarters (Prime Minister) on designation or alteration of regional areas that necessitates emergency measures to be taken, and technical matters on the implementation of emergency response measures and dissolution of a nuclear emergency. For that purpose, the NSC organizes the “Technical Advisory Organization in an Emergency” which consists of the NSC Commissioners and the Investigators for Emergency Response;

- The Minister of METI designates a facility in the vicinity of a nuclear installation as an Off-Site Center to be used in an emergency. In the case of an emergency, the national government, the local governments and the licensee establish at the Off-Site Center the "Joint Council for Nuclear Emergency Response", in order to share information and to coordinate their activities. Off-Site Centers are built on the areas indicated in Fig. 16-1, and have necessary facilities and equipment capable to communicate with the Prime Minister’s Official Residence, the Cabinet Office, the Emergency Response Center of NISA, the Emergency and Emergency Preparedness Center of MEXT and related local governments;

- Each Off-Site Center is equipped with means by which the related organizations monitor environmental radiation levels and the status of the nuclear installation. The environmental radiation levels, other than temporary data measured in an emergency, can be monitored at every moment since the monitoring equipment is connected on line with the monitoring posts located in the vicinity of the nuclear installation. The on-line status of the nuclear installation that is sent from the licensee in an emergency can be displayed on the monitor panels. The results of estimation are also displayed by means of an Emergency Response Support System (ERSS), which forecasts progress of an abnormal condition of the nuclear installation using plant information.

- The national government establishes arrangements to secure quick and coordinated activities in an emergency; and,

- The national government conducts the comprehensive emergency drill based on the program established by the competent minister.

2) Responsibilities of local governments

Each local government shall develop and revise the regional emergency preparedness plan in accordance with Article 40 of the Basic Law on Emergency Preparedness, and shall consult beforehand with the Prime Minister for its development or revision.

3) Responsibilities of licensees:

- Each licensee shall develop his Nuclear Licensee Emergency Action Plan after consulting with relevant local governments, and submit it to the Minister of METI before the commissioning of the reactor;

- Each licensee shall establish an on-site organization for nuclear emergency preparedness, and designate a Nuclear Emergency Preparedness Manager who administers the organization; and

- The Nuclear Emergency Preparedness Manager shall notify specific initial events to the competent authorities.

(2) Measures for On-site and Off-site Nuclear Emergency Preparedness of Nuclear Installations

In order to prepare the "Nuclear Emergency Preparedness" described in paragraph (3), related organizations keep themselves ready to collect and send information and also to start a quick response against an emergency, conduct emergency drills, disseminate knowledge and promote research on emergency preparedness. Outline of roles and responsibilities of related organizations are as follows.

1) On-Site Emergency Preparedness of Nuclear Installations

When the licensee detects abnormal release of radioactive material or an abnormal level of radiation at a nuclear installation, he takes necessary measures to prevent progression of the event into an emergency.

The licensee, to cope with the emergency properly, prepares the Nuclear Licensee Emergency Action Plan after consulting with related local governments, which provides for the prevention of, emergency measures against, and post-emergency restoration of a nuclear emergency, including on-site and off-site cooperation with other organizations. Especially, quick and accurate notification of occurrence of specific initial events to related organizations is a very important obligation of the licensee.

The licensee is required to take part in comprehensive drills with related organizations, and keep close contact with them.

2) Emergency Preparedness in the Vicinity of Nuclear Installations

Roles and responsibilities of the national government and local governments in emergency preparedness in the vicinity of nuclear installations are defined in the Special Law for Nuclear Emergency and the Basic Plan for Emergency Preparedness. Each local government develops its own regional emergency preparedness plan. They carry out emergency environmental radiation monitoring, and implement evacuation or sheltering of residents receiving advice or direction from the Prime Minister based on the report of the Minister of METI. The taking of stable iodine tablets for prevention, as well as sheltering or evacuation, are defined as some of the protective measures.

(3) Nuclear Emergency Preparedness concerning Nuclear Installations (Fig. 16-2)

Quick initial response and closely coordinated cooperation among relevant organizations are important in a nuclear emergency:

- The Special Law for Nuclear Emergency defines specific initial events in a nuclear installation (see Table 16-1), the occurrence of which the licensee shall immediately notify the Minister of METI and the heads of related local governments;

- The Minister of METI, receiving the notification, triggers activities according to the procedure stipulated by law. Staff with expertise in emergency measures will be sent to local governments on request. The Senior Specialist for Nuclear Emergency collects information and coordinates activities preventing expansion of the events;

- When the Minister of METI recognizes that the specific initial event exceeds the predetermined level and has developed into an emergency, the Minister immediately reports it to the Prime Minister;

- The Prime Minister has powerful authority to declare a "Nuclear Emergency", and to advise or direct relevant local governments on necessary measures such as sheltering or evacuation to be taken by them, as well as to request for dispatch of the Self-Defense Forces concerning implementation of emergency measures;

- The Prime Minister establishes the "Nuclear Emergency Response Headquarters" in Tokyo, which he will head, and the "Local Nuclear Emergency Response Headquarters" at the site;

- In a nuclear emergency, the NSC convenes with the "Technical Advisory Organization in an Emergency" that is composed of commissioners and the Advisors for Emergency Response. The Organization gives technical advice to the Prime Minister;

- Local governments establish their own emergency response headquarters; and,

- In order to share information between the national government and related organizations such as local governments, nuclear licensees, etc., and, if necessary, to coordinate emergency measures to be implemented by the respective organizations, the "Joint Council for Nuclear Emergency Response" is to be established at the Off-Site Center.

16.3. Implementation of Nuclear Emergency Drill

The emergency preparedness action plan in accordance with the Basic Law on Emergency Preparedness, and the Off-Site Center in the vicinity on the nuclear installation provided in the previous section has been established for each nuclear installation, and a nuclear emergency drill is implemented to confirm the effectiveness of these measures. The purpose of the nuclear emergency drill is 1) to enhance understanding of, and adequate actions for, nuclear emergency preparedness by responsible personnel of the national government, local governments, the licensee, and residents, and 2) to verify whether emergency measures function in a predetermined way, and whether information sharing and cooperation among related organizations are adequate. The national government, local governments, designated public organizations and the licensee cooperate and participate in the drill, which cover communication, monitoring, decision on emergency measures to be taken, sheltering or evacuation etc., ranging from a large scale national drill to the licensee's on-site drill. Drills implemented in past years are shown below.

(1) Drills Planned by the National Government (Table 16-2 (1))

Nuclear emergency drills used to be planned and conducted by local governments with support and coordination of the national government before the JCO criticality accident. The Special Law for Nuclear Emergency stipulated the drills to be planned and conducted by the national government.

Drills including accident management activities assuming a scenario resulting in core damage have been implemented in the national emergency drills.

The drill planned by the national government has been conducted once a year as the comprehensive nuclear emergency drill in collaboration with the national government, local governments, licensees, etc.

Drills implemented *during the reporting period* are as follows:

As for 2004, although a drill for the Kashiwazaki Kariwa Nuclear Power Station, Tokyo Electric Power Co., Inc. was scheduled on November 1 and 2, it was canceled due to the impact of the Niigata-ken Chuetsu earthquake which occurred just before conducting the drill.

As for 2005, the drill for Unit 4 of the Kashiwazaki Kariwa Nuclear Power Station (located in Kashiwazaki City and Kariwa Village, Niigata Prefecture) which had been canceled in 2004 was conducted on November 9 and 10 in collaboration with the national government, the local governments of Niigata Prefecture and relevant municipalities, Tokyo Electric Power Co., Inc., and organizations related to the emergency preparedness. About 2,600 persons including about 400 local residents participated in the drill.

On October 25 and 26, 2006, a drill for Unit 3 of the Ikata Power Station, Shikoku Electric Power Co., Inc. (located in Ikata Town, Ehime Prefecture) was conducted in collaboration with the national government, local governments of Ehime Prefecture and relevant municipalities, Shikoku Electric Power Co., Inc., and organizations related to the emergency preparedness. About 3,700 persons including about 300 local residents participated in the drill. In this instance, while verifying urgent dispatch and urgent conveyance to a remote place, speeding up of determining evacuation areas was achieved by verifying actions for emergency measures in an alert stage.

Results of the drills held every year are assessed and reflected to the items and methods of drills to be implemented in and after the following fiscal year. Three kinds of methods, a participant's questionnaire, check by an independent assessment agency, and observation by external experts, are adopted for the assessment.

(2) Drills Planned by the NSC

The NSC is conducting communication drills that aim enhancing the emergency communication system and keeping up and/or improving its functions. The NSC also is conducting field-training drills of a Technical Advisory Organization in an Emergency that aim to confirm the emergency response capability and improve its effectiveness.

(3) Drills Planned by Local Government (Table 16-2 (2))

The regional emergency preparedness plan prescribes the local drills to be planned and conducted by each local government, which NISA and the NSC support by dispatching expert staff. *Drills reflecting the framework of the Civil Protection Law are also conducted in part of those drills.*

(4) Drills Planned by Licensees

Each licensee has implemented an on-site drill once a year including establishment of an emergency response headquarters, notification and communication, emergency environmental radiation monitoring, etc. based on the Nuclear Licensee Emergency Action Plan defined for each place of business.

Each licensee also has implemented a drill taking into consideration the accident management activities, if necessary, in order to comprehensively confirm effectiveness of the organization implementing the accident management.

When the said place of business is subject to the drill conducted by the local government, the on-site drill has been implemented at the same time with the local drill implemented by the local government etc.

16.4 International Framework and Relationship with Neighboring Countries

Japan is a contracting party to the Convention on Early Notification of a Nuclear Accident, and to the Conventions on Assistance in the Case of a Nuclear Accident or Radiological Emergency. The following domestic implementation system has been established for the notification of a nuclear accident to neighboring countries:

- For nuclear installations, the Ministry of Foreign Affairs has been designated as the authority for notification and as the competent authority for foreign accidents, and METI has been designated as the competent authority for domestic accidents;
- METI receives a report immediately upon an accident in a nuclear installation on the basis of legal obligation of the licensee; and,
- When an accident occurrence is confirmed and it is predicted that release of radioactive materials may affect neighboring countries, the IAEA and the countries that may be affected by the accident are notified of the accident.

In accordance with an arrangement aiming at enhancement of the safety level of commercial nuclear installations concluded with the People's Republic of China, on the basis of the bilateral agreement for peaceful use of nuclear energy, the governments should mutually notify without delay of major accidents of nuclear installations. A bilateral inter-governmental agreement with the Republic of Korea calls for cooperation in the establishment and operation of an early notification network for nuclear safety.

If an accident should occur at a foreign nuclear installation and a request for assistance is made, Japan will dispatch, on the basis of the Conventions on Assistance in the Case of a Nuclear Accident or Radiological Emergency, specialists in emergency monitoring and emergency exposure medical treatment, and will provide materials and equipment such as radiation measurement equipment and radiation protection equipment.

In addition, Japan has supported Asian countries to enhance their infrastructures of emergency countermeasures and response for nuclear installations in an operation of the Asian Nuclear Safety Network (ANSN), which is a part of the IAEA cooperation businesses to Asia, and has cooperated in workshops, such as emergency medical treatment and emergency drills.

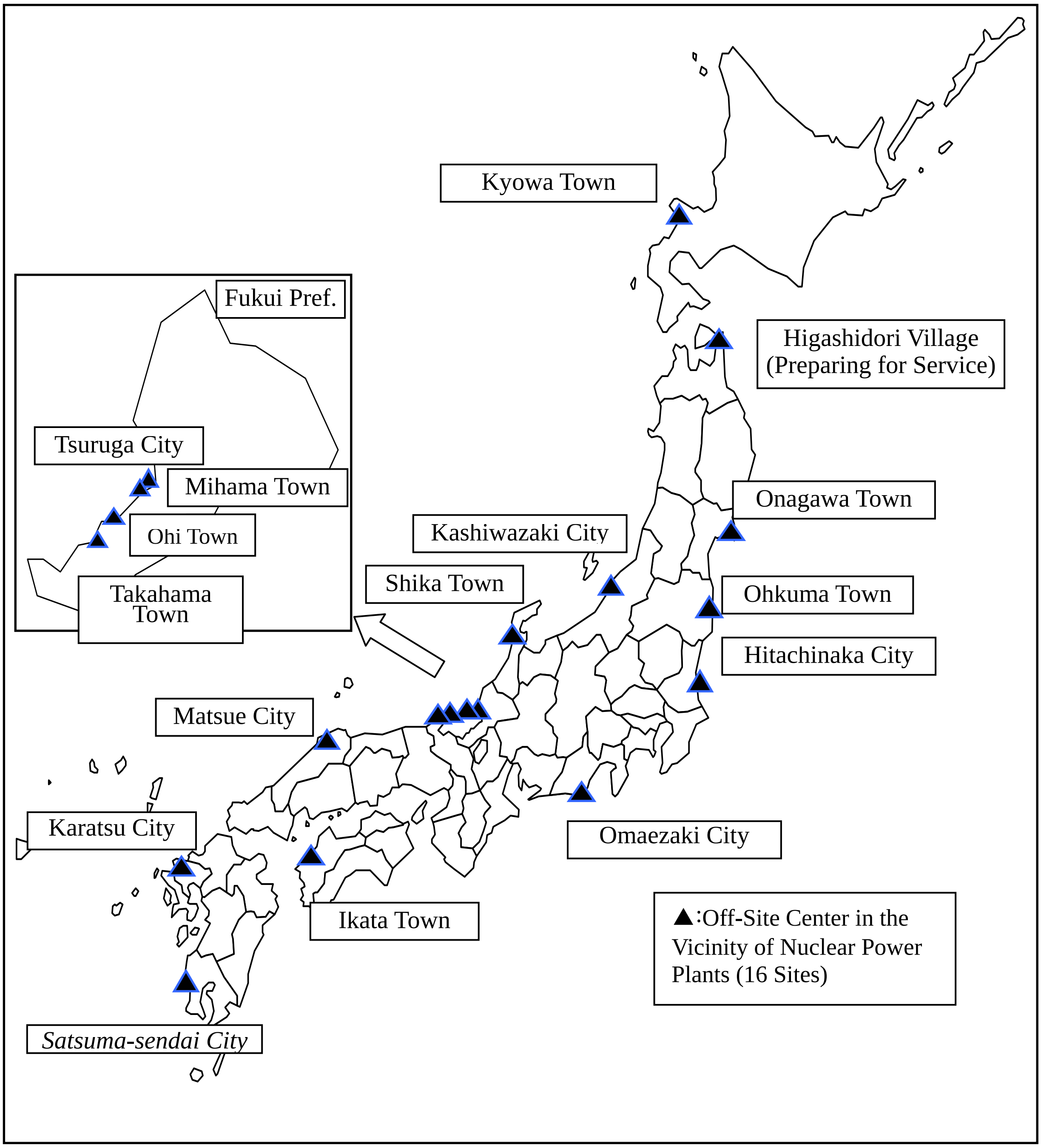

Fig. 16.1 Location of Off-Site Centers

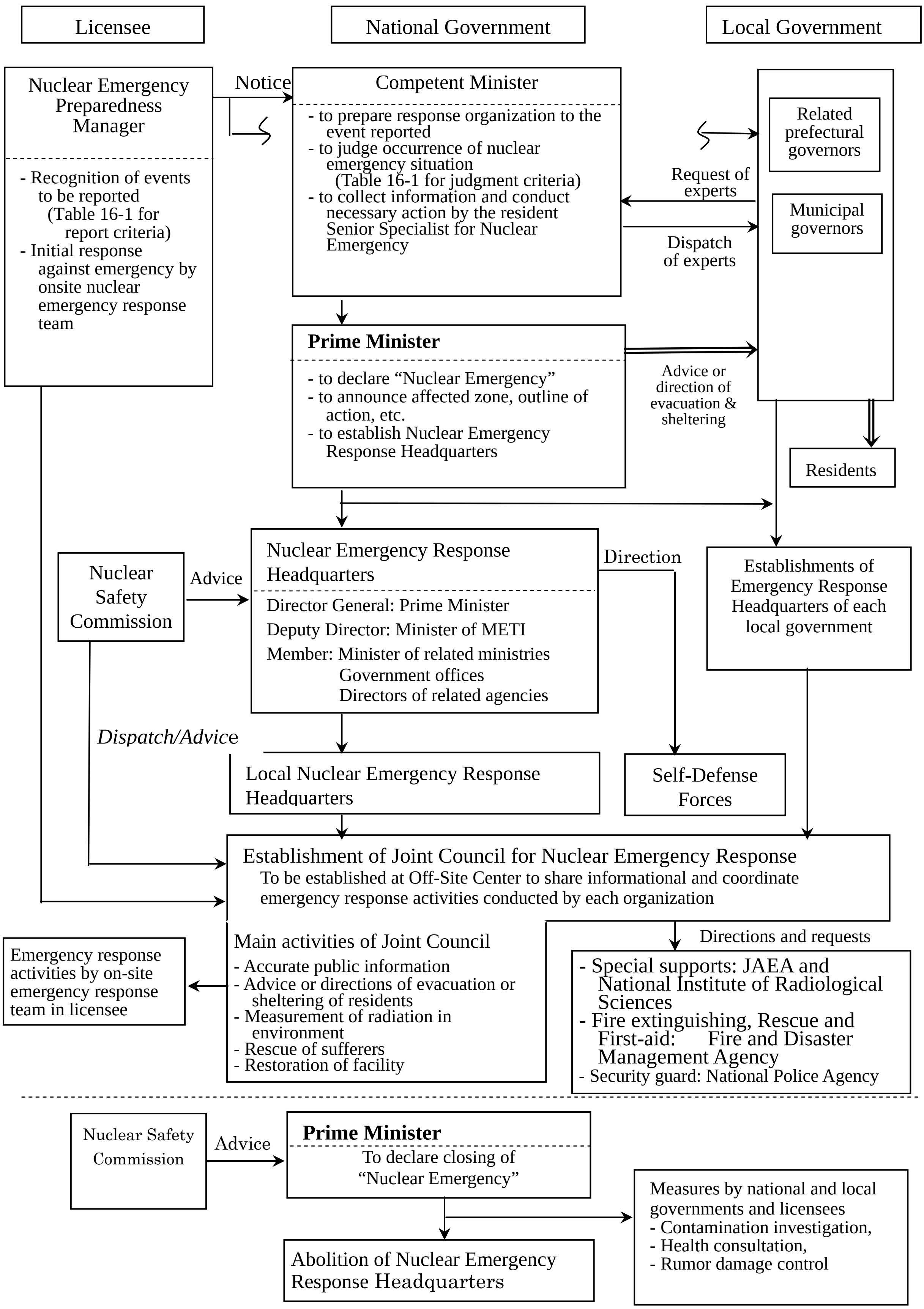

Fig. 16-2 Nuclear Emergency Preparedness based on the Special Law for Nuclear Emergency

Table 16 – 1 Main Specific Events and the Nuclear Emergency specified in the Special Law for Nuclear Emergency

	Events	Criteria for reporting by licensees and "Nuclear Emergency" declaration by national government	conditions of declaration of "Nuclear Emergency"
Events that licensees should report.	a) Dose of radiation near the site boundary detected dose	5 micro Sv/h at one point for more than 10 min 5 micro Sv/h at more than 2 points at the same time	500 micro Sv/h at one point for more than 10 min, 500 micro Sv/h at more than 2 points at the same time
	b) Detection of the radioactive materials in usual release points, such as exhaust pipes	Radioactive materials worth more than 5 micro Sv/h	Radioactive materials worth more than 500 micro Sv/h
	c) Radiation by fire, explosion, etc. or detection of radioactive materials (outside the control zone)	Radiation dose of more than 50 micro Sv/h Release of radioactive materials worth more than 5 micro Sv/h	Radiation dose of more than 5 mSv/h Release of radioactive materials worth more than 500 micro Sv/h
	d) Individual events of each nuclear installation (Example for reactor)		
	- Failure of scram	When the nuclear reactor shutdown cannot be performed by usual neutron absorbers	When all of reactor shutdown functions are lost
	- Loss of reactor coolant	Occurrence of leakage of nuclear reactor coolant which needs operation of the emergency core coolant system (ECCS)	When water cannot be poured to the nuclear reactor by all ECCSs
	- Loss of all AC power supplies	When all AC power supplies stop power supply for more than 5 minutes	When all measures for the cooling reactor core are lost with loss of all AC power supplies.

Response of the national government	- The Minister of METI sends staff with expertise on request of local governments. - The resident Specialist on Nuclear Emergency Preparedness carries out necessary work. The following responses are carried out based on the agreement of related ministries, not specified in the Special Law for Nuclear Emergency. - Related ministries and agencies organize a joint task group for the incidents in Tokyo. - Related local organizations organize a joint local task group in the Off- Site Center.	- The Minister of METI reports the nuclear emergency to the Prime Minister immediately after confirming the situation. - The Prime Minister declares "Nuclear Emergency" and takes the following responses; - to advice or direct related local governments on necessary measures such as sheltering or evacuation. - to establish the Nuclear Emergency Response Headquarters in Tokyo and Local Nuclear Emergency Response Headquarters at Off-Site Center. - to establish the Joint Council for Nuclear Emergency Response for information exchange among the national government and local governments.

Table 16-2 Nuclear Emergency Drills

Conductor	Date	Nuclear Power Station
(1) Drills conducted by the National Government *(April 2004 - March 2007)*		
METI	*11/09/2005 (Wed.) –11/10/2005 (Thu.)*	*Kashiwazaki Kariwa Nuclear Power Station (Tokyo Electric Power Co., Inc.)*
METI	*10/25/2006 (Wed.) –10/26/2006 (Thu.)*	*Ikata Power Station (Shikoku Electric Power Co., Inc.)*
(2) Drills conducted by Local Governments *(April 2004 - March 2007)*		
Shizuoka Pref.	*06/29/2004 (Tue.)*	*Hamaoka Nuclear Power Station (Chubu Electric Power Co., Inc.)*
Shimane Pref.	*10/08/2004 (Fri.)*	*Shimane Nuclear Power Station (The Chugoku Electric Power Co., Inc.)*
Miyagi Pref.	*10/19/2004 (Tue.)*	*Onagawa Nuclear Power Station (Tohoku Electric Power Co., Inc.)*
Hokkaido	*10/22/2004 (Fri.)*	*Tomari Power Station (Hokkaido Electric Power Co., Inc.)*
Ehime Pref.	*10/26/2004 (Tue.)*	*Ikata Power Station (Shikoku Electric Power Co., Inc.)*
Aomori Pref.	*11/16/2004 (Tue.)*	*Higashidori Nuclear Power Station (Tohoku Electric Power Co., Inc.)*
Saga Pref. Nagasaki Pref.	*11/22/2004 (Mon.)*	*Genkai Nuclear Power Station (Kyushu Electric Power Co., Inc.)*
Fukushima Pref.	*11/24/2004 (Wed.)*	*Fukushima Daiichi Nuclear Power Station (Tokyo Electric Power Co., Inc.)*
Kagoshima Pref.	*01/30/2005 (Sun.)*	*Sendai Nuclear Power Station (Kyushu Electric Power Co., Inc.)*
Fukui Pref. Kyoto Pref.	*03/21/2005 (Mon.)*	*Takahama Power Station (The Kansai Electric Power Co., Inc.)*
Ishikawa Pref.	*03/24/2005 (Thu.)*	*Shika Nuclear Power Station (Hokuriku Electric Power Co.)*
Fukui Pref.	*08/02/2005 (Tue.)*	*Mihama Power Station (The Kansai Electric Power Co., Inc.)*
Aomori Pref.	*08/10/2005 (Tue.)*	*Higashidori Nuclear Power Station (Tohoku Electric Power Co., Inc.)*
Hokkaido	*10/21/2005 (Fri.)*	*Tomari Power Station (Hokkaido Electric Power Co., Inc.)*
Ehime Pref.	*10/21/2005 (Fri.)*	*Ikata Power Station (Shikoku Electric Power Co., Inc.)*
Miyagi Pref.	*10/28/2004 (Fri.)*	*Onagawa Nuclear Power Station (Tohoku Electric Power Co., Inc.)*
Niigata Pref.	*11/09/2005 (Wed.) –11/10/2005 (Thu.)*	*Kashiwazaki Kariwa Nuclear Power Station (Tokyo Electric Power Co., Inc.)*
Fukushima Pref.	*11/14/2005 (Mon.) –11/15/2005 (Tue.)*	*Fukushima Daini Nuclear Power Station (Tokyo Electric Power Co., Inc.)*
Ishikawa Pref.	*11/17/2005 (Thu.)*	*Shika Nuclear Power Station (Hokuriku Electric Power Co.)*

Shimane Pref.	*11/17/2005 (Thu.)*	*Shimane Nuclear Power Station (The Chugoku Electric Power Co., Inc.)*
Kagoshima Pref.	*11/19/2005 (Sat.)*	*Sendai Nuclear Power Station (Kyushu Electric Power Co., Inc.)*
Saga Pref.	*11/21/2005 (Mon.)*	*Genkai Nuclear Power Station (Kyushu Electric Power Co., Inc.)*
Fukui Pref.	*11/27/2005 (Sun.)*	*Mihama Power Station (The Kansai Electric Power Co., Inc.)*
Shizuoka Pref.	*02/15/2006 (Wed.)*	*Hamaoka Nuclear Power Station (Chubu Electric Power Co., Inc.)*
Fukui Pref.	*08/01/2006 (Tue.)*	*Ohi Power Station (The Kansai Electric Power Co., Inc.)*
Ishikawa Pref.	*08/20/2006 (Sun.)*	*Shika Nuclear Power Station (Hokuriku Electric Power Co.)*
Ibaraki Pref.	*09/29/2006 (Fri.)*	*Tokai No.2 Power Station (The Japan Atomic Power Co.)*
Miyagi Pref.	*10/23/2006 (Mon.) –10/24/2006 (Tue.)*	*Onagawa Nuclear Power Station (Tohoku Electric Power Co., Inc.)*
Hokkaido	*10/30/2006 (Mon.)*	*Tomari Power Station (Hokkaido Electric Power Co., Inc.)*
Niigata Pref.	*11/10/2006 (Fri.)*	*Kashiwazaki Kariwa Nuclear Power Station (Tokyo Electric Power Co., Inc.)*
Aomori Pref.	*11/14/2006 (Tue.)*	*Higashidori Nuclear Power Station (Tohoku Electric Power Co., Inc.)*
Kagoshima Pref.	*11/17/2006 (Fri.)*	*Sendai Nuclear Power Station (Kyushu Electric Power Co., Inc.)*
Fukui Pref.	*11/19/2006 (Sun.)*	*Ohi Power Station (The Kansai Electric Power Co., Inc.)*
Saga Pref.	*11/26/2006 (Sun.)*	*Genkai Nuclear Power Station (Kyushu Electric Power Co., Inc.)*
Shimane Pref.	*01/30/2007 (Tue.)*	*Shimane Nuclear Power Station (The Chugoku Electric Power Co., Inc.)*
Shizuoka Pref.	*02/01/2007 (Thu.)*	*Hamaoka Nuclear Power Station (Chubu Electric Power Co., Inc.)*
Fukushima Pref.	*02/06/2007 (Tue.) –02/07/2007 (Wed.)*	*Fukushima Daiichi Nuclear Power Station (Tokyo Electric Power Co., Inc.)*

D. Safety of Installations

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Article 17 Siting

Each Contracting Party shall take the appropriate steps to ensure that appropriate procedures are established and implemented:

(i) **for evaluating all relevant site-related factors likely to affect the safety of a nuclear installation for its projected lifetime;**

(ii) **for evaluating the likely safety impact of a proposed nuclear installation on individuals, society and the environment;**

(iii) **for re-evaluating as necessary all relevant factors referred to in sub-paragraphs (i) and (ii) so as to ensure the continued safety acceptability of the nuclear installation;**

(iv) **for consulting Contracting Parties in the vicinity of a proposed nuclear installation, insofar as they are likely to be affected by that installation and, upon request providing the necessary information to such Contracting Parties, in order to enable them to evaluate and make their own assessment of the likely safety impact on their own territory of the nuclear installation.**

In Japan, in order to judge the suitability of the site for a nuclear installation, it is deemed necessary to conduct the safety impact assessment of natural phenomena and human induced external events to the nuclear installation, the safety impact assessment on the postulated accident of the nuclear installation to the general public in the vicinity, and the impact assessments on the siting of the nuclear installation to the environment other than the safety, therefore, required legislations and regulations for implementing relevant assessments has been provided, and the assessments are being conducted.

17.1 Basic Concept on the Siting of Nuclear Installations

The following assessments must be taken into consideration when deciding upon the siting of nuclear installations, and are incorporated in the relevant legislation, etc.

- Safety impact assessment on a nuclear installation by natural phenomena and postulated human induced external events

- Safety impact assessment of the radioactive impact to the environment by a nuclear installation should reactor accidents occur

- Assessment on environmental impact due to the siting of a nuclear installation

17.2 Principal Assessment System Concerning the Siting of Commercial Power Reactors

The Reactor Regulation Law requires that location of a Commercial Power Reactor must be selected and its structure and equipment must be designed so that the radiological hazards can be prevented. The adequacy of siting is examined in accordance with the Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and Application Criteria (hereinafter called, the Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and application Criteria) etc. as part of safety examination of licensing for establishment.

The Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and application Criteria requires that no such event that might induce serious accidents has occurred in the past or could be expected to occur in the future at the proposed site and furthermore, there should not be events that may aggravate accidents, the reactors are located at a sufficient distance away from the public in terms of safety and protection facilities, and *the site and the vicinity are in suitable circumstances to take, if needed, measures protecting the public.*

When deciding a site, an adequate attention in design shall be paid to the events caused by external factor specific to the site, in addition to the site conditions stipulated by the Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and application Criteria.

In this respect, the Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities (hereinafter called as the "Regulatory Guide for Reviewing Safety Design") prescribes that structures, systems and components with safety functions shall be designed to sufficiently withstand appropriate design basis earthquake forces. As well, they shall be so designed that the safety of the Commercial Power Reactor will not be impaired by other possible natural phenomena than earthquake and also by postulated human induced external events.

The Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and application Criteria also prescribes that the dose to the public shall meet with the application criteria in consideration of the engineered safety features by establishing an non-residential area and low population zone and ensuring sufficient distance from high population zones, when assessing radiation impact to the public in the vicinity imposed by the postulated accidents in Commercial Power Reactor. Meanwhile, the Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities provides events to be evaluated in siting, acceptance criteria and specific conditions, etc. to be used in the analysis.

Environmental impact assessment of all of the power stations including commercial nuclear installation is performed in accordance with the Environmental Impact Assessment Law enforced in June 1999, before when the departmental council decision of MITI (present METI) dated in July 1977 was applied. This subject is described in Section 17.5.

17.3 Evaluation on Events Caused by External Factor

The Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities prescribes that the earthquakes, natural phenomena other than earthquake and human induced external events shall be addressed in the design, being in accordance with the prescription in the Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and application Criteria, stating "no such event that might cause serious accidents has occurred in the past nor could be expected to occur in the future at the proposed site and furthermore, there should not be events that may aggravate accidents".

On the seismic design, it is required that the structures, systems and components (SSCs) with safety functions shall be designed in accordance with seismic classification, *and shall be designed to maintain safety functions.*

The NSC established the Sub-committee on Investigation of the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities in July 2001, for the purpose of making the guide more appropriate with incorporation of the latest knowledge in the seismic safety guide used for a safety review and assessment, and promoted the amendment study of the Guide. The Nuclear Safety Commission revised the Regulatory Guide and related guides in September, 2006. Description on the new "Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities" is provided in Section 18.1.

For the assumed natural phenomena other than earthquake (floods, tsunami, breeze, freezing, snowfall, landslides, etc.), the SSCs with safety functions are required to be designed so that the safety of the nuclear reactor facility will not be failed by any of these natural phenomena. Those SSCs with safety functions of particularly high importance shall be designed to withstand against the most severe conditions of natural phenomena or to withstand against combination of such natural forces and loads induced by an accident.

Moreover, the SSCs with safety functions are required to be so designed that the Commercial Power Reactor should not be impaired by postulated human induced external event (airplane crashes, dam collapse, explosions etc.).

On the consideration on airplane crash accident, a guide is provided to judge whether or not it is necessary to take it into design consideration as "an assumed human induced external event", as well as the standard evaluation method, in "Evaluation Standards of the Probability of Airplane Falling to a Commercial Power Reactor Facility (NISA Regulatory Guide)", which NISA published as a Regulatory Guide in July 2002. Besides, for airplanes, the flight over nuclear installations is prohibited in principle,

Commercial Power Reactors are required to be provided with appropriate measures to prevent illegal access to the reactor by third persons in Japan.

17.4 Evaluation for the Impacts to the Public of Accidents

In order to ensure safety of the public even in the case of the worst accident, the Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and application Criteria prescribes, as a fundamental siting condition, that a Commercial Power Reactor be located with a sufficient distance from the public taking into account the engineered safety features. The conditions for fulfilling this requirement are as follows:

A) The area within a specified distance from a Commercial Power Reactor shall be the non-residential area, and no radiation hazard is imposed on the public in the vicinity outside the non-residential area, even postulating the occurrence of the major accident.

   The major accident is defined in the above Guide to be an accident, occurrence of which is conceivable as a worst scenario from a technical point of view with considering such factors as the conditions at the site vicinity, the characteristics of the reactor and the engineered safety features.

B) The area within a specified distance beyond the non-residential area shall be the low population zone, and no substantial radiation hazard is imposed on the public in the vicinity of the low population zone, even postulating the occurrence of the hypothetical accident.

The hypothetical accident is defined in the above Guideline to be an accident, which exceeds a major accident, and the occurrence of that is not conceivable from a technical point of view. The Guide, for example, hypothesizes that some of engineered safety features in the reactor, which are assumed to be effective in postulating a major accident, do not function and corresponding release of radioactive materials occurs.

C) A site of a nuclear reactor shall be located at a specified distance from high population zones.

The specified distance means a distance where cumulative value of whole-body dose in case of a hypothetical accident shall be small enough to be deemed acceptable based on the collective dose of view.

The application criteria on dose rate are specified in the attachments of the Regulatory Guide for Reviewing Nuclear Reactor Siting Evaluation and application Criteria. The meteorological observation methods, the statistical processing methods of the observed data and the methods for the analysis of the atmospheric diffusion of the released radioactive materials, to be used in the dose assessments, are prescribed in the Regulatory Guide for Meteorological Observation for Safety Analysis of Nuclear Power Reactor Facilities.

17.5 Environmental Impact Assessment

The Environmental Impact Assessment Law was established to ensure business operators, that are undertaking large-scale projects that could have a serious impact on the environment, to conduct an environmental impact assessment properly and reflect the results of the assessment in implementing the project in term of protecting the environment, and also set forth the procedures in conducting the environmental impact assessment. The assessment for commercial power stations including a nuclear installation must be performed in accordance with the provisions of the Environmental Impact Assessment Law and the Electricity Utility Industry Law. All of nuclear installations are subject to assessment regardless of their scale. Figure 17-1 shows an outline of procedures for environmental impact assessment concerning establishment of a commercial nuclear installation.

Business operator, prior to application for reactor establishment, must prepare a Scoping Document presenting information concerning the contents of the project, items to be considered in an environmental impact assessment, method of survey, prediction, and assessment method to be utilized, and must submit it to NISA, as well as to the local governments having jurisdiction over the area deemed likely to be environmentally impacted by the project. NISA examines the Scoping Document taking into consideration the comments submitted by the related prefecture governor(s), as well as the comments of the residents and the views of the business operators regarding such comments, and gives recommendations on the contents of Scoping Document to the business operator, if needed.

Then business operator shall prepare a draft environmental impact statement (draft EIS) after conducting survey, prediction and assessment in consideration with the recommendations received from NISA and establishing the measures for protecting the environment. The draft EIS must be submitted to NISA, as well as to the related local governments. NISA, after examining the draft EIS, taking into account the opinion of the Minister of Environment and the related Governors as well as the comments of the residents and the views of the business operators regarding such comments, and receiving the view of advisers on the environment protection, gives recommendations on the environmental assessment to the business operator if needed. Meanwhile, concerning the items other than those with little environmental impact, business operators shall check and provide the necessary measures for protecting the environment so that the environmental impact by the project would be reduced as low as practical, considering the project plan and the state of the area environmentally impacted by the project.

Finally, business operator shall prepare the environmental impact statement (EIS), taking into account the recommendation on the draft EIS, and submit it to NISA. NISA, after examining the EIS, orders alteration of the EIS if needed, otherwise notices acceptance of the EIS to business operator. The accepted EIS is distributed to the Ministry of Environment and related local governments.

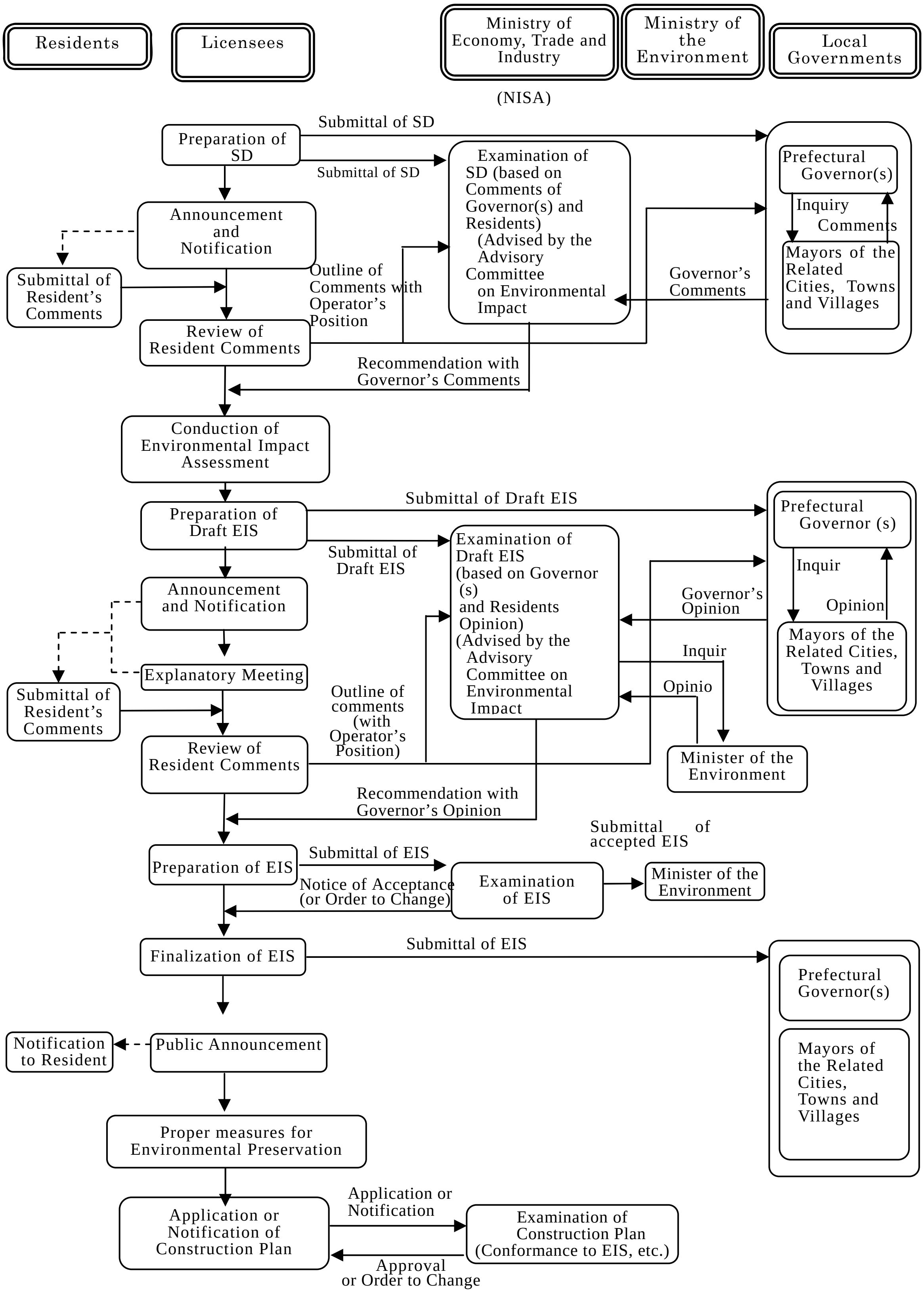
At the stage of examining construction plan, NISA does not approve it in case that the plan does not conform to the accepted EIS.

17.6 Re-evaluation of Site Related Factors

All the factors related to site selection must be re-evaluated at the time of alteration of an establishment license, such as additional new nuclear installation construction at the existing site, so as to ensure the continuous safe operation of the nuclear installation. Adequacy of the safety design is re-evaluated referring to new findings and new experiences having impact on the design.

17.7 Arrangements with Neighboring Countries on Safety Impact of Nuclear Installations

Nuclear installation in Japan is so located at the place where there are no events liable to induce serious accidents and so designed to secure the safety against postulated initiating events including natural phenomena. It also implements the measures for the accident management. Furthermore, because of the fact that Japan is an archipelago country and separated from neighboring countries by a considerable distance, adverse impact of Japanese Commercial Power Reactor over neighboring countries is deemed to be extremely small. Accordingly, no consultation has been made so far with neighboring countries on the siting of nuclear installations.



NOTE : EIS: Environment Impact Statement, SD: Scoping Document

Fig. 17-1 Outline of the Environmental Impact Assessment on Nuclear Power Plant

Article 18 Design and Construction

Each Contracting Party shall take the appropriate steps to ensure that:
the design and construction of a nuclear installation provides for several reliable levels and methods of protection (defense in depth) against the release of radioactive materials, with a view to preventing the occurrence of accidents and to mitigating their radiological consequences should they occur;
the technologies incorporated in the design and construction of a nuclear installation are proven by experience or qualified by testing or analysis;
the design of a nuclear installation allows for reliable, stable and easily manageable operation, with specific consideration of human factors and the man-machine interface.

The nuclear installations in Japan (light water reactors in commercial stage and fast breeder reactors in demonstration stage) were designed, constructed and operated based on the safety design concept, which is common to most Western countries, adopting fundamentally the same defense in depth system as prescribed in the Nuclear Safety Standards "NUSS" of the IAEA. Moreover, the knowledge obtained from operating experiences and various kinds of examination, analyses, research and development are utilized, on a continuous basis, to realize safer and easier facilities to carry out the maintenance management. Furthermore, this new knowledge is reflected appropriately and successively on planning and revising of guides etc. in order to improve the safety and reliability of nuclear reactors.

18.1 Review at the Design and Construction Stage of Nuclear Installations

Licensing process and relevant law and regulatory requirements applied at the design and construction stage for nuclear installation in Japan are described in the report of Article 7. *Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities, (hereinafter called as the "Regulatory Guide for Reviewing Safety Design"), Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities (hereinafter referred to as the "Regulatory Guide for Reviewing Safety Assessment") and related guides are used for the review of the design concerning safety of a nuclear installation.*

The basic policies of the safety design of nuclear installations for electricity generation are provided in the Regulatory Guide for Reviewing Safety Design. The Regulatory Guide requires that each system, structure and component constituting nuclear installations achieve the assigned functions under environmental and loading conditions during their in-service period (not only in the normal operating conditions but also in abnormal conditions to be postulated).

The Regulatory Guide for Reviewing Safety Assessment is used to confirm in the safety assessment that the reactor facilities consisting of such systems, structures and components, should be sufficiently safe ones as a whole. The Regulatory Guides provides the postulated events, criteria and items that should be taken into consideration.

When design alteration is required on licensed commercial nuclear installation, the licensee must undergo the verification on the safety impacts due to the alteration as well as the inspection on the altered segment, including the safety analysis influenced by the design alteration, in the same procedure as the licensing process as licensing a new installation.

18.2 Realization of Defense in Depth and Confinement of Radioactive Materials at the Design and Construction Stage

Commercial nuclear installation (light water reactors (BWRs and PWRs)) in Japan are designed, constructed and operated based on the safety design principals, which are common among most Western countries and fundamentally the same concept of "defense in depth" as prescribed in the Nuclear Safety Standards (NUSS) of the IAEA. In this section the first 3 levels of defense in depth concepts, which are closely related to design and construction of nuclear installations, are discussed. Forth and fifth levels of defense in depth concept, which are severe accident management and emergency preparedness, are discussed in sections 18.6 and report on Article 16. Original design of light water reactors in Japan was introduced from the United States. But, the later design of reactors has been improved so that the facilities have become safer and easier in maintenance management through a series of Improvement and Standardization Program led by METI (then MITI), reflecting the operating experiences of those who have obtained license for reactor establishment and knowledge obtained in research and development program of nuclear power industries.

(1) Implementation of the Defense in Depth Concept

The principle of "defense in depth" is as follows:

- Prevention of deviating from normal operation conditions by means of conservative design, manufacturing and construction of the nuclear plant in accordance with the relevant quality level and engineering practices.
- Detection of the occurrence of an abnormal event at an early stage and taking preventive measures against its progression into an accident. And
- Control of the progression of accident and mitigation of its consequences on the assumption that progression to an accident might not be prevented at the preceding stage.

In order to apply these fundamentals to design of commercial nuclear installation, the Regulatory Guide for Reviewing Safety Design (see Table 18-1) that was established by the NSC, stipulates the following items. The first defense is preventive measures for the occurrence of an abnormal event. More specifically, as stated in the requirements in guidelines 1 to 10 (overall nuclear reactor facility) of the Regulatory Guide for Reviewing Safety Design, the first defense implies such measures of designing with a safety margin, implementing strict quality control in fabrication, inspecting the facilities and component to be fabricated as required by the design and preventing degradation of performance through monitoring, check and maintenance during the operation. Each component, equipment and system of nuclear reactor facility is to be designed considering the importance of its safety function. The Regulatory Guide for Classification of the Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities requires that the quality control during design and manufacturing be conducted corresponding to the importance of safety function.

The second defense is to prevent expansion of abnormalities. More specifically, as stated in the requirements in guidelines 15 to 18 (Reactor Shutdown System), and 34 to 40 (Safety Protection System) of the Regulatory Guide for Reviewing Safety Design, the second defense implies the early detection of the abnormal condition, its correction or taking measures in advance to prevent the progression into an accident.

The third defense is to mitigate the consequence of an accident. More specifically, as stated in the requirements in guides 25 (Emergency Core Cooling System) and 28 to 33 (Reactor Containment) of the Regulatory Guide for Reviewing Safety Design, the third defense implies taking measures to secure the safety of the public in the vicinity by controlling the progression of the accident and mitigating its consequence.

The safety of nuclear facilities is ensured through rigorous safety measures on the basis of the defense in depth concept, which includes 1) preventing the occurrence of an abnormal event, 2) detection of the abnormal event and the preventing progression into an accident, and 3) mitigating an accident consequence. Consequently in Japan, through these measures, it is possible to reduce the potential for the occurrence of a severe accident to the extent that its actual occurrence would be technologically inconceivable, and to maintain the risk of the nuclear installation at a sufficiently low level. Based on such a status, preparation of the accident management can be regarded as a measure to reduce this low risk furthermore beyond these protection levels. In addition, preparation of the accident management and the emergency measures, which has been carried on in Japan, are described in section 18.4, and in Article 16, respectively.

(2) Confinement of Radioactive Materials (or Three Barriers of Radiation Protection Walls)

Nuclear facilities shall be designed, constructed and operated, in such a way as to confine radioactive materials within a series of physical barriers. These physical barriers are the fuel pellet, the fuel cladding, the reactor coolant pressure boundary and the reactor containment. The requirements for these physical barriers in the Regulatory Guide for Reviewing Safety Design etc. and the outcome of the design improvements in them are as follows:

1) Fuel (Including Claddings)

The fuel assembly shall be so designed that a) the integrity will be retained under the various conditions that could occur in the nuclear reactor in service; b) the safety protection system will actuate the reactor shutdown system, etc. so that the allowable design limit of the fuel shall not be exceeded at an abnormal transient during operation; c) the reactor core cooling will not be impaired by a reactivity insertion accident and, more specifically, the maximum fuel enthalpy by analysis will not exceed the specified value; and the emergency core cooling system will be capable of preventing major damage to the fuel in a loss of coolant accident, and the fuel cladding metal water reaction will be limited to sufficiently small amount.

Regarding item a), the design requirements are stipulated in guidelines 11 and 12 of the Regulatory Guide for Reviewing Safety Design. Regarding item b), the design requirements are stipulated in guidelines 34 to 40 (Safety Protection System). Regarding item c), the design requirements are stipulated in guidelines 12, 14 and 25. The requirements for safety assessment are also stipulated in the Regulatory Guide for Evaluating Reactivity Insertion Events of Light Water Nuclear Power Reactor Facilities and the Regulatory Guide for Evaluating Emergency Core Cooling System Performance of Light Water Nuclear Power Reactors.

2) Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be so designed that the integrity will be maintained during normal and abnormal operating conditions; that the boundary will not exhibit brittle behavior or develop rapid brittle fracture during normal operation, maintenance, testing, or abnormal conditions; that the leakage will be detected immediately and surely; that tests and inspections will verify its integrity throughout the service life of the nuclear reactor, which are required in guidelines 19 to 22 of the Regulatory Guide for Reviewing Safety Design. Pressure on reactor coolant pressure boundary will not exceed the specified value during reactivity insertion events, which is required in guideline 14 of the Regulatory Guide for Reviewing Safety Design.

3) Reactor Containment

The reactor containment shall be so designed that it will withstand the loads of design basis accident and the appropriate design basis earthquake; that it will prevent leakage exceeding the predetermined leakage rate; that it will allow periodic testing on the leakage rate; that its boundary will not exhibit brittle behavior or develop rapid brittle fracture during normal operation, maintenance, testing and in abnormal conditions; and that isolation valves should be placed in the pipelines that penetrate its walls, which are required in guidelines 28 and 29 of the Regulatory Guide for Reviewing Safety Design.

18.3 Systems for preventing the occurrence of abnormalities and for mitigating the impact of abnormalities (Regulatory Guide for Reviewing Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities)

In Japan, the Regulatory Guide for Reviewing Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities (hereinafter called as the “Regulatory Guide for Classification of Importance") prescribes the system for preventing the occurrence of abnormalities and the system for mitigating the impact of abnormalities. That is, since the Regulatory Guide for Reviewing Safety Design used at the safety review and assessment must be appropriately applied according to the safety importance of the subject structures, systems and components, safety functions and the classification of importance of the structures, systems and components are defined in this "Regulatory Guide for Reviewing

Classification of Importance".

(1) The Concept of the Classification of Importance for the Safety Design

The importance of safety functions of the structures, systems and components are classified into the following two classes and shown in the Regulatory Guide for Reviewing Classification of Importance.

1) Those of which loss of the function could result to cause an abnormality of the nuclear reactor facility, which causes excessive radiation exposure on general public or the working personnel (the system for preventing the occurrence of abnormalities, hereinafter called as “PS”).

2) Those that have the function to prevent the propagation of abnormality or terminate it quickly in an abnormal situation of a nuclear reactor facility, and to protect general public or the working personnel from possible excessive radiation exposure (the system for mitigating the impact of abnormalities, hereinafter called as “MS”). The structures, systems and components, which belong to these PS and MS respectively, are classified into three classes in accordance with the importance of their safety function. It stipulates, from the standpoint to ensure the safety function, that the basic objective for each class shall meet the following requirements according to the technologies of design, construction and tests, and operation management.

Class 1: Secure and maintain as high as reasonably achievable level of reliability.

Class 2: Secure and maintain a high level of reliability.

Class 3: Secure and maintain a level of reliability equal to or higher than that for general industry.

Moreover, it is required that functional isolation and physical separation among two or more systems, structures and components, which have safety functions, are taken into consideration appropriately. When connecting systems, structures and components having different degree of importance, it is required that the design requirements equivalent to those on systems, structures, and components with higher degree of importance should be applied to the other systems, structures and components with lower importance or that the appropriate functional isolation should be taken into consideration.

The classifications of the structures, systems and components and their safety functions are listed in Table 18-2.

(2) Installation of PS and MS

The PS and MS installed in the light water reactors in Japan are as follows. After grouping all light water reactors, currently installed in Japan, based on the reactor type and the containment type, the essential system for PS and MS, which are installed in each nuclear reactor facilities are shown in Table 18-3 and Table 18-4 for BWR and PWR respectively. These tables summarize the system configuration and their classification of reactor shutdown system, emergency core cooling system and heat removal system, the number of diesel generators and the containment shape, as essential systems for preventing the occurrence of abnormalities and system for mitigating the impact of abnormalities.

18.4 Safety Design Assessment

In the safety design assessment, postulated event groups are defined for "abnormal transients during operation" and "accidents", respectively, based on the Regulatory Guide for Reviewing Safety Assessment, as mentioned later, then the safety is evaluated by conducting safety analysis. These event groups conform to the classification defined in the Nuclear Safety Standards (NUSS) of IAEA almost.

The person who intends to install a nuclear reactor conducts the safety analysis for these postulated event groups, compares the analysis results with each criterion, and confirms that the safety design is appropriate.

On the other hand, the Nuclear and Industrial Safety Agency examines the safety analysis of the person who intends to install a nuclear reactor, and confirms its validity, getting an independent analysis report performed by the incorporated administrative agency, Japan Nuclear Energy Safety Organization, if necessary. The postulated events for the safety assessment are selected and evaluated in the following manners;

Malfunctions and erroneous actions of the systems or components, which are applied in the basic design, are analyzed, and the event which results in the severest case is selected among similar events in the propagation process as the postulated event group for the safety assessment. Depending on the possibility of occurrence and the degree of its impact at the time of occurrence, these postulated events are classified into "abnormal transients during operation" or "accident" as provided in the Regulatory Guide for Reviewing Safety Assessment, and the safety of those postulated events are also evaluated based on the criteria defined to each classification.

a. "Abnormal transients during operation"

"Abnormal transients during operation" are defined as events that result in abnormal conditions caused by an single equipment failure, erroneous action or single disoperation assumable in the lifetime of the nuclear installation, and the external disturbance assumable to occur with similar frequency of the single equipment failure, etc. during the operation of commercial power reactors, and fourteen events and twelve events are selected for pressurized water reactors (PWR) and boiling water reactors (BWR), respectively. The safety analysis is conducted for these events based on the criteria of the Regulatory Guide for Reviewing Safety Assessment, the integrity of core and reactor coolant pressure boundary is confirmed, and the adequacy of the safety design of important safety related equipment, such as the safety protection system and the reactor shut-down system is logically clarified.

b. "Accident"

The "accident" is an abnormal condition exceeding the "abnormal transients during operation", which is assumed from a necessity to evaluate a release of radioactive materials from a commercial power reactor when it occurs, although the frequency of occurrence is small. Ten events and nine events for PWR and BWR, respectively, are selected. The safety analysis is conducted for these events based on the criteria of the Regulatory Guide for Reviewing Safety Assessment, and it is confirmed that the core does not result in a significant damage and a reactor containment boundary is sound. Moreover, no risk of excess radiation exposure to the general public in the vicinity is confirmed. And it is logically confirmed that the safety design of engineered safety features is appropriate.

In addition, the analysis of the "accidents" event are verified and evaluated for the loss of coolant accidents based on the Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities and the "Regulatory Guide for Evaluating Emergency Core Cooling System Performance of Light Water Nuclear Power Reactor" and for the reactivity insertion events based on the Regulatory Guide for Reviewing Safety Assessment and the "Evaluation Guide for Reactivity Insertion Events of Light Water Nuclear Power Reactor Facility", etc.

18.5 *New Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities*

"The Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities", which is a specific seismic guide related to the general reviewing guide "the Regulatory Guide for Reviewing Safety Design", was revised in September 2006. The old Regulatory Guide for Reviewing Seismic Design was issued in 1978 by the Atomic Energy Commission. The provisions about static seismic force was revised in 1981 by the Nuclear Safety Commission, and the terminology was revised based on the recommendations of the International Committee on Radiation Protection (ICRP) in 2001. Since then, lots of new knowledge about seismology and earthquake engineering has been obtained and the design and technology concerning seismic safety have been remarkably improved, and in particular, after the South Hyogo Earthquake which occurred in 1995, knowledge and information have been obtained through the researches and studies which have been carried out relating to that earthquake.

In order to make the Regulatory Guide for Reviewing Seismic Design etc. more appropriate, taking in the latest knowledge etc., the discussions and reviews were promoted by the Nuclear Safety Commission in 2001. After open discussions and reviews and hearing of opinion from the public, the new Regulatory Guide for Reviewing Seismic Design was issued in September 2006.

Summary of the new Regulatory Guide for Reviewing Seismic Design is as follow;

1) *Advanced methods to evaluate /to determine design basis earthquake ground motions (geological survey etc.)*

(i) Extension of geologic age of active-fault evaluation

The geologic age investigated for surveying the active-faults (traces of past earthquakes), which should be taken into consideration on the seismic design, has been traced back to the late Pleistocene age, if activities cannot be denied. Formerly it was the age until 50,000 years ago.

(ii) Careful active-fault survey

The active-fault survey required when deciding the design basis earthquake ground motion to be used in a seismic design should be carried out more in detail and carefully, depending on the distance from the site, integrating various methods of tectonic geomorphologic examination, surface-of-the-earth geological survey and geophysical survey, so that all possible measures can be applied in the evaluation of active faults that should be taken into consideration on the seismic design

2) Method to evaluate / to determine earthquake ground motions (Determination of the design basis earthquake ground motion)

(i) Unification of design basis earthquake ground motions

The design basis earthquake ground motion is changed from conventional set of two types (the design basis earthquake ground motion S_1 based on the design basis maximum earthquake and the design basis earthquake ground motion S2 based on a design basis extreme earthquake and/or a near field earthquake) to set of one type (S_S) and of which setting conditions are more strict than that of S_2. The basic requirements for S_S are that the safety function of facilities important to seismic safety should be kept.

(ii) Advanced evaluation method of the earthquake ground motion, which is determined by identifying the source for each site

In addition to the old experiential evaluation method (the method using a response spectrum), the "fault model", which is the newest evaluation method, was introduced newly in full scope, so that the evaluation method of earthquake ground motions are enhanced by taking advantages of both methods.

(iii) Introduction of evaluation method of an earthquake ground motion, which is determined without identifying its source

In the case of an earthquake in the inland earth's crust, it sometimes occurs difficult to identify the relationship between the source and the active fault. Therefore, it was decided to determine an earthquake ground motion by setting up a response spectrum based on the observation records of such earthquakes. This has realized the preparation for the evaluation on an earthquake, for which the evaluation would not be possible even if a careful active-fault survey is carried out. As a result, the *provision of the near field earthquake with magnitude 6.5 in the former guide was abolished.*

(iv) Individual evaluation on the vertical earthquake ground motion

In the new Regulatory Guide for Reviewing Seismic Design, it was decided to determine the vertical dynamic earthquake ground motion also with the horizontal one in determining S_S earthquake ground motion..

(v) Consideration on "a residual risk"

As it is impossible to completely deny the occurrence of an earthquake with the ground motion exceeding S_S, a "residual risk" was decided to be taken into consideration. Furthermore, it is required to take into consideration factors and the magnitudes of the "uncertainties" to the size, position, propagation etc. with an appropriate method, and to refer to the probability of exceeding S_S earthquake as the reference information at the safety review and assessment of the plant.

3) Reexamination on the importance classification with regards to seismic safety

(i) Expansion of the scope of the facilities important for the safety

For the seismic safety design, the scope of the most important facilities, such as the reactor containment (previous As class), was expanded to include the emergency core cooling system etc. (previous A class)

(ii) Requirement to take into consideration the accompanying events of earthquake

It was described clearly to take consideration of the accompanying events of earthquake (collapse of the inclined planes around facilities, tsunami etc.).

(iii) Improvement of requirements for a rock-bed support

In consideration of the progress of seismic isolating technology etc. ,the "rock-bed support" requirement for the building and structure has been changed to a performance based requirement prescribing, "construct on a soil with adequate support performance".

4) Effort to use the probabilistic-safety-assessment methodologies

(i) It was decided that all licensees should make efforts to make the "residual risk" as low as reasonably achievable, and that the effort towards extensive introduction of the probabilistic safety assessment methods should be made in the future.

The above-mentioned new Regulatory Guide for Reviewing Seismic Design is applied to the nuclear installations on which the application of the establishment approval will be made newly from now on, and for existing nuclear installations, it *is requested that all licensees evaluate the seismic safety based on the revised contents. Actions to the existing nuclear installations are provided in Section 14.5.*

18.6 Preparation of Accident Management Measures

Since the TMI-2 accident, the researches on phenomena of severe accidents and PSA have been conducted extensively worldwide. The NSC decided "Accident Management of Severe Accidents at Power Generating Light Water Reactor Facilities" in 1992, and revised it in 1997. Licensees in Japan also have voluntarily implemented their own measures for preventing severe accidents and for mitigating their consequences at the request of the MITI (then) based on the NSC's decision. *Typical facility modifications for the accident management to prevent an occurrence and to mitigate the consequence of a severe-accident are as shown in the following;*

PWR: Alternative recirculation (installation of alternative sump-pumps, or core flooding using the containment spray system by installing the tie-line between the containment spray system and the residual heat removal system), containment natural convection cooling (utilization of the common containment cooling system), alternative component cooling (utilization of the HVAC chilled-water system etc.), water injection into a reactor containment (utilization of the fire protection system), common usage of power supply among units (usage of power from the neighboring nuclear installation), hydrogen-concentration control (only for ice condenser type PWR)

BWR: Alternative reactivity control (recirculation pump trip and automatic alternative control rod insertion), alternative cooling water injection(utilization of the fire protection system), automatic reactor depressurization(automatic depressurization at the low water level of RPV), heat-removal from a reactor containment(pressure venting for preventing vessel rupture and utilization of drywell cooler), power supply system(common usage of power supply among the neighboring nuclear installation)

For implementing accident management at operating commercial nuclear installations, the licensees have been developing the accident management measures progressively, substantiating the facilities *as mentioned above* during the outage of the periodic inspection as well as establishing operational measures such as implementing system, procedures, education of personnel, etc.

The accident management measures that were prepared by licensees were reported to NISA in May, 2002, together with the PSA results of internal events for representative reactor types for the purpose of quantitatively verifying the effectiveness of enhancement of the safety. While licensees were developing accident management measures, NISA requested NUPEC (then) to evaluate the effectiveness of the accident management measures, and established the "Accident Management Workgroup" under the Nuclear and Industrial Safety Subcommittee to obtain the opinion of specialists, and evaluation report was compiled and issued in October 2002. The report was submitted to the NSC by NISA in the same month. The PSA results of the internal events for all commercial power reactor facilities under operation (excluding representative reactor types) were reported to NISA by the licensees in March 2004.

The development programs of the accident management for commercial nuclear installation under construction (three units) were reported to NISA by the licensees in July 2003, and the evaluation results etc. were reported to the NSC by NISA in September 2003. The NSC evaluated the report and concluded it was reasonable in December 2003. In addition, the accident management measures for the reactor facilities concerned are being prepared by the licensees.

18.7 Measures to Ensure the Technical Reliability by Experience, Test and Analysis

In Japan, such actions as feedback of the operating experience and utilization of the technical knowledge obtained through testing and analysis have been taken, so that the safety and reliability of commercial nuclear installation has been enhanced .The brake-downs are described below. The new knowledge obtained through these actions has been timely incorporated in existing guidelines and used to develop new guidelines.

(1) Feedback of Operating Experiences from Commercial Nuclear Installations

- Good practices and non-compliance examples identified during periodic inspections, as well as the experiences in design, construction and operation of domestic and foreign commercial nuclear installation, were analyzed, and the results are incorporated in design modification, improvement of construction methods, etc., when they can be recognized to be effective, during the course of licensing for establishment, approval of construction plans and pre-service inspection.

- For accidents or failures occurred in the domestic commercial nuclear installations as well as in foreign reactors, the corrective measures are implemented after identifying the cause of failures.

- From the standpoint of the comprehensive preventive maintenance of nuclear reactor facilities, periodic safety review is performed for each commercial power reactor with the interval of approximately ten years. And its safety and reliability are confirmed, reflecting the results of comprehensive evaluation on operating experiences and the latest technical knowledge. The situation of periodic safety review is described in Article 14.

- *Since the Three Mile Island accident, the habitability of a central control room has been reexamined. Meanwhile in Japan, there was an event in which the steam invaded the central control room at the time of the pipe break accident due to the wall thinning of the second system piping of the Mihama Unit No. 3 in 2004. It was found that the air-tightness of the central control room was insufficient. Although filling was provided as a temporary measure to keep the air-tightness, a fundamental measure has to be taken. On the occasion when the technical standard was amended to be based on performance requirements, the air-tightness requirement was added on the central control room, and it will be a matter to be evaluated at the safety* *examination of a new nuclear installation. And the tests of leak-tightness of the central control room have been conducted on 3 BWRs and one PWR so far and there are the plans to conduct leak tightness test on some additional control rooms further. Based on the results of these testing, the private sectors' standards, relating to leak tightness testing will be developed..*

- *As many fire events have occurred repeatedly in and outside Japan and as the past OSART review has made recommendations/suggestions. About the fire-protection management, the standards and the guidelines on design and management for fire protection of nuclear installations have been re-examined and improved in Japan. On an occasion when the technical standards was amended to more performance-based one, the requirements of the fire protection in the technical standard were reexamined, and the requirements were clarified for each stage of "fire prevention", "early stage detection and extinguishing of fire" and "fire consequence mitigation". After these activities, the fire-protection standards for the design and development of standards of academic societies and industrial associations for operation management have been re-examined. Furthermore, safety researches started, including the study for development of a fire PSA method and the participation to the international project for various fire experiments..*

- *The electric cables currently used in the nuclear installation, while the aging advances gradually by oxidation etc., sudden performance degradation may occur in the environment of high temperature steam and high radiation at the time of a postulated design base accident. The studies are carried out to evaluate such aging and performance degradation and to confirm the integrity of cables during in-service operation. The studies to obtain the heat deterioration data and the heat and radiation deterioration data, etc. have been conducted using test samples of the safety-related cable currently used in the nuclear installation, aiming at comprehensive assessment of the cable aging characteristics as well as at correct assumption of the environmental design conditions and at establishment of integrity judging methods, adding the new knowledge obtained in recent years to the study results. Thereby the evaluation methods and evaluation test guideline of aged cables will be established on the basis of the actual conditions in a nuclear installation.*

(2) Feedback of the Knowledge Obtained Through Test and Analysis

Recognizing the importance of assuring safety in development and utilization of nuclear energy, the researches for advancement of safety standards, guidelines, reference materials for acceptance decision in safety review and assessment etc., as well as improvement of the safety itself, are promoted in Japan. *The explanation of promoting safety research in Japan is provided in Section 14.8.*

The major subjects related to the research are shown in the following;

1) Study corresponding to advanced light water reactor fuels

- *Study on the safety of high-burn-up MOX fuel*
- *Confirmatory test on the safety margin of high burn-up fuels*
- *Reliability demonstration test of 9X9 type fuel*
- *Reliability demonstration test of nuclear design methodology for the full MOX core*

2) Study on advanced safety assessment technologies

- *Research of advanced nuclear and thermal-hydraulic best estimate method*
- *Improvement of safety analysis codes for nuclear power reactors*

3) Study on severe accident

- *Study on the water hammer by rapid void growth at the time of severe accident*
- *Study on maintenance of the containment confinement function at the late stage of a severe accident*

4) Seismic safety study of nuclear installations

- *Study on the design earthquake ground motion with a consideration of the characteristics of the postulated earthquake*
- *Test on the seismic-assessment technologies of nuclear installations*
- *Study on the hazard map for seismic design*

18.8 Consideration of Human Factors and Man-Machine Interface

It is the safety requirements regarding operating management to make nuclear installation more reliable, more stable and more easily manageable taking into human factors and man-machine interface. These requirements are implemented in design and operation of the commercial power reactors in Japan.

Considerations of the design to an operator behavior, requirements on the design of a control room and concrete design to approach these requirements are described in the report of Article 12.

Table 18-1 Individual guides established in the NSC Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities (1/2)

(General requirements for nuclear reactor facilities) Guideline 1. Applied codes and standards Guideline 2. Design considerations against natural phenomena Guideline 3. Design considerations against external human initiated events Guideline 4. Design considerations against internal missile Guideline 5. Design considerations against fire Guideline 6. Design considerations against environmental conditions Guideline 7. Design considerations for share use Guideline 8. Design considerations against operator actions Guideline 9. Design considerations for reliability Guideline 10. Design considerations for testability
(Nuclear reactor and reactor shutdown system) Guideline 11. Reactor Core design Guideline 12. Fuel design Guideline 13. Reactor characteristics Guideline 14. Reactivity control system Guideline 15. Independence and testability of reactor shutdown system Guideline 16. Reactor shutdown margin by control rods Guideline 17. Shutdown capability of reactor shutdown system Guideline 18. Reactor shutdown system capability at the accident
(Reactor cooling system) Guideline 19. Integrity of reactor coolant pressure boundary Guideline 20. Prevention of reactor coolant pressure boundary failure Guideline 21. Detection of the reactor coolant pressure boundary leaks Guideline 22. In-service test and inspection of reactor coolant pressure boundary Guideline 23. Reactor coolant make-up system Guideline 24. Systems for removing residual heat Guideline 25. Emergency core cooling system Guideline 26. System for transporting heat to ultimate heat sink Guideline 27. Design considerations against loss of power
(Reactor containment) Guideline 28. Function of reactor containment Guideline 29. Prevention of reactor containment boundary failure Guideline 30. Isolation function of reactor containment Guideline 31. Reactor containment isolation valves Guideline 32. Reactor containment heat removal system Guideline 33. System for controlling containment facility atmosphere
(Safety protection system) Guideline 34. Redundancy of safety protection system Guideline 35. Independence of safety protection system Guideline 36. Function of safety protection system during transients Guideline 37. Function of safety protection system in case of the accident Guideline 38. Function of safety protection system in case of failure Guideline 39. Separation of safety protection system from instrumentation and control systems Guideline 40. Testability of safety protection system

Table 18-1 Individual Guidelines established in the NSC Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities (2/2)

(Control room and emergency facilities) Guideline 41. Control room Guideline 42. Reactor shutdown function from outside of control room Guideline 43. Design considerations for control room habitability Guideline 44. On-site emergency station Guideline 45. Design considerations for communication equipment Guideline 46. Design considerations for evacuation routes
(Instrumentation and control system and electrical system) Guideline 47. Instrumentation and control system Guideline 48. Electrical system
(Fuel handling system) Guideline 49. Fuel storage and handling system Guideline 50. Prevention of fuel criticality Guideline 51. Monitoring of fuel handling area
(Radioactive waste processing facility) Guideline 52. Radioactive gaseous waste processing facility Guideline 53. Radioactive liquid waste processing facility Guideline 54. Radioactive solid waste processing facility Guideline 55. Radioactive solid waste storage facility
(Radiation management) Guideline 56. Environmental radiation protection Guideline 57. Radiation protection for personnel engaged in radiation work Guideline 58. Radiation management for personnel engaged in radiation work Guideline 59. Radiation monitoring

Table 18-2 Definitions and functions with respect to classifications of importance of safety function (1/2)

<table>
<tr><th colspan="2">Classification</th><th>Definition</th><th>Function</th></tr>
<tr><td rowspan="3">Class 1</td><td>PS-1</td><td>Structures, systems and components where there is concern that
(a) a conspicuous damage to the core, or
(b) significant damaging the core may occur, due to an event caused by such damage or malfunction.</td><td>(i) Reactor coolant pressure boundary function
(ii) Excessive reactivity insertion prevention function
(iii) Core shape maintenance function</td></tr>
<tr><td rowspan="2">MS-1</td><td>(i) Structures, systems and components that implement an emergency shutdown of the nuclear reactor, remove the residual heat, prevent excess pressure in the reactor coolant pressure boundary and prevent an impact of excessive radiation on the public in the site vicinity, at the occurrence of an abnormal condition.</td><td>(i) Reactor emergency shutdown function
(ii) Sub-criticality maintenance function
(iii) Function to prevent over-pressurization of reactor coolant pressure boundary
(iv) Cooling function after reactor shutdown
(v) Core cooling function
(vi) Radioactive material confinement function, shielding of radiation and release reduction functions</td></tr>
<tr><td>(ii) Other essential safety related structures, systems and components</td><td>(i) Generation function of an actuation signal for the engineered safety features and to the reactor shutdown system
(ii) Specially important safety related functions</td></tr>
<tr><td rowspan="2">Class 2</td><td rowspan="2">PS-2</td><td>(i) Structures, systems and components for which there is no concern of the immediate causing of conspicuous reactor damage or significant fuel damage due to an event that occurs due to such a damage or malfunction, however, for which there is a concern of excessive release of radioactive materials outside the site vicinity.</td><td>(i) Function the builds in reactor coolant (However, this excludes small diameter piping, such as instrumentation, etc., excluded from the reactor coolant pressure boundary and those that are not connected directly to the boundary.)
(ii) Components not directly connected to the reactor coolant pressure boundary, which have the radioactive materials storage function
(iii) Function for the safe handling of fuel</td></tr>
<tr><td>(ii) Structures, systems and components which must be actuated during normal operation and upon an abnormal transient during operation for which there is a high potential that core cooling will be lost due to the concerned malfunction.</td><td>(i) Safety valve and relief valve re-closing function</td></tr>
</table>

Reference: “Regulatory Guide for Reviewing Classification of Importance of Safety Functions for Light Water Nuclear power Reactor Facilities”, decided by the NSC in August 30, 1990.

Table 18-2 Definitions and functions with respect to classifications of importance of safety function (2/2)

Classification		Definition	Function
	MS-2	(i) Structures, systems and components for adequately reducing the impact of radiation on the general public in the vicinity of the site, due to damage or malfunction in the structures, systems and components of the PS-2.	(i) Fuel pool water supply function (ii) Function to prevent the discharge of radioactive materials
		(ii) Structures, systems and component with an especially important function in the response of abnormal situations.	(i) Function for determining the situation of the plant at the time of an accident (ii) Function for mitigation of abnormal situations (iii) Function for safe shutdown from outside the control room
Class 3	PS-3	(i) Structures, systems and components where initiating events of abnormal situations take place, and which are other than PS-1 and PS-2 components.	(i) Reactor coolant preserving function (components other than PS-1 and PS-2) (ii) Reactor coolant circulation function (iii) Radioactive material storage function (iv) Power supply (excluding emergencies) (v) Plant instrumentation and control function (excluding safety protection function) (vi) Plant operation supporting functions
		(ii) Structures, systems and components which hold the concentration of the radioactive materials in the reactor coolant to a level low enough not to impair normal operation	(i) Function for preventing the diffusion of fission products into the reactor coolant (ii) Reactor coolant purification function
	MS-3	(i) Structures, systems and components which mitigate events in conjunction with the MS-1 and MS-2, even when there is an abnormality during operation	(i) Function for mitigation of reactor pressure increase (ii) Function to control the power increases (iii) Reactor coolant make-up function
		(ii) Structures, systems and components required for the response of abnormal situations	(i) Important for emergency response and function for recognizing abnormal situations

Table 18-3 Establishment situation of prevention and mitigation system (BWR nuclear installation) (1/2)

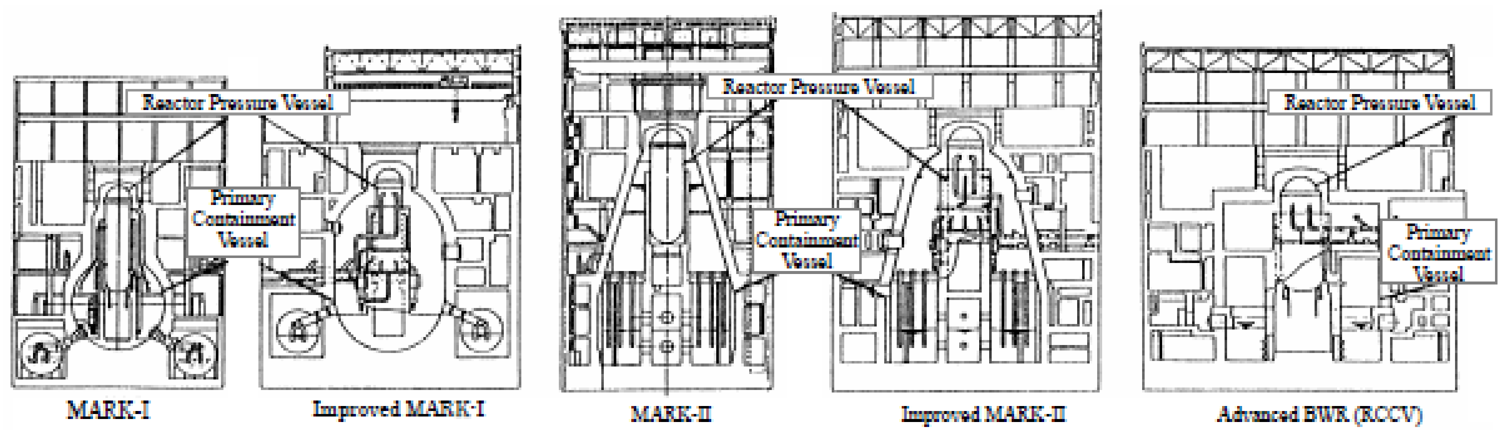
Plant type	BWR 2 & 3	BWR 4	BWR 5	ABWR
Containment type	MARK-I	MARK-I	Improved MK-I, MK-II and Improved MK-II	RCCV
Name of power station	Unit 1 of Tsuruga PS (BWR 2) Unit. 1 of Fukushima Daiichi, (BWR 3)	Unit 1 of Onagawa NPS Unit 1 of Shimane NPS Unit 2 of Fukushima Daiichi NPS Unit 3 of Fukushima Daiichi NPS Unit 4 of Fukushima Daiichi NPS Unit 5 of Fukushima Daiichi NPS Unit 2 of Hamaoka NPS	Unit 1 of Shika NPS (Improved MK-I) Unit 2 of Shimane NPS (-ditto-) Unit 2 of Onagawa NPS (-ditto-) Unit 3 of Onagawa NPS (-ditto-) Unit 3 of Hamaoka NPS (-ditto-) Unit 4 of Hamaoka NPS (-ditto-) Tokai No.2 (MK-II) Unit 6 of Fukushima Daiichi NPS, (-ditto-) Unit 1 of Fukushima Daini NPS (-ditto-) Unit 1 of Kashiwazaki Kariwa NPS (-ditto-) Unit 2 of Fukushima Daini NPS (Improved MK-II) Unit 3 of Fukushima Daini NPS (-ditto-) Unit 4 of Fukushima Daini NPS (-ditto-) Unit 2 of Kashiwazaki Kariwa NPS (-ditto-) Unit 3 of Kashiwazaki Kariwa NPS (-ditto-) Unit 4 of Kashiwazaki Kariwa NPS (-ditto-) Unit 5 of Kashiwazaki Kariwa NPS (-ditto-), *Others*	Unit 6 of Kashiwazaki Kariwa NPS Unit 7 of Kashiwazaki Kariwa NPS *Unit 2 of Shika NPS* *Unit 5 of Hamaoka NPS*
Reactor shutdown system	SCRAM system Stand-by Liquid Control System	SCRAM system Stand-by Liquid Control System	SCRAM system Stand-by Liquid Control System	SCRAM system Stand-by Liquid Control System
Containment shape				

Table 18-3 Establishment situation of prevention and mitigation system (BWR nuclear installation) (2/2)

Plant type	BWR2&3	BWR4	BWR5	ABWR
Containment type	MARK-I type	MARK-I type	Improved MK-I, MK-II and Improved MK-I	RCCV type
System configuration of ECCS and heat removal system	HPCI IC / IC CS / CS CS / CS CCS / CCS CCS / CCS SHC / SHC ADS	RCIC / HPCI CS / CS LPCI / LPCI LPCI / LPCI RHR / RHR ADS	HPCS RCIC LPCS LPCI/RHR / LPCI LPCI/RHR ADS	HPCF LPFL/RHR RCIC LPFL/RHR / HPCF LPFL/RHR ADS
Divisions of system configuration	2 partitions	2 partitions	3 partitions	3 partitions
Number of D/G	2	2	3	3

IC: Isolation Condenser.
CS: Core Spray Sys.
CCS: Containment Cooling Sys.
RHR: Residual Heat Removal Sys.
SHC: Shutdown Cooling Sys.
HPCI: High Pressure Core Injection Sys.
LPCI: Low Pressure Coolant Injection Sys.
HPCF: High Pressure Core Flooder
RCIC: Reactor Core Isolation Cooling Sys.
ADS: Automatic Depressurization Sys.
HPCS: High Pressure Core Spray Sys.
LPFL: Low Pressure Core Flooder

Table 18-4 Establishment situation of prevention and mitigation systems (PWR nuclear installation) (1/2)

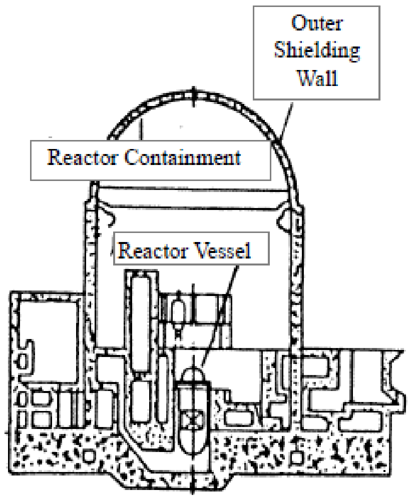
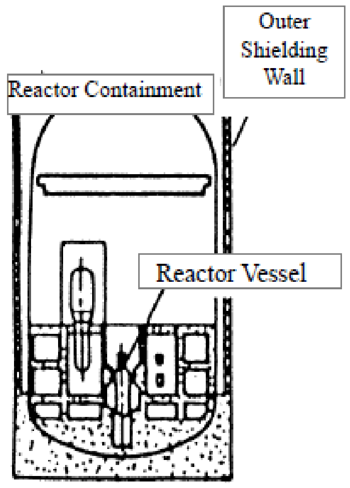
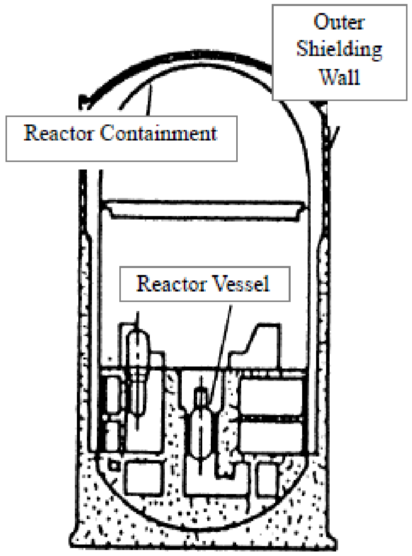
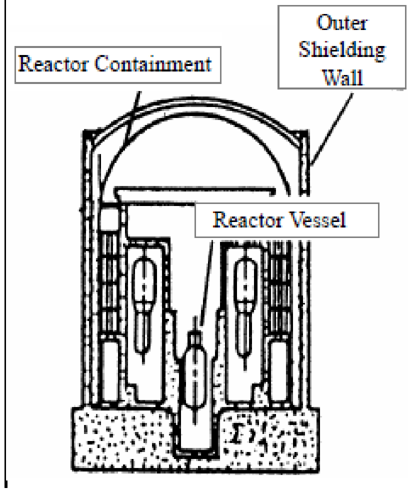
Plant Type	4 Loop	2 Loop	3 Loop	4 Loop
Containment type	PCCV type	Dry type	Dry type	Ice condenser type
Name of power station	Unit 3 of Ohi PS Unit 4 of Ohi PS Unit 2 of Tsuruga PS Unit 3 of Genkai NPS Unit 4 of Genkai NPS	Unit 1 of Ikata PS Unit 2 of Ikata PS Unit 1 of Mihama PS Unit 2 of Mihama PS Unit 1 of Genkai NPS Unit 2 of Genkai NPS Unit 1 of Tomari PS Unit 2 of Tomari PS	Unit 1 of Takahama PS Unit 2 of Takahama PS Unit 3 of Takahama PS Unit 4 of Takahama PS Unit 3 of Mihama PS Unit 1 of Sendai NPS Unit 2 of Sendai NPS Unit 3 of Ikata PS	Unit 1 of Ohi PS Unit 2 of Ohi PS
Reactor shutdown system	Scram system Boric acid injection system	Scram system Boric acid injection system	Scram system Boric acid injection system	Scram system Boric acid injection system
Containment shape	PCCV type 	Dry type  Freestanding steel type (no top dome)	 Freestanding steel type (with top dome)	Ice condenser type 

Table 18-4 Establishment situation of prevention and mitigation systems (PWR nuclear installation) (2/2)

Plant Type	4 Loop	2 Loop	3 Loop	4 Loop
Containment type	PCCV type	Dry type	Dry type	Ice condenser type
System configuration of ECCS and heat removal system	ACC 4 units HPIS LPIS (/RHR) AFWS (motor driven) / HPIS LPIS (/RHR) AFWS (motor driven) AFWS (turbine driven)	ACC 2 units HPIS LPIS (/RHR) AFWS (motor driven) / HPIS LPIS (/RHR) AFWS (motor driven) AFWS (turbine driven)	ACC 3 units HPIS (/CHP) HPIS (/CHP) LPIS (/RHR) AFWS (motor driven) / HPIS (/CHP) LPIS (/RHR) AFWS (motor driven) AFWS (turbine driven)	ACC 4 units HPIS HPIS C/CHP) LPIS (/RHR) AFWS (motor driven) / HPIS HPIS C/CHP) LPIS (/RHR) AFWS (motor driven) AFWS (turbine driven)
Divisions of system configuration	2 systems	2 systems	2 systems	2 systems
	HPIS boosting unnecessary	HPIS boosting necessary	HPIS boosting necessary	HPIS boosting necessary
Number of D/Gs	2	2	2	2

ACC: Accumulator, AFWS: Auxiliary Feed Water Sys, LPIS: Low Pressure Coolant Injection, PIS: High Pressure Injection Sys.
RHR: Residual Heat Removal Sys., CHP: Charging Pump

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Article 19 Operation

Each Contracting Party shall take the appropriate steps to ensure that:

(i) the initial authorization to operate a nuclear installation is based upon an appropriate safety analysis and a commissioning programme demonstrating that the installation, as constructed, is consistent with design and safety requirements;

(ii) operational limits and conditions derived from the safety analysis, tests and operational experience are defined and revised as necessary for identifying safe boundaries for operation;

(iii) operation, maintenance, inspection and testing of a nuclear installation are conducted in accordance with approved procedures;

(iv) procedures are established for responding to anticipated operational occurrences and to accidents;

(v) necessary engineering and technical support in all safety-related fields is available throughout the lifetime of a nuclear installation;

(vi) incidents significant to safety are reported in a timely manner by the holder of the relevant license to the regulatory body;

(vii) programmes to collect and analyse operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies;

(viii) the generation of radioactive waste resulting from the operation of a nuclear installation is kept to the minimum practicable for the process concerned, both in activity and in volume, and any necessary treatment and storage of spent fuel and waste directly related to the operation and on the same site as that of the nuclear installation take into consideration conditioning and disposal.

Licensees are allowed to commence operation after the licensing conditions specified by the Reactor Regulation Law etc. are complied with in the stage from licensing for establishment to the construction of commercial nuclear installations.

As legal regulations for licensees to operate reactors safely, it is obligated to provide the Operational Safety Programs approved by the Minister of METI before commencement, and to observe necessary measures for operation and maintenance of the commercial nuclear installation like the Periodic Inspection etc., and all through its operating life.

By amendment of the Electricity Utilities Industry Law in 2003, *the Periodic Inspection and the Periodic Licensee's Check were clearly established, and the scheme of the Periodic Safety Management Review was newly established. Moreover, quality assurance activities, maintenance management activities, the periodic safety review, etc. were decided to be included into the Operational Safety Program. The Task Force on the Inspection System including the use of risk assessments and performance-based evaluations is underway aiming to start at 2008.*

19.1 Initial License

In Japan Licensees are required by the Reactor Regulation Law to take necessary measures for the safe operation of nuclear installations and protection of specific nuclear fuel materials.

(1) “Reactor Establishment”

Throughout each stage of detailed design, construction and operation of a nuclear installation, the basic design (items defined in the main text of application documents for Reactor Establishment) approved by the Minister of METI must be followed

(2) “Construction Plan approval”

Licensees are also required to observe the conditions of Reactor Establishment within the Construction Plan approval (or Design Approval for fuel assembly), in which the detailed design for each facility of a nuclear installation is reviewed.

(3) “Pre-service Inspection”

Before the commissioning of a nuclear installation, NISA conducts the pre-service inspection to ensure compliance with the licensing conditions.

19.2 Limiting Conditions for Operation

Operation and maintenance of nuclear installations are carried out in accordance with the Operational Safety Program approved by the Minister of METI in Japan *(items included in the Operational Safety Program are shown in “Operational Safety Program” in Section 19-3)*.

Limiting Conditions for Operation (hereinafter referred to as LCO) of a nuclear installation include shutdown margin, reactor thermal limits, etc. Table 19-1 shows the details.

If the LCO is not complied with, the Minister of METI can *order* the licensee to suspend the operation of the nuclear installation, etc. in accordance with the Reactor Regulation Law.

Since December 2001, the rated thermal power operation that limits only thermal power was allowed. Power stations shown in Table 19-3 have been subject to this operating mode to date.

19.3 Regulations for Operation, Maintenance, Inspection and Testing

The overviews of (1) present regulatory framework (2) inspection systems and (3) the on-going Task Force on the Inspection System in line with the Reactor Regulation Law are as follows:

(1) Present Regulatory Framework

Elements that consist of the regulation in Japan after commissioning are as follows;

- *“Application for alteration of the Reactor Establishment and application of Construction Plan approval”*

In the case of reconstruction or repair after commissioning, an application for alteration of the Reactor Establishment or an application for Construction Plan approval may be needed,

- *"Operational Safety Program"*

Licensees are obliged to describe the following items related to the operation in the Operational Safety Program in accordance with the rules of the Reactor Regulation Law,

1) *Duties of personnel engaged in the operation and management of the nuclear reactor facility, and the organization*
2) *Items with respect to safe operation education for personnel engaged in the operation and management of the nuclear reactor facility*
3) *Operation of the nuclear reactor facility*
4) *Safety reviews on the operation of the nuclear reactor facility*
5) *Designation of radiation controlled areas, access controlled areas and environment monitoring areas, and restriction of access to these areas*
6) *Ventilation and drainage monitoring equipment*
7) *Monitoring of the dose, the dose equivalent, the concentration of radioactive materials and the surface contamination density of radioactive materials of objects contaminated by radioactive materials, and the decontamination*
8) *Management of radiation measuring instruments*
9) *Patrols and checks of the nuclear reactor facility and their associated measures*
10) *Receipt, delivery, transport, storage and other handling of nuclear fuel materials*
11) *Disposal of radioactive waste*
12) *Measures to be taken in an emergency*
13) *Records on safe operation of the nuclear reactor facility (including compliance of the Operational Safety Program)*
14) *Maintenance management of the nuclear reactor facility (except those contained in the next item)*
15) *Periodic assessment of the nuclear reactor facility*
16) *Quality assurance of the nuclear reactor facility*
17) *Other necessary items for safe operation of the nuclear reactor facility*

The "Operational Safety Program" is also subject to change as appropriate, during the usual operational period.

The LCO of nuclear installations, such as the shutdown margin and reactor thermal limits, etc., are included in the Operational Safety Program. Table 19-1 shows the items of the LCO of nuclear installations in Japan.

Items 14, 15, and 16 are modified as new items after October 2003;

- *Maintenance management of the nuclear reactor facility (except those contained in the next item): Licensees must establish and implement matters related to policies and objectives of maintenance management, a plan for implementation of the maintenance management, evaluations of the results, corrective actions, records etc,*

- *Periodic assessment of the nuclear reactor facility: Licensees must perform the Periodic Safety Review of reactor facilities after the commissioning of a reactor facility every ten years, and,*

- *Quality assurance of the nuclear reactor facility: Licensees must establish the organization in charge of the quality assurance and the system where planning, implementation, evaluation and improvement are continuously conducted.*

Licensees prepare various kinds of operation manuals and test procedures so as to determine more detailed operation procedures on the basis of Operational Safety Programs. Licensees, by establishing committees, assess important matters related to the alternation of Operational Safety Programs or procedures and safe operation of nuclear reactors in advance of implementation thereof.

NISA gave an order to change the Operational Safety Programs as a result of the Comprehensive Check referred in Article 6. The contents of the changes here vary from one licensee to another; the principal changes were as follows;

- *Stronger commitment of the licensees' management into safety enhancement activities*

- *Ensure independency of Chief Engineers of Reactors and provide sufficient responsibility and authority in order that they can competently achieve their duties of supervision of safe operation and,*

- *Ensure thorough consistency that actual work shall be conducted in compliance with the official work procedures etc. for operation management of reactor facilities among the each organization of the licensee and contractors in charge of maintenance and repair work.*

- *"Periodic Inspection"*

The Periodic Inspection of nuclear power generation facilities (excluding those under decommissioning) is conducted periodically in order to prevent accidents and failures or propagation thereof for electric facilities that are provided for power generation, such as nuclear reactors and associated facilities and steam turbine facilities. It is an obligation to implement the inspection in accordance with the rules defined in Article 54 of the Electricity Utilities Industry Law.

As stated in the law, the Periodic Inspection is conducted in shutdown condition at the interval not to exceed one year plus 13 months for steam turbines and 13 months for other electric facilities (i.e., nuclear reactors and associated facilities). During the Periodic Inspection it shall be verified that these facilities are maintained and operated in conformance to the application for Construction Plan approval and the Technical Standards defined by the METI ordinance.

At the Periodic Inspection, more than 60 items are the subjects for the inspection currently. The inspection is conducted aiming at the following points in order to ensure the integrity of each facility;

1) *Conformance to the content of the Construction Plan approved in accordance with Article 47 of the Electricity Utilities Industry Law and the content of the Construction Plan submitted in accordance with Article 48 of the said law,*

2) *Compliance to the Technical Standards relating to nuclear equipment,*

3) *Actions to prevent recurrence of trouble which occurred in the past and,*

4) *Conformance to the content of the Reactor Establishment in accordance with the Reactor Regulation Law.*

- *"Periodic Licensee's Check and Audit Of Licensee's Periodic Check System"*

Although licensees have voluntarily verified conformance to Technical Standards of the nuclear power generation facilities heretofore, by the amendment of the Electricity Utilities Industry Law in 2003, this action was defined as the "Periodic Licensee's Check" of licensees which are subject to the regulatory body's periodical and mandatory check since then. Specifically, JNES examines the implementing system of this Periodic Licensee's Check by reviewing documents and witnessing from the standpoint of the inspecting organization, inspection methods, schedule control, recordkeeping, control of contractors, and appropriateness in education and training. This is called "Audit of Licensee's Periodic Check System." The national government establishes the "Evaluation Committee on Audit of Licensee's Periodic Check System for Nuclear Power Stations (hereinafter referred to as the "Evaluation Committee")" in NISA. The Evaluation Committee performs a comprehensive evaluation of the implementing system of the licensee related to Periodic Licensee's Check referring to the results of the Audit of Licensee's Periodic Check System performed by NISA, and officially announces the evaluation results so that the licensee is encouraged to establish the implementing system to conduct the Periodic Licensee's Check appropriately, making continuous improvement.

The "Periodic Inspection" is the inspection especially important to the facilities and for the equipment to ensure the safety of nuclear power generation facilities that the national government has performed heretofore. Specifically, it is an inspection of the equipment as an important safety function *among the "Periodic Licensee's Check" performed by the licensee voluntarily. It verifies appropriateness of inspection procedures, inspectors and judgment of results utilizing quality assurance standards while paying attention to the process of the Periodic Licensee's Check,* by the national government and JNES attending to or verifying the record.

- *"Operational Safety Inspection"*

It is required to establish, implement, and continuously assess and improve the quality assurance program for licensees' operational safety activities, and to define such quality assurance in the Operational Safety Program.

Thereby, the following two matters are positioned comprehensively as implementation of the quality assurance;

1) *Maintenance management that defines inspection methods etc. of facilities and equipment, and operation management that defines limitations during operation etc., and,*

2) *Operational safety activities that licensees perform for ensuring the safety of reactor facilities, such as fuel management, radioactive waste management, radiation management, and emergency measures.*

The national government verifies the appropriateness of the quality assurance program, the situation of implementation, evaluation, and corrective actions through the Operational Safety Inspection that inspects such an observance status of Operational Safety Program. The Operational Safety Inspection was introduced in 2000 and it is conducted 4 times per year for about three weeks of duration.

- *"Periodic Safety Review"*

Licensees are also obliged to perform the so-called "Periodic Safety Review" every ten years, which reviews the situation of the operational safety activities and the situation of reflection of the newest technical knowledge to the operational safety activities.

At the same time, as measures for aging management, licensees are obliged to perform technical evaluation of aged deteriorations no later than 30 years after commissioning and to establish a ten-year (in length) term maintenance plan based on the evaluation.

The national government verifies through the Operational Safety Inspection whether the operational safety activities are appropriately performed reflecting the result of the Periodic Safety Review. Moreover, the implementation of the long-term maintenance plan is verified at the Operational Safety Inspection, the Periodic Inspection, and the Audit of Licensee's Periodic Check System (see Section 14.3 for details).

Concerning the aging management, establishment of a technical assessment and long-term maintenance plan for reactor facilities had both been required as part of the licensee's voluntary efforts. By amendment of the related ministerial orders in October 2003, this has been shifted to the mandatory duty of licensees and since January 2006 reporting of the results to NISA has been required, while NISA verifies the adequacy of the above assessment.

- *"Resident inspectors and the on-site inspection"*

Nuclear Safety Inspectors reside in each power station permanently and they make inspection tours to observe the status of the Operational Safety Program every day, as well as perform the Operational Safety Inspections 4 times per year in accordance with the Reactor Regulation Law. Moreover, as the result of the above-mentioned "Comprehensive Check," it is more encouraged that each Nuclear Safety Inspector's Office is fully informed of and shared the findings by the Nuclear Safety Inspector checking the safety of reactor facilities through the free access within the site.

In addition, an on-the-spot inspection by the personnel of NISA can be performed at any time when the Minister of METI deems necessary. During the on-the-spot inspection, NISA inspector can inspect documents, records and other objects and question relevant persons through the on-the spot inspection at the licensee's offices.

- *"Chief Reactor Engineer and Responsible Operator"*

The Chief Reactor Engineers allocated to each nuclear reactor by the licensee need to have their qualification authorized by the national examination and the appointment or dismissal of them needs to be notified to NISA. The Chief Reactor Engineers can offer their opinion to the superintendents of the plants when they recognize it is necessary for the safe operation, can give advice or recommendation to respective duty positions, and can take part in establishing plans for safe operation.

Responsible Operator is designated by the licensee and allocated for each nuclear reactor. The mission of the person responsible for operation is to perform the monitoring of the overall operation and the command and supervision of operators. He understands the situation of operations and the present condition of the safe operation by periodical patrol to the premises.

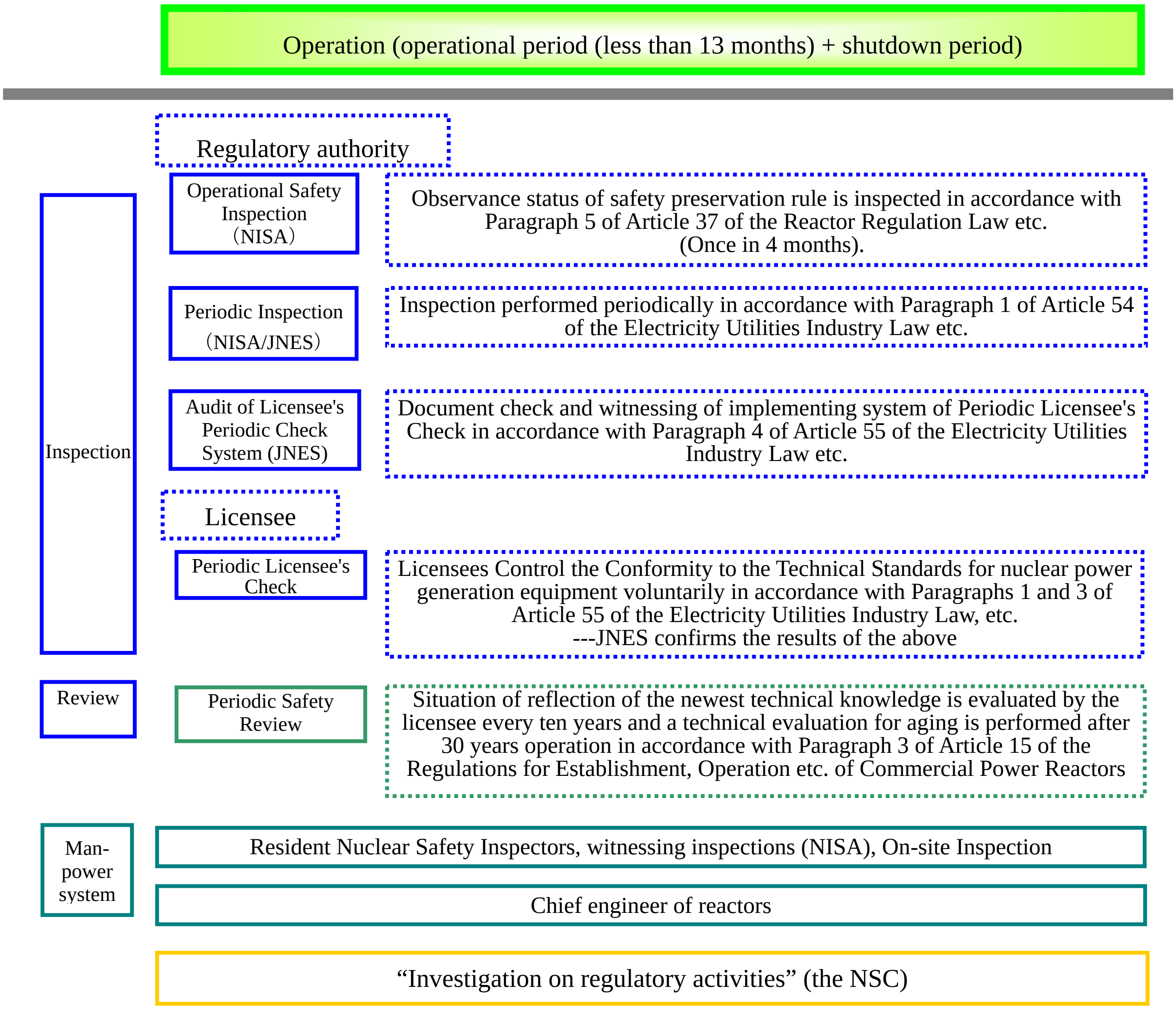
The documentation developed and kept by the licensee should include the records relating to fuel assemblies, inspection of the nuclear reactor, operation, radiation management, maintenance, abnormalities and accidents, and whether it is in accordance with the Reactor Regulation Law. Moreover, subjects, methods, results, etc. of the inspection should be recorded as a result of the Periodic Licensee's Check in accordance with the Electricity Utilities Industry Law.

- *"Cooperation in the investigation on regulatory activity performed by the NSC"*

Licensees are obliged to cooperate with the NSC when it conducts investigation by the law concerning with the reports by NISA which are relating to the regulatory approval and inspection and so on.

The overall view of the regulatory activities mentioned above is illustrated below.

Overall View of the Regulatory Activities for Safe Operation



(2) Pre-History Leading to the Present Inspection System

History leading to the present inspection system is described below.

- *The "first meeting of the Task Force on the Inspection System was held" in February 2002.*

"Task Force on the Inspection System" was established in the Nuclear and Industrial Safety Subcommittee of the Advisory Committee for Natural Resources and Energy in February 2002, and the study started. The result was summarized in June 2002 as a report named "On the Reexamination of the Inspection System" of the Nuclear and Industrial Safety Subcommittee (hereinafter referred to as the "Nuclear and Industrial Safety Subcommittee Report").

As a fundamental policy for reexamination of the inspection systems, a new philosophy was introduced to move from the conventional prescriptive approach "to confirm the integrity of the *facilities through the pre-determined manner which are pre-selected" to more performance-based approach where "installation methods of facilities and licensee's safety activities as a whole will be checked by introducing unannounced inspection".*

- *The following actions were recognized as necessary in order to improve the effectiveness of the inspection;*

  - *(i) to enhance quality assurance activities,*
  - *(ii) to introduce an unannounced inspection,*
  - *(iii) to utilize quantitative risk assessment,*
  - *(iv) to apply the performance-base inspection,*
  - *(v) to establish criteria and standards as needed,*
  - *(vi) to be flexible in applying legal measures and,*
  - *(vii) to utilize lessons learned from minor troubles.*

(viii) "Action taken after an inspection data falsification scandal by TEPCO." in August 2002

The inspection data falsification by TEPCO was uncovered and study was started to reform the current inspection system at the Nuclear and Industrial Safety Subcommittee, NISA, etc.

(ix) "Revision of legislations" in October 2003

As the results of the study mentioned above, the Reactor Regulation Law was revised and the present framework of the inspection system was introduced. The inspection system is assuming that operational safety activities including maintenance management by licensees are defined as activities to implement the quality assurance, while the Operational Safety Inspection, the Periodic Inspection, and the Audit of Licensee's Periodic Check System are all instruments of the regulatory authorities of the national government etc. to check how this system is properly functioning.

In addition, the system for fitness-for-service assessment for cracks was introduced into the regulations at that time. Initially, there had been no definite maintenance standard for cracks in Japan. Therefore, any crack etc. discovered had to be repaired. Introduction of the system for fitness-for-service assessment enabled in-service operation without repairing such cracks etc. within a certain period. Thereby, licensees could make an appropriate and rational maintenance plan. Reduction of the collective dose by optimizing the number of work projects resulting from the introduction of fitness-for-service will be attempted.

(3) Reexamination of the Present Inspection System

The inspection system mentioned above has become much more established nowadays as a result of the accumulation of experience obtained in the past two years or so. However, measures for aging *management are in need of further enhancement, taking into consideration;*

1) the present situation of conducting the inspection system after two years passage
2) the causal analysis of the secondary system pipe rupture accident at the Mihama Unit 3 which occurred in August 2004,
3) projected increase of number of nuclear power stations under long term operation.

Thereby, the "Task Force on the Inspection System" was resumed in November 2005 and a report of the Task Force on the Inspection of Nuclear Safety ("Improvement in the Inspection System for Nuclear Power Generation Facilities") was issued in September 2006. It pointed out the following issues, and led to the understanding that these issues required immediate improvement;

a. Enhancement of maintenance activities for individual nuclear installation

As for the measures for aging management, licensees are obliged to assess each nuclear installation and the national government verifies the adequacy of their assessments. In the Periodic Licensee's Check or the Periodic Inspection, refined inspections reflecting the individual features of each nuclear installation are not easy, since only the routine type inspection system has been available. Since conditions of equipment and operation, such as plant operating year, design of nuclear installation, operating history such as incidents and troubles, etc., and licensees' management system including contractors' vary from one nuclear installation to another, it is considered more desirable, in order to take thorough measures for maintenance activities for a nuclear installation, to ask each licensee to understand the operational status of each nuclear installation more individually and perform the maintenance activities based on this understanding even before aging measures have to be introduced.

In order to enhance the measures for aging management, it is necessary to control deterioration due to aging in a proper manner. It would be effective to introduce new monitoring and evaluation technologies (including on-power monitoring) in order to;
-understand the progress of deterioration based on scientific knowledge and conduct lifetime prediction and,
-inspect and repair at an appropriate time by monitoring conditions of equipment and systems based on the condition-based maintenance. For this purpose, it is necessary to monitor the situation of deterioration of the equipment and systems of every nuclear installation continuously or regularly according to a fixed time period, and to evaluate the trend of deterioration.

Therefore, introduction of an inspection system that asks each licensee to daily enhance the maintenance activities based on the features of each nuclear installation is a new attempt.

b. Further thoroughness to ensure safety in the operational safety activities

The operational safety activities for nuclear installations include those performed during operation time, such as routine plant patrol, condition monitoring of equipment and surveillance test of important safety related equipment. Therefore, the regulator should conduct inspections exactly when these activities are being conducted for verifying appropriateness of the operational safety activities.

Items related to the operational management occupies seventy percent of all the serious non-conformances in relation to the operational safety management (conditions of not conforming to the requirements of quality assurance) that were revealed at the Operational Safety Inspection and were ordered to be remedied immediately after the introduction of new inspection system in October 2003. Therefore operational management represents the essential operational safety activities along with maintenance management.

The Nuclear Safety Inspector conducts the safe-operation survey when the inspector is not conducting the Operational Safety Inspection. There are many cases where comparatively minor non-conformances are identified at the safe-operation survey and remedial actions by the licensee are to be confirmed at the subsequent Operational Safety Inspection. Taking such cases into consideration, it is necessary to conduct inspections during the operation time as well in order that non-conformances are remedied at an early stage and measures to be taken avoiding an accident or trouble by means of performing witness inspections etc. and confirming the appropriateness of the operational safety activities of licensees. Introduction of an inspection system during operation and shutdown, which further ensures licensees' safety, is also a new attempt.

It is also necessary to check if the Operational Safety Inspection is efficiently and effectively functioning and to eliminate duplication in practices by utilizing the results of the Audit of Licensee's Periodic Check System for the quality assurance of the Periodic Licensee's Check.

c. Thoroughness to remedy non-conformances by licensees

Although the safety level of nuclear installations in Japan is satisfactory in general, the ratio of accidents and troubles resulting from human error to the total events does not present downward tendency. Some human errors have organizational factors or show deterioration of safety culture and/or organization culture as their background.

Accidents and troubles that occurred due to organizational factors or the deterioration of safety culture and/or organization culture have been increasing after the latter half of the 1990s. It was pointed out that deterioration of the safety culture of the licensee served as a background to the secondary system pipe rupture accident of the Mihama Unit 3.

In order to raise the safety level of nuclear installations further, thorough challenge should be made to eliminate non-conformances due to those factors.

(4) Future Direction for Improving the Inspection System

In order to pursue the above-mentioned three attempts, namely, to enhance individual maintenance activities for each nuclear installation, to ensure further safety in operational safety activities, and to eliminate non-conformances thoroughly by licensees, it is necessary to improve the inspection system as follows and this study by the various organizations is underway with an aim to start the new system in April 2008;

(i) Introduction of the inspection system as the integrated part of the operational safety activities

based on the "overall maintenance plan"

a. Further comprehensive "overall maintenance plan"

This "overall maintenance plan" is defined as part of the quality assurance program and covers the maintenance activities as a whole performed by licensees. It is essential to define and establish the organization to implement the maintenance activities, scope of the components and structure subject to the maintenance activities and implementing program

b. Inspection for the purpose of verifying the situation of the "overall maintenance plan"

This inspection is required to verify that the licensee is actually performing the maintenance activities appropriately in accordance with the "overall maintenance plan". The "overall maintenance plan" includes maintenance activities during plant operation as well as those during shutdown. For this reason, the inspection performed by the national government mentioned above also needs to include the inspection during plant operation in addition to that of during shutdown. Moreover, a system is needed that enables the national government to ask licensees for necessary alternation or improvement of the "overall maintenance plan" when improper acts etc. are identified as a result of inspecting the licensee's implementing situation of the "overall maintenance plan."

(ii) Introduction of an inspection system focused on the important operational actions to ensure safety

a. Inspection system focused on the important operational actions to ensure safety

For introduction of an inspection system to thoroughly ensure safety in the licensees' operational safety activities, it is necessary to identify the important operational actions to ensure safety whether during operation or during shutdown.

b. More efficient and effective inspection system

As a result of performing the inspection also during operation, which has been periodically conducted as the Operational Safety Inspection until now, introduction of the new inspection system focused on the important operational actions to ensure safety during operation and shutdown will enable to avoid the duplication of periods for Periodic Inspection and Operational Safety Inspection,

(iii) Development of a guideline for root cause analysis (refer to Section 10.2)

(5) Actual examples of the records of the Inspection System etc.

Actual examples of the past records of the Operational Safety Inspection, the Audit of Licensee's Periodic Check System and the fitness-for-service assessment are shown in the following:

(i) Past records of the "Operational Safety Inspection"

Two examples of the Operational Safety Inspection conducted in the 2006 fiscal year are shown in the following. At the early stage of the Operational Safety Inspection, it was likely to detect more formalistic deficiencies related to the quality management system but now more quality assurance related aspects are emphasized.

(Example 1) A domestic BWR nuclear power station
June 5 (Mon.) to June 23 (Fri.)
• Situation of management review and internal audit (inspection at the head office) • Situation of maintenance management • Situation of quality assurance activities including procurement controls of reactor feedwater flowmeter etc. (including inspection at the head office) • Surveillance test (manual start-up test of standby gas treatment system); an unannounced inspection • Situation of improvement measures relating to the past violation
• Situation of management review and internal audit (inspection at the head office) • Situation of maintenance management • Situation of quality assurance activities including procurement controls for reactor feedwater flowmeter (including inspection at the head office)
In this Operational Safety Inspection, the “situation of management review and internal audit (inspection at the head office),” “situation of maintenance management,” “situation of quality assurance activities including procurement controls f of reactor feedwater flowmeter etc. (including inspection at the head office),” etc. were inspected. Since the inspection found out items to be observed concerning the plan for piping thickness control and competence of maintenance management personnel for the “situation of maintenance management” and the action plan for the reactor feedwater flowmeter for the “situation of quality assurance activities including procurement controls for reactor feedwater flowmeter (including inspection at the head office),” in the future the improvements should be checked everyday by the patrol, the Operational Safety Inspection, etc. In addition, concerning the procurement control for reactor facilities including the feedwater flowmeter, it was found out that; - effort to avoid the recurrence was in place but still at planning stage and specific approach should be still needed. - there are still some room for improvement in terms of avoiding the reoccurrences and control the procurement methods. So the monitoring of the implementation situation to avoid recurrence will be continued. Check of the results of past remedial actions (i.e., “duties of personnel in charge of civil engineering and the building of a power station” and “corrective actions concerning water chemistry control”), showed that appropriate improvements had been achieved, respectively. Concerning the situation of routine operations management during the period of the Operational Safety Inspection, by interviewing the licensee for the situation of operational management of facilities, check of the record of operations, patrol of reactor facilities, witnessing of the surveillance test (manual start-up test of standby gas treatment system), etc. presented no special problems. From the above results, summarizing this Operational Safety Inspection, it is judged that the operational safety activities concerning the selected inspection items are acceptable in general.

(Example 2) A domestic PWR nuclear power station
May 29 (Mon.) to June 16 (Fri.)
• Situation of fuel management of Unit 2 • Situation of plant patrol • Situation of management review • Situation of work management • Situation of education on operational safety: an unannounced inspection • Witnessing surveillance tests (turbine driven auxiliary feed pump start-up test, etc. of Unit 1 and 2)
• Situation of fuel management of Unit 2 • Situation of plant patrol
In this Operational Safety Inspection, the "situation of fuel management of Unit 2," "situation of plant patrol," "situation of management review," "situation of work management," etc. were inspected. As the results of the inspection items for remedy were found in the following; - unclear descriptions in the documentations - insufficient control of the component parts in stock in relation to the procedures of "situation of plant patrol," "situation of work management," and "situation of education on operational safety," Check should be done on the corrective action through routine patrol, the Operational Safety Inspections, etc. Concerning the situation of routine operations management during the period of the Operational Safety Inspection, by interviewing the licensee for the situation of operational management of facilities, check of the record of operations, patrol of reactor facilities, witnessing of the surveillance test (turbine driven auxiliary feed pump start-up test, etc. of Unit 1 and 2), etc. presented no special problems. From the above results, summarizing this Operational Safety Inspection, it is judged that the operational safety activities concerning the selected inspection items are acceptable in general.

(ii) Past records of the "Audit of Licensee's Periodic Check System"

The actual application of the Audit of Licensee's Periodic Check System is shown in the following:

JNES selects items of the actual audit from the Periodic Licensee's Check and performs the audit by document check and witness. The basic idea of the sampling is as follows;

(a) *Number of items subject to the audit shall be around ten percent of that of the Periodic Licensee's Check items, and shall be dependent on the preceding evaluation result,*

(b) *Items for the audit shall be selected without any prejudice taking into consideration the types of inspection, equipment and commercial contract within the scope of the Periodic Licensee's Check .*

(c) *The preceding audit result at the place of business shall be reflected. Especially, if there was any indicated item or finding in the preceding audit, its improvement should be checked and,*

(d) *If the preceding audit identified the items for further monitoring, those items should be checked with priority.*

The national government evaluates the results of the Audit of Licensee's Periodic Check System being informed from JNES, indicates the rating according to the following three categories and informs the licensee of the evaluation results;

A: *The implementing organization of the licensee is capable of meeting the requirement for the Periodic Licensee's Check voluntarily and appropriately,*

B: *The implementing organization of the licensee is capable of meeting the requirement for the Periodic Licensee's Check voluntarily and appropriately despite room for improvement and*

C: *The implementing organization of the licensee is capable of meeting the requirement for the Periodic Licensee's Check voluntarily despite considerable room for further improvement.*

Incentives based on evaluation results

An incentive type control where selection of items for next audit is dependent of the rating of the audit of the preceding year has been conducted in order to encourages licensee's effort for safety and ensure transparency and reliability of the Periodic Licensee's Check.

Past records of the audit after this audit system was started are shown below.

Fiscal Year	Number of Audit	Audit Result		
		A	B	C
2003	12	2	4	5
2004	37	1	28	7
2005	35	4	26	2
2006	24	–	2	–
Total	108	7	60	14

An example result of the Audit of Licensee's Periodic Check System of a specific power station in 2006 fiscal year is shown below.

Example Applicant of Audit of Licensee's Periodic Check System	President, A domestic Electric Power Co., Inc. (Application date: x/x/2005 Application Number: xxxxx No. xxx)
Items subject to audit	Periodic licensee's check in xx Periodic Inspection for xxxx Nuclear Power Station, Unit x
Audit of Licensee's Periodic Check System (NISA)	1. Audit period x/xx/2005 to x/x/2005
	2. Notification date of results of Audit of Licensee's Periodic Check System x/x/2005 (Notification number: 05xxxx-xxxx)
	3. Summary of audit results According to JNES's notification and description of the results of the Audit of Licensee's Periodic Check System, it identified no significant non-conformity, but identified three items determined necessary to be improved. Corrective actions for two of the three items were confirmed during the audit period. Any corrective action for another one could not be confirmed during the audit period, and it shall be followed up in the next Audit of Licensee's Periodic Check System for the said power station. Concerning follow-up of the four items that had been determined to be improved in the Audit of Licensee's Periodic Check System for Unit x, Unit x, Unit x and Unit x which had been subject to the precedent audit, corrective actions for two items were confirmed. Effort for the corrective actions for the other two items effort was visible but could not be confirmed because there was no inspection applied this time for the unit. Those shall be followed up in the next Audit of Licensee's Periodic Check System. Consequently JNES judges that the quality management system is in general functioning well and the Periodic Licensee' Check is conducted with the voluntary and appropriate system. Since the corrective actions for the three items which were notified for improvement during the audit of this unit and the units subjected to the precedent audit could not be confirmed this time, further investigation and effort for the improvement are encouraged steadily and addressed continuously.
	4. Audit item Thirteen items of document audit and actual spot audit (inspection of (1) monitoring system, (2) steam turbine facility, (3) core-internal, etc.)
Evaluation (NISA)	1. Audit result: B
	2. Notification date of the evaluation xx/x/2006 (Notification number xxxxxx Nuclear No.x

	3. Reason for the audit result (result and grounds) NISA carefully examined the audit results of the notification concerned and the explanation by JNES, and determines that three items, which were determined necessary to be improved but could not be verified for adequacy of the corrective action, needs to be monitored continuously. Consequently, NISA determines that the implementing organization of the licensee is capable of meeting the requirement for the Periodic Licensee's Check voluntarily and appropriately despite room for improvement.
	4. Situation of holding of the evaluation committee Hearing of the explanations concerning the audit result, Q&A, x/xx/2006 Study on the evaluation result, x/xx/2006
	5. Special notes for rating N/A
Others	

(iii) Historical experiences of fitness-for-service assessment

Since the assessment system for fitness-for-service was introduced, cracks were found on the shroud etc. of total of 15 nuclear power generation units, and the fitness-for-service assessments have been performed. Situation of crack propagation is monitored and whether these meet the safety level or not is verified at the time of the Periodic Inspection performed about every 13 months. (As of October 2006)

Licensee	Power Station	Unit	Notification Date	Part Subject to Evaluation
Chubu Electric Power Co., Inc.	Hamaoka	Unit 4	12/21/2004	(1) Crack on weld of the shroud support ring (2) Crack on the shroud lower ring and lower shell
Tohoku Electric Power Co., Inc.	Onagawa	Unit 1	01/06/2005	Crack on the shroud middle ring and lower ring
The Chugoku Electric Power Co., Inc.	Shimane	Unit 2	02/09/2005	Crack on the PLR piping
Shikoku Electric Power Co., Inc.	Ikata	Unit 1	03/01/2005	Micro-crack on the surface of the reactor vessel inlet nozzle
Tokyo Electric Power Co., Inc.	Kashiwazaki Kariwa	Unit 3	04/13/2005	Crack near outside weld-line of the shroud middle shell and lower ring
The Chugoku Electric Power Co., Inc.	Shimane	Unit 2	04/13/2005	Crack on inside weld-line of the shroud middle shell
Tohoku Electric Power Co., Inc.	Onagawa	Unit 2	05/27/2005	Crack inside the shroud support ring
Chubu Electric Power Co., Inc.	Hamaoka	Unit 3	05/24/2005	(1) Crack on the shroud support ring (2) Crack on the shroud support cylinder and weld on the support leg

The Japan Atomic Power Company Co.	Tokai		07/13/2005	Crack on the vertical weld-line on the shroud support cylinder
The Chugoku Electric Power Co., Inc.	Shimane	Unit 1	07/20/2005	Crack on the PLR piping
Tokyo Electric Power Co., Inc.	Fukushima Daini	Unit 3	08/18/2005 04/19/2006	Crack on the PLR piping
Tokyo Electric Power Co., Inc.	Kashiwazaki Kariwa	Unit 2	11/04/2005	Crack on the weld-line between the shroud middle shell and the shroud support ring
Shikoku Electric Power Co., Inc.	Ikata	Unit 2	02/26/2006	Micro-crack on the surface of the reactor vessel inlet nozzle
Tokyo Electric Power Co., Inc.	Kashiwazaki Kariwa	Unit 1	04/19/2006	Crack on the PLR piping
Tokyo Electric Power Co., Inc.	Kashiwazaki Kariwa	Unit 3	04/19/2006	Crack on the PLR piping

(PLR: Primary Loop Recirculation)

19.4 Response to Accidents and Anticipated Operational Occurrences

Licensees are required to include the items related to “operation of nuclear reactor facility” in the Operational Safety Program.

Those include the operational procedures for accidents and anticipated operational occurrences (abnormal events) as well as normal operation are described so as to cope with any incidents or abnormal events smoothly.

In the procedures of “measures for any abnormal events,” the following matters are included:

- *Recognition of the situation;*
- *Elimination of the cause;*
- *Emergency measures;*
- *Measure after reactor automatic scram and,*
- *Manual startup of an emergency AC power supply and a gas treatment system.*

Moreover, licensees are required to prepare “emergency measures” in the operation of a nuclear reactor facility stipulated by the Reactor Regulation Law.

As “emergency measures,” the licensee defines the following in the Operational Safety Program:

- *Establishment of the nuclear emergency preparedness organization in accordance with the requirements of the Special Law of Emergency Preparedness for Nuclear Disaster;*
- *Preparation of resources necessary for nuclear emergency preparedness;*
- *Maintenance of the communication system among the related parties;*

- *Implementation of nuclear emergency exercises;*
- *Official announcement of the emergency system and,*
- *Clearance of the emergency preparedness organization, etc.*

The details of emergency preparedness are described in Article 16.

19.5 Engineering and Technical Support

In Japan, the reliability verification test and safety research programs on major components and equipment have been carried out to enhance the safety of nuclear installations. Promotion of these tests and research is shown in Section 14.8.

Main subjects of related research are shown below:

(1) Study on aging of light water reactors;

- *Survey of technology related to measures for aging management (accumulation of technical knowledge, aging evaluation technology for cable, etc.),*
- *Demonstration of the nondestructive inspection technology of component material for nuclear reactors as technical measures for aging management, and,*
- *SCC crack-growth evaluation technology etc. as measures for aging management (international information sharing in OECD/NEA).*

(2) Study on probabilistic safety assessment for nuclear installations etc. and,

- *Study on utilization method of risk information and,*
- *Study on Nuclear Emergency Preparedness.*

(3) Study on safety, such as seismic safety of nuclear installations.

- *Clarification study on rupture process of wall-thinned pipe under seismic load*

19.6 Reporting of Incidents

Reactor Regulation Law and the Electricity Utilities Industry Law require licensees to report the situation and measures taken to the incidents or failures occurred in nuclear installations to NISA. The reporting criteria prescribed in these laws are shown in Table 19-2.

Licensees are making efforts to perform feedback of the lessons learned from the situation and measures taken to these incidents, and the corrective actions which were derived by root cause analysis to other nuclear installations.

As the result of the "Comprehensive Check" referred in Article 6, incidents related to the move of the *control rod when it is not allowed to be operated is now the subject for the reporting to the national government.*

The frequency of unplanned reactor shutdowns per year of nuclear installations in Japan is around 0.2 times per reactor-year in recent years and is well below the world average.

The International Nuclear Event Scale (INES) was introduced in August 1992 to assess events that occurred in commercial power reactor facilities. INES results in Japan are shown in Annex 2.5.

Of the past cases identified by the "Comprehensive Check," those that NISA rated as 1 or over to the electric utilities of Japan are as follows:

Cases identified by the Comprehensive Check relating to power generation facilities

Event Date	*Facility Name*	*Subject*	*INES*
December 24, 1996	*The Japan Atomic Power Company Tsuruga Power Station Unit 2*	*Leak from piping of the chemical volume control system etc.*	*1*
June 11, 1998	*Tohoku Electric Power Co., Inc. Onagawa Nuclear Power Station Unit 1*	*Reactor automatic shutdown due to high neutron flux during reactor shutdown operation*	*1*
June 18, 1998	*Hokuriku Electric Power Co. Shika Nuclear Power Station Unit 1*	*Criticality accident*	*2*

19.7 Collection, Utilization and Sharing of Operating Experience Information

NISA makes a public news release and reports to the NSC incidents or failures upon immediate receipt of the information from the licensees. NISA also makes a public news release and reports to the NSC the causes and recurrence-preventive measures when the investigation is completed. The NSC points out issues on the content of the report when necessary. *The NSC has established the Special Committee on Analysis and Evaluation of Nuclear Accidents and Failures and investigated and reviewed in and outside Japan. In March 2007 this committee has summarized the guidance to use the incident and failure information of nuclear installations.*

NISA assesses each incident or failure in detail to abstract the lessons learned with respect to the safety, being advised by subcommittee members of the Advisory Committee on Nuclear and Industrial Safety, who are experts on operation management, inspection and radiation control.

JNES has the system to collect and analyze safety information in and outside Japan. JNES provides collected safety information and the results of analysis for NISA for quick sharing with NISA. JNES and NISA have jointly established the "Safety Information Review Meeting" to evaluate and to take adequate regulatory measures. The "Safety Information Review Meeting" is held periodically.

In regard to the international information exchange by NISA and JNES, a mechanism has been established to share the information on nuclear incidents and failures with the IAEA and OECD/NEA as well as in the bilateral corporations with China, France, Korea, Sweden and the United States.

Meanwhile, the licensees established the specialized organization named, "Japan Nuclear Technology Institute" (JANTI), in March 15, 2005 in order to develop the technical infrastructure and promote the voluntary operational safety activity for the purpose of contributing to the activation of nuclear power industry. JANTI unified, reorganized and inherited the activities conducted by the Nuclear Information Center (NIC) of CRIEPI and the activities of the Nuclear Safety Network (NS Net) conducted by the research organizations and nuclear industry. JANTI collects and analyses domestic and overseas operational experience information. Concerning the safety information of domestic nuclear installations, they developed a system named "NUCIA" that is a nuclear information publication library, with which information including minor events can be shared all over Japan, and it is posted on the Internet site of JANTI. In May 2007, JANTI reexamined the criteria to register the incidents and events to NUCIA.

Overview of the "NUCIA" is shown in Fig. 19-1.

Moreover, the licensees and JANTI perform overseas information exchange through the Institute Of Nuclear Power Operation (INPO) and the World Association of Nuclear Operators (WANO) Tokyo Center. Furthermore, each licensee utilizes individual agreements on information exchange with utilities and manufacturers in France, Germany and the United States.

There are many feedback of operating experiences by licensees, which are reflected in preventive maintenance and planned repair and replacement of parts. Examples for BWR are replacements of the core shroud and the in-core monitoring housing etc. An example for PWR is replacement of the upper head of reactor vessel.

At manufacturer side as well, for the purpose of information, two groups are formed, firstly "Japan BWR Owners Group" consisting of the electric power companies of Tohoku, Tokyo, Chubu, Hokuriku, Chugoku, JAPC and J-POWER and Toshiba/Hitachi and secondly "Japan PWR Owners Group" consisting of the electric power companies of Hokkaido, the Kansai, Shikoku, Kyushu and JAPC and Mitsubishi Heavy Industry and Mitsubishi Electric.

19.8 Management of Spent Fuel and Radioactive Waste

Details are described in the Second National Report of Japan (October 2005) for the "Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management".

Table 19-1 (Part 1) Items of Limiting Conditions for Operation (BWR)

System	Item of limiting conditions for operation
Reactivity control system	Shutdown margin, reactivity monitoring, control rod motion monitoring, control rod scram time, control rod operation, boron water injection system
Power distribution	Reactor thermal limit, reactor thermal power and core flow
Control & Instrumentation	Instrument and control equipment
Reactor coolant system	Reactor re-circulation pump, jet pump, main steam relief and safety valve, reactor coolant leak rate, system pressure monitoring of the emergency core cooling system and reactor isolation cooling system, concentration of iodine 131 in reactor coolant, reactor shutdown cooling system, limit of temperature & temperature change rate limit of primary coolant, reactor pressure, *reactor water level when moving a fuel or control rod*
Emergency core cooling system	Emergency core cooling system, reactor core isolation cooling system
Reactor containment vessel system	Main steam isolation valve, reactor containment vessel & isolation valve, vacuum break valve from suppression chamber to drywell, average temperature of suppression pool, flammability control system, oxygen concentration in containment vessel, reactor building, reactor building heating and ventilation isolation valve, standby gas treatment system
Plant system	Cooling system and cooling sea water system for residual heat removal system, emergency diesel generator cooling system, cooling system and cooling sea water system for diesel generator of high pressure core spray system, water level & temperature of spent fuel pool, central control room heating and ventilation system
Emergency power supply system	Offsite power supply system, emergency diesel generator, emergency diesel fuel, dc power supply, station power system
Others	Withdrawal of single control rod during reactor shutdown, removal of single control rod drive mechanism, inspection with withdrawal of multiple control rods, in-service leak-rate or hydrostatic test, inspection with switching of reactor mode, *inspection causing the temperature increase in the reactor*

Table 19-1 (Part 2) Items of Limiting Conditions for Operation (PWR)

System	Item of limiting conditions for operation
Reactivity control system	Shutdown margin, critical boron concentration, moderator temperature coefficient, control rod motion function, control rod insertion limits, control rod position indication, physics tests, chemical and volume control system (function of boron concentration)
Power distribution	Reactor thermal power limit, heat flux hot channel factor, nuclear enthalpy rise hot channel factor, axial neutron flux difference, quadrant power tilt ratio
Control & Instrumentation	Instrument and control equipment
Primary coolant system	DNB ratio, temperature & pressure and temperature change rate of primary coolant, primary coolant system, pressurizer, pressurizer safety valve, pressurizer relief valve, low temperature over-pressurization protection, primary coolant leak rate, steam generator tube leak monitoring, leak monitoring to residual heat removal system, iodine 131 concentration in primary coolant
Emergency core cooling system	Accumulator tank, emergency core cooling system, refueling water storage tank, boron injection tank
Reactor containment	Reactor containment vessel, reactor containment vessel vacuum relief valve, reactor containment vessel spray system, annulus air cleanup system, annulus
Plant system	Main steam safety valve, main steam isolation valve, main feedwater isolation valve, main feedwater control valve, main feedwater bypass control valve, main steam relief valve, auxiliary feedwater system, condensate water tank, component cooling water system, sea water system for component cooling water system, emergency circulation system of control room, air cleanup system of safety auxiliary equipment room, air cleanup system of fuel handling building
Emergency power supply system	Offsite power supply, diesel generator, emergency diesel fuel & lubricating oil and starting air for emergency diesel generator, emergency dc power supply, station bus bar for emergency
Others	Boron concentration in primary coolant, water level of reactor cavity, reactor containment penetrations, water level & temperature of spent fuel pit, *pressure-proof and leak inspections of reactor coolant system, leak rate inspection of safety injection system check valve*

Table 19-2 Reporting Criteria of Incidents and Failures Provided in Legislations

Provision of the Reactor Regulation Law

1. When nuclear fuel material was stolen or lost**.**
2. When a reactor was shut down by failure of a reactor facility or when it became necessary to shut down a reactor during operation, or when reactor power output changes more than 5%, or when reactor power output change of more than 5% was required. Except when it was one of the following and the licensee announced officially about the situation of the concerned failure.
   i) When it occurs in the term of the Periodic Inspection provided in Article 54-1 of the Electricity Utilities Industry Law (Law No. 170, 1964) i.e. the failure in the equipments that the functional and operational conditions of the failed equipment cannot be checked under the reactor shutdown condition.
   ii) When the failure did not cause any deviation from the limit of operation (it is a requirement defined in the Operational Safety Program for operation of the nuclear reactor facility, and when it cause any deviation from the concerned conditions the measure that the licensee should take is also defined in the Operational Safety Program, the same in this paragraph), and there is no change observed related with the concerned failure, and when the licensee performs inspection of the failed equipment concerned.
   iii) When the reactor output change is required to follow the limit of operation.
3. When a licensee checked the equipment and structure important to the safety of the nuclear reactor facility provided by the Minister of METI (hereinafter called as "equipment etc. important to safety" in this paragraph as), and when concerned equipment etc. important to safety was considered that it does not satisfy the standard described in Article 9 or in Article 9-2 of the Ministerial Ordinance of Establishing Technical Standards for Nuclear Power Generation Equipment (Ordinance No. 62 of MITI, 1965), or when it was considered that it does not have function to ensure safety of the nuclear reactor facility.
4. When there was a failure of equipment etc. important to safety by the fire. Except the concerned failure was associated to the measure of fire extinguishing or prevention of the spread of fire.
5. Except for the preceding three items, when deviation from the limit of operation by the failure of a the nuclear reactor facility (except those minor troubles whose impact on operation of nuclear reactor is insignificant) was caused, or when the measure for the concerned deviation defined in the Operational Safety Program was not implemented at the time of deviation from the limit of operation.
6. When the failure of a reactor facility or occurrence of other undetermined situation had caused any trouble to discharge gaseous radioactive wastes through the ventilation facility or to discharge liquid radioactive wastes through the drainage facility.
7. When the concentration of radioactive materials in the air outside the environment monitoring area exceeds the allowable limit in the case of discharge of gaseous radioactive wastes through the ventilation facility.
8. When the concentration of radioactive materials in the water outside the environment monitoring area exceeds the allowable limit in the case of the discharge of liquid radioactive wastes through the drainage facility.
9. When nuclear fuel materials or materials contaminated with nuclear fuel materials (hereinafter referred to as "nuclear fuel materials etc.") leaked out of the controlled area.

10. When nuclear fuel materials etc. leaked within the controlled area associated to failure of a nuclear reactor facility or occurrence of other undetermined situation. Exceptions are the followings (except the case when new measures such as access control into the leakage-related place and key control have been taken or when the leaked substances have spread outside the controlled area):
    i) When revealed liquid nuclear fuel materials etc. did not spread out of the floodgate that is installed in the circumference of the equipment of the concerned leakage for prevention of leakage spread.
    ii) When the ventilation facility of the concerned area of the leakage was working properly at the time when gaseous nuclear fuel materials etc. leaked.
    iii) When the amount of radioactivity of the leaked nuclear fuel materials etc. is very little and when the degree of the leakage is minor.
11. When the person who enters into the controlled area suffered radiation exposure due to the failure of a nuclear reactor facility or occurrence of other undetermined situation, and when the effective dosage of concerned exposure exceeds or could exceeds five mSv for a personnel engaged in radiation work or 0.5 mSv for a person other than the personnel engaged in radiation work.
12. When the dosage of personnel engaged in radiation work exceeds or could exceed the allowable dose limit.
13. Other than those above items,_when persons are injured or could be injured in the nuclear reactor facility (except when the injury was other than radiation hazard and was not necessary for hospitalization).

Provision of the Electricity Utilities Industry Law

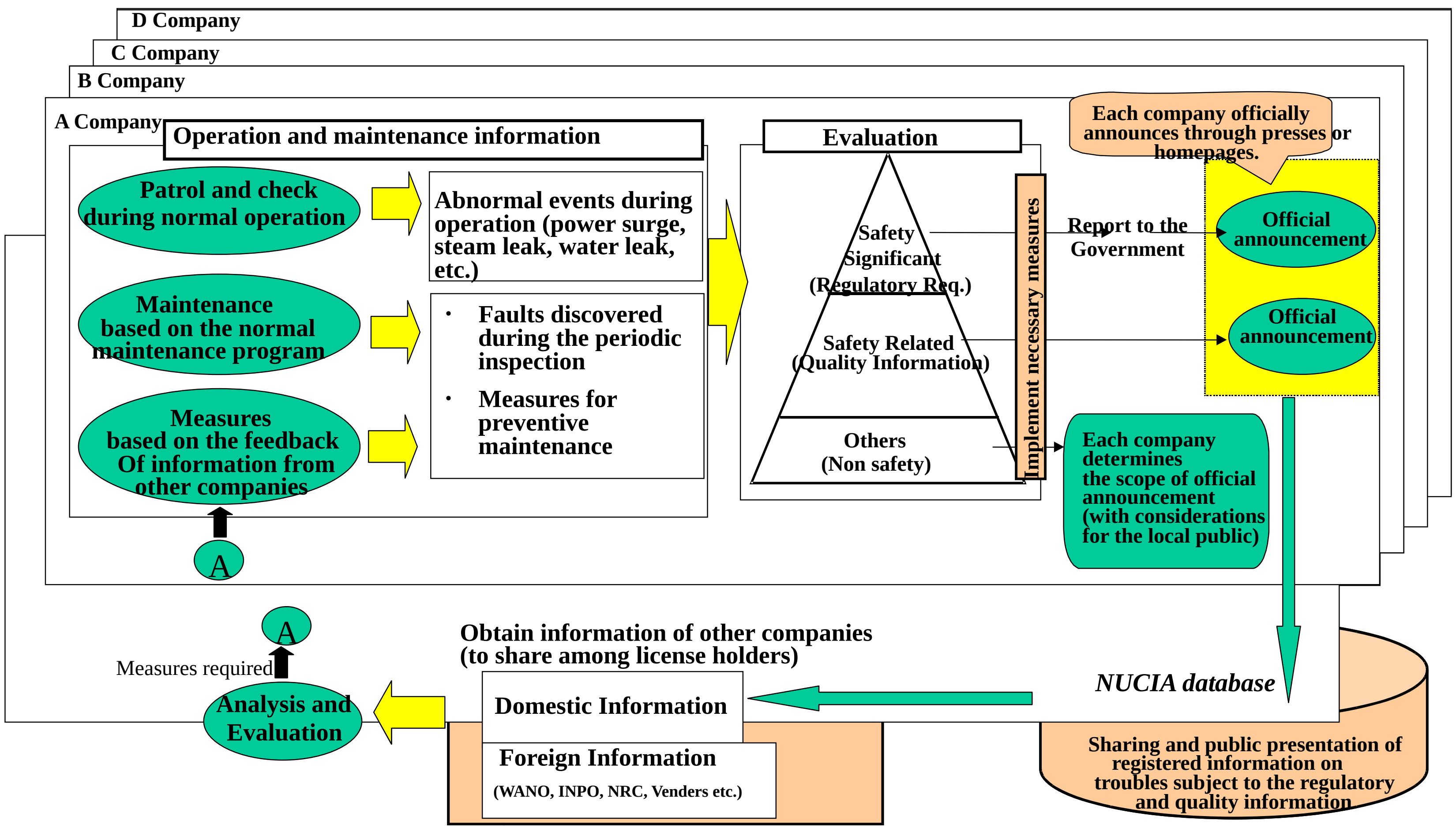
1. Fatal and injury accidents associated to electric shock, or breakdown of the electric structure of a nuclear power generation, or miss-operation, or omission of the necessary operation of the electric structure of a nuclear power generation (limited to the case of death or being hospitalized at hospital or clinic for treatment).
2. Electric fire accident (limited to the case of more than the partial destruction by fire. Except matters referred to in the previous item and the next to the fifth item).
3. Causing damage to public property due to failure of the electric structure of a nuclear power generation, or miss-operation, or omission of necessary operation of the electric structure, accident by which usage of road, park, school and other institution or structure were made impossible or accident which did influence socially (except referred to previous two items).
4. Breakdown accident of main electric structures (except referred to previous three items and the next item).
5. Incident that influenced other electric utility to force suspension of electric power supply of than 7,000 kW and less than 70,000 kW for more than one hour, or suspension of electric power supply more than 70,000 kW for more than ten minutes, associated to the breakdown of the electric structure of a nuclear power generation, or miss-operation, or omission of operation necessary for the electric structure of a nuclear power generation.

Table 19-3 Power station performing "rated thermal power operation"

Electric power company	*Nuclear installation*	*Date of Introduction*
Hokkaido Electric Power Co., Inc.	*Tomari Unit 1*	*06/18/2003*
	Tomari Unit 2	*02/21/2003*
Tohoku Electric Power Co., Inc.	*Higashidori Unit 1*	*–*
	Onagawa Unit 1	*08/11/2003*
	Onagawa Unit 2	*12/08/2003*
	Onagawa Unit 3	*05/06/2003*
Tokyo Electric Power Co., Inc.	*Fukushima Daiichi Unit 1*	*–*
	Fukushima Daiichi Unit 2	*–*
	Fukushima Daiichi Unit 3	*12/25/2003*
	Fukushima Daiichi Unit 4	*–*
	Fukushima Daiichi Unit 5	*12/25/2003*
	Fukushima Daiichi Unit 6	*02/22/2005*
	Fukushima Daini Unit 1	*09/03/2003*
	Fukushima Daini Unit 2	*07/09/2002*
	Fukushima Daini Unit 3	*05/27/2002*
	Fukushima Daini Unit 4	
	Kashiwazaki Kariwa Unit 1	*04/15/2004*
	Kashiwazaki Kariwa Unit 2	*05/07/2002*
	Kashiwazaki Kariwa Unit 3	*04/27/2004*
	Kashiwazaki Kariwa Unit 4	*07/31/2003*
	Kashiwazaki Kariwa Unit 5	*05/07/2002*
	Kashiwazaki Kariwa Unit 6	*07/04/2003*
	Kashiwazaki Kariwa Unit 7	*08/06/2002*
Chubu Electric Power Co., Inc.	*Hamaoka Unit 1*	*–*
	Hamaoka Unit 2	*08/21/2003*
	Hamaoka Unit 3	*12/10/2003*
	Hamaoka Unit 4	*11/28/2003*
	Hamaoka Unit 5	*12/20/2004*
Hokuriku Electric Power Co.	*Shika Unit 1*	*04/01/2003*
	Shika Unit 2	*–*
The Kansai Electric Power Co., Inc.	*Mihama Unit 1*	*11/02/2002*
	Mihama Unit 2	*07/17/2002*
	Mihama Unit 3	*06/19/2003*
	Takahama Unit 1	*02/15/2003*
	Takahama Unit 2	*06/06/2002*
	Takahama Unit 3	*11/06/2002*
	Takahama Unit 4	*06/17/2003*
	Ohi Unit 1	*06/04/2003*
	Ohi Unit 2	*12/18/2002*

	Ohi Unit 3	*02/25/2003*
	Ohi Unit 4	*04/16/2002*
The Chugoku Electric Power Co., Inc.	*Shimane Unit 1*	*01/05/2004*
	Shimane Unit 2	*11/14/2003*
Shikoku Electric Power Co., Inc.	*Ikata Unit 1*	*05/10/2002*
	Ikata Unit 2	*04/16/2002*
	Ikata Unit 3	
Kyushu Electric Power Co., Inc.	*Genkai Unit 1*	*03/20/2002*
	Genkai Unit 2	*03/22/2002*
	Genkai Unit 3	*03/07/2003*
	Genkai Unit 4	*11/12/2002*
	Sendai Unit 1	*03/20/2002*
	Sendai Unit 2	*06/28/2002*
The Japan Atomic Power Company	*Tokai No.2*	*12/20/2002*
	Tsuruga Unit 1	*03/14/2003*
	Tsuruga Unit 2	*07/15/2002*

Fig. 19-1 Use of Operation and Maintenance Information by License Holders



WANO: World Association of Nuclear Operators
INPO: Institute of Nuclear Power Operations
NRC: Nuclear Regulatory Commission, USA

Activities for Improvement of Safety

Activities for Improvement of Safety

Activities for improvement of safety are described in each article, but the future activities are compiled in this report.

(1) Assessment of the Seismic Safety of Existing Nuclear installations Following the Revision of the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities (Section 14.5)

The Nuclear Safety Commission revised the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities, etc. in September 2006

Following the revision, NISA required licensees to assess the seismic safety for existing nuclear installations in accordance with the revised Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities, and to report the results.

Each licensee submitted its implementation plan for the seismic safety evaluation to NISA in October 2006.

Each licensee submitted the process to implement the safety assessment within two to three years respectively, and the implementation plan describes that the assessment will be implemented in the order of geological and earthquake survey, determination of the reference earthquake ground motion, and seismic safety evaluation.

NISA decided to confirm the adequacy of contents of the licensees' reports for the seismic safety assessment in the order of submittal.

(2) Training of Human Resources in Nuclear Fields (Section 11.4)

In order to ensure safety of nuclear power generation, highly capable human resources should be trained and ensured. In Japan, it has been an issue to keep future human resources, under the severe conditions of low birthrate and increase of elderly people, decrease in population, retirement of skillful engineers, and decrease of construction opportunities of nuclear installation.

The Ministry of Education, Culture, Sports, Science and Technology and the Ministry of Economy, Trade and Industry will implement the nuclear human resource training program focusing on the following items from 2007 fiscal year; i) support of educational activities, such as enhancement of education and research in nuclear field, of internship for nuclear engineering students, and of preparation of core curriculum for the nuclear engineering, ii) support of research activities focusing on fundamentals and infrastructure of nuclear technology which might contribute to bring up researchers in nuclear fields.

(3) Ensuring Safety of Existing Nuclear installation (Section 14.3 (2))

NISA imposes on licensees the duty for the periodic safety review conducted every ten years interval, and assesses the feedback of the operating experiences and latest technical knowledge and probabilistic safety assessment. NISA also continues the assessment as a part of the periodic safety review from the viewpoint of improvement in aging management and prevention of degradation of licensee's organizational climate.

As measures for aging management of existing nuclear installation, NISA enhances the periodic safety review, the Periodic Inspection, makes efforts for preparation of the technical standards, and promotes technological developments.

(4) Nuclear Emergency Drill (Section 16.3)

In executing emergency exercises, it is important for relevant persons on emergency preparedness of the national government, local governments, licensees and residents to understand the measures for nuclear emergency preparedness and to take actions appropriately in an emergency. The emergency exercises will be continued, including participation to international drills.

(5) Promotion of Nuclear Safety Research (Section 14.6)

The NSC proposed nuclear safety researches (prioritized nuclear safety researches) that should be performed selectively for about five years from 2005 fiscal year. In the proposal, the following important research areas were proposed as safety researches for nuclear installations; 1) the regulatory system area (example: use of risk information, assessment of root-cause of failures and accidents), 2) light water reactor area (example: safety analysis, material degradation and aging management, seismic safety technologies) and nuclear reactor emergency prevention technologies. The advancement of these studies is checked at the Special Committee on Nuclear Safety Research of the NSC.

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ANNEXES

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Annex 1　List of Nuclear Installations

(As of August, 2007)

(1)　Commercial Power Reactors

	License Holder	Power Station & Unit	Reactor Type	Power [MWe]	Commis-sioning
In Operation	The Japan Atomic Power Co.	Tokai II Power Station	BWR	1,100	11/28/78
		Tsuruga Power Station,			
		Unit 1	BWR	357	03/14/70
		Unit 2	PWR	1,160	02/17/87
	Hokkaido Electric Power Co., Inc.	Tomari Power Station,			
		Unit 1	PWR	579	06/22/89
		Unit 2	PWR	579	04/12/91
	Tohoku Electric Power Co., Inc.	Onagawa NPS,			
		Unit 1	BWR	524	06/01/84
		Unit 2	BWR	825	07/28/95
		Unit 3	BWR	825	01/30/02
		Higashidori NPS, Unit 1	BWR	1,100	12/08/05
	Tokyo Electric Power Co., Inc.	Fukushima Daiichi NPS,			
		Unit 1	BWR	460	03/26/71
		Unit 2	BWR	784	07/18/74
		Unit 3	BWR	784	03/27/76
		Unit 4	BWR	784	10/12/78
		Unit 5	BWR	784	04/18/78
		Unit 6	BWR	1,100	10/24/79
		Fukushima Daini NPS,			
		Unit 1	BWR	1,100	04/20/82
		Unit 2	BWR	1,100	02/03/84
		Unit 3	BWR	1,100	06/21/85
		Unit 4	BWR	1,100	08/25/87
		Kashiwazaki Kariwa NPS,			
		Unit 1	BWR	1,100	09/18/85
		Unit 2	BWR	1,100	09/28/90
		Unit 3	BWR	1,100	08/11/93
		Unit 4	BWR	1,100	08/11/94
		Unit 5	BWR	1,100	04/10/90
		Unit 6	ABWR	1,356	11/07/96
		Unit 7	ABWR	1,356	07/02/97
	Chubu Electric Power Co., Inc.	Hamaoka NPS,			
		Unit 1	BWR	540	03/17/76
		Unit 2	BWR	840	11/29/78
		Unit 3	BWR	1,100	08/28/87
		Unit 4	BWR	1,137	09/03/93
		Unit 5	ABWR	1,267	01/18/05
	Hokuriku Electric Power Co.	Shika NPS,			
		Unit 1	BWR	540	07/30/93
		Unit 2	ABWR	1,358	03/15/06

In Operation	The Kansai Electric Power Co., Inc.	Mihama Power Station,			
		Unit 1	PWR	340	11/28/70
		Unit 2	PWR	500	07/25/72
		Unit 3	PWR	826	12/01/76
		Takahama Power Station,			
		Unit 1	PWR	826	11/14/74
		Unit 2	PWR	826	11/14/75
		Unit 3	PWR	870	01/17/85
		Unit 4	PWR	870	06/05/85
		Ohi Power Station,			
		Unit 1	PWR	1,175	03/27/79
		Unit 2	PWR	1,175	12/05/79
		Unit 3	PWR	1,180	12/18/91
		Unit 4	PWR	1,180	02/02/93
	The Chugoku Electric Power Co., Inc.	Shimane NPS,			
		Unit 1	BWR	460	03/29/74
		Unit 2	BWR	820	02/10/89
	Shikoku Electric Power Co., Inc.	Ikata Power Station,			
		Unit 1	PWR	566	09/30/77
		Unit 2	PWR	566	03/19/82
		Unit 3	PWR	890	12/15/94
	Kyushu Electric Power Co., Inc.	Genkai NPS,			
		Unit 1	PWR	559	10/15/75
		Unit 2	PWR	559	03/30/81
		Unit 3	PWR	1,180	03/18/94
		Unit 4	PWR	1,180	07/25/97
		Sendai NPS,			
		Unit 1	PWR	890	07/04/84
		Unit 2	PWR	890	11/28/85
	Subtotal		(55 units)	49,467	
Under Construct-ion	Hokkaido Electric Power Co., Inc.	Tomari Power Station Unit3	PWR	912	2009/12 (Planned)
	The Chugoku Electric Power Co., Inc.	Shimane NPS Unit3	ABWR	1,373	2011/12 (Planned)
	Subtotal		(2 units)	2,285	
In Planning	The Japan Atomic Power Co.	Tsuruga Power Station			
		Unit3	APWR	1538	2016/03 (Planned)
		Unit4	APWR	1538	2017/03 (Planned)
	Electric Power Development Co. LTD	Ohma NPS, Unit 1	ABWR	1,383	2012/03 (Planned)
	Tokyo Electric Power Co., Inc.	Higashidori			
		Unit 1	ABWR	1,385	FY2014
		Unit 2	ABWR	1,385	FY2016 or later
	The Chugoku Electric Power Co., Inc.	Kaminoseki NPS,			
		Unit 1	ABWR	1,373	FY2014
		Unit 2	ABWR	1,373	FY2017
	Subtotal		(7 units)	9,975	

Under Decommis sioning	The Japan Atomic Power Co.	Tokai Power Station	GCR	166	07/25/66 Commercial Operation discontinued on 03/31/1998 Notification of Dismantling on 10/04/2001

(2) Reactors at the stage of research and development

	License Holder	Power Station & Unit	Reactor Type	Power [MWe]	Commis-sioning
In Preparation of Decommissioning	Japan Atomic Energy Agency	Fugen*	ATR	165	03/20/79 Commercial Operation discontinued on 03/29/2003
Under Construction		Monju **	FBR	280	Criticality on 04/05/94

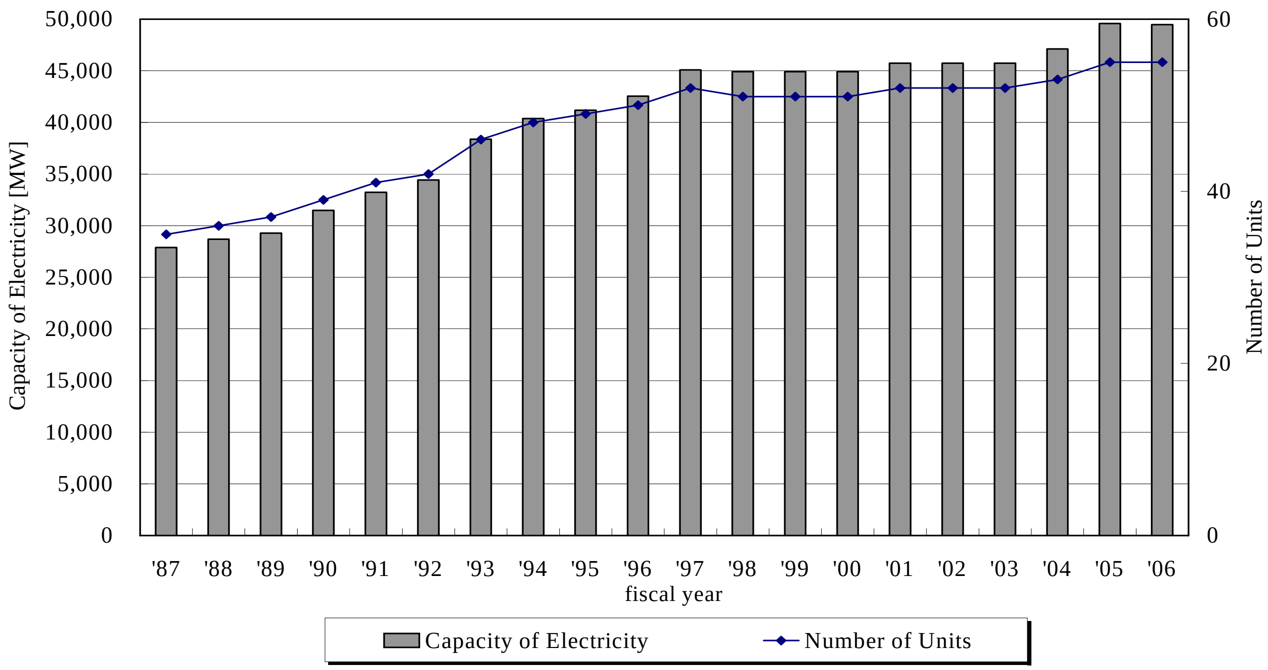
Note: In planning: Projects that were designated as sites for electric power development by the Minister of Economy, Trade and Industry, and have not obtained construction plan approval.

* : This plant discontinued commercial operation on March 29, 2003, and is in preparation for decommissioning.

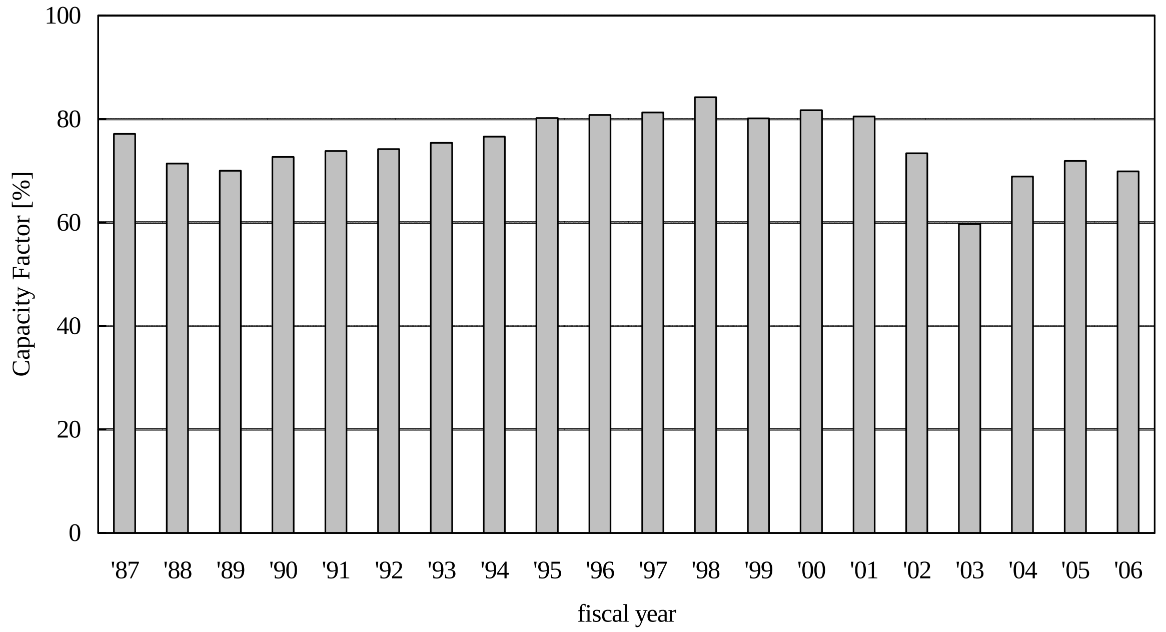
**: These plants reached criticality and correspond to the category “reactor in operation” of the Convention on Nuclear Safety

Annex 2 Data on Nuclear Installations

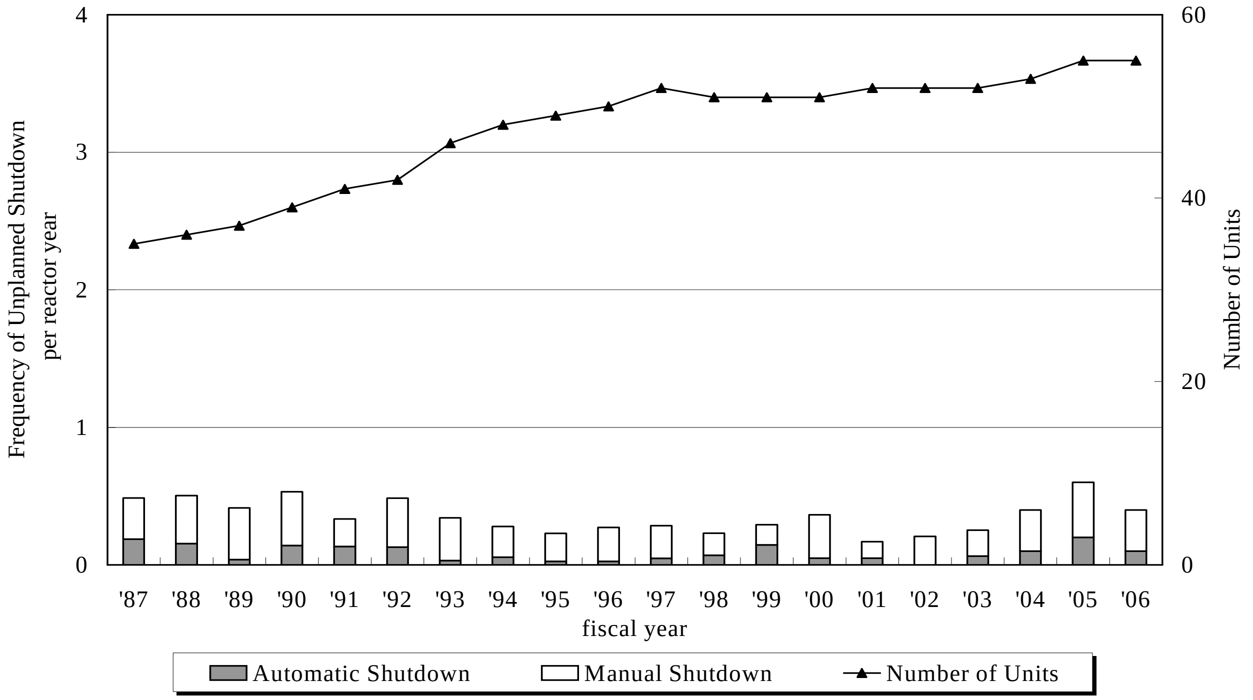
2.1 Capacity of Electricity of Commercial Nuclear Power Reactors



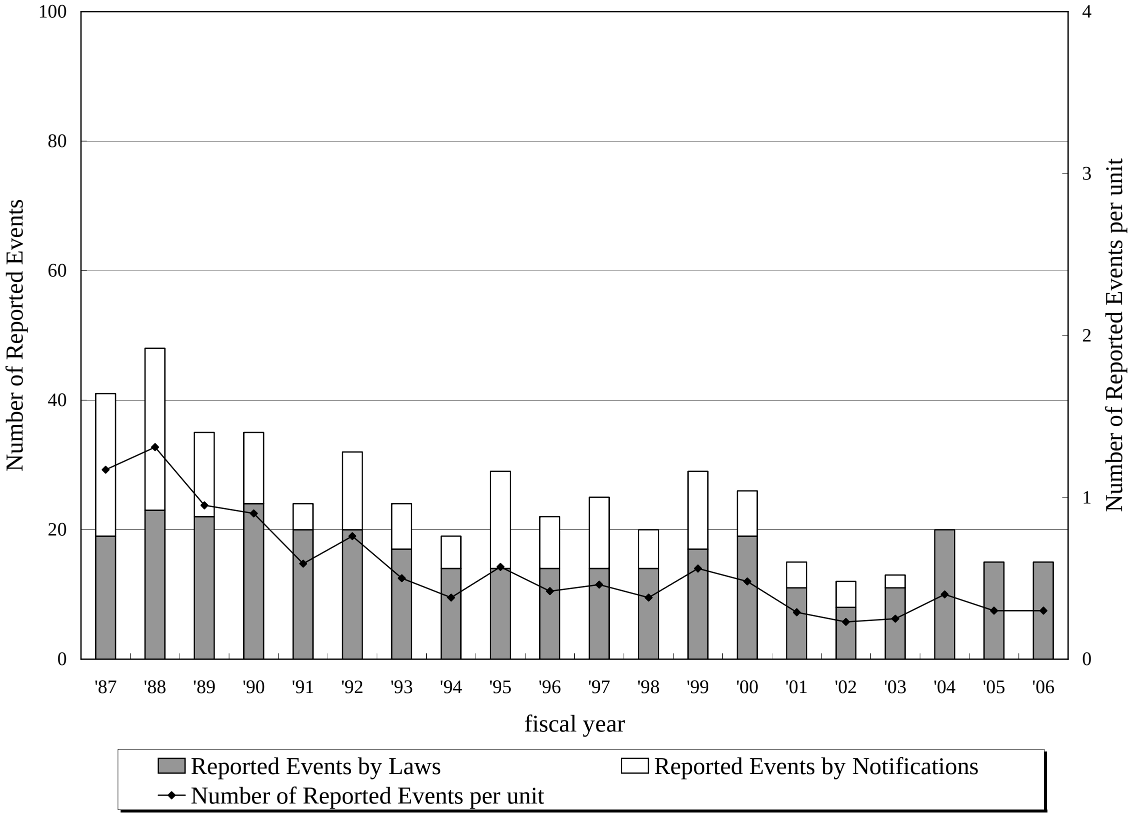
2.2 Capacity Factor of Commercial Nuclear Power Reactors



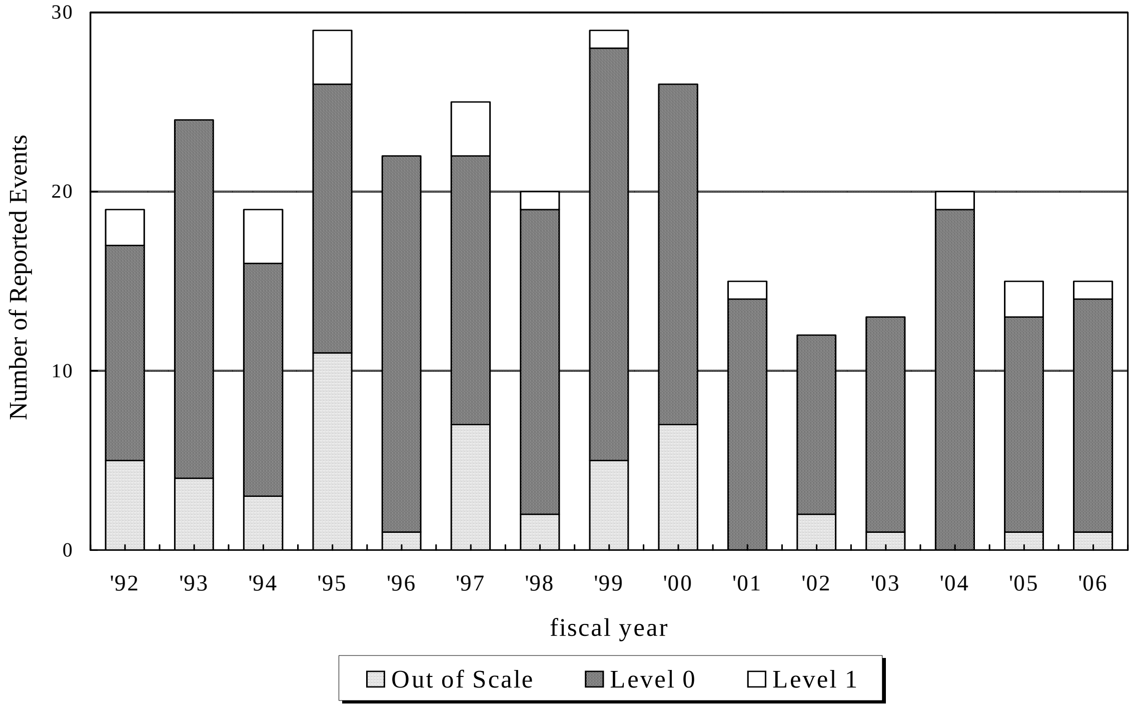
2.3 Frequency of Unplanned Shutdown at Commercial Nuclear Power Reactors
(except during commissioning)



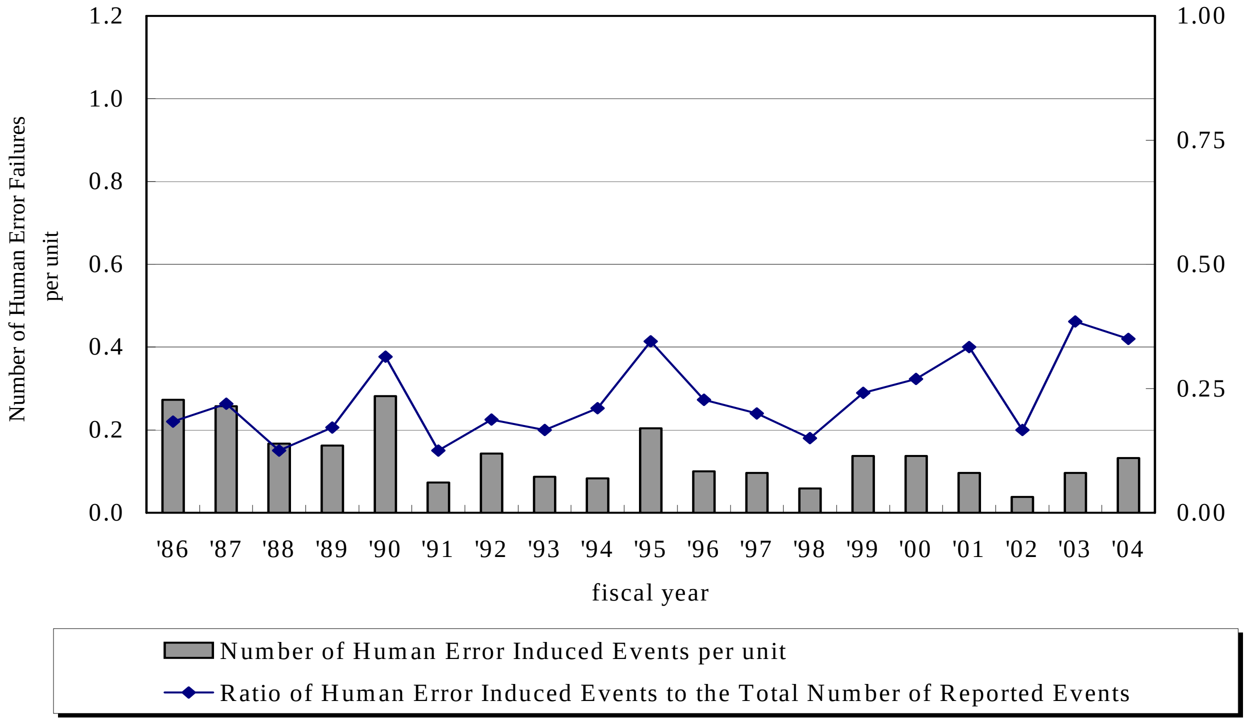
2.4 Reported Events (by Laws & Notifications)
of Commercial Nuclear Power Reactors



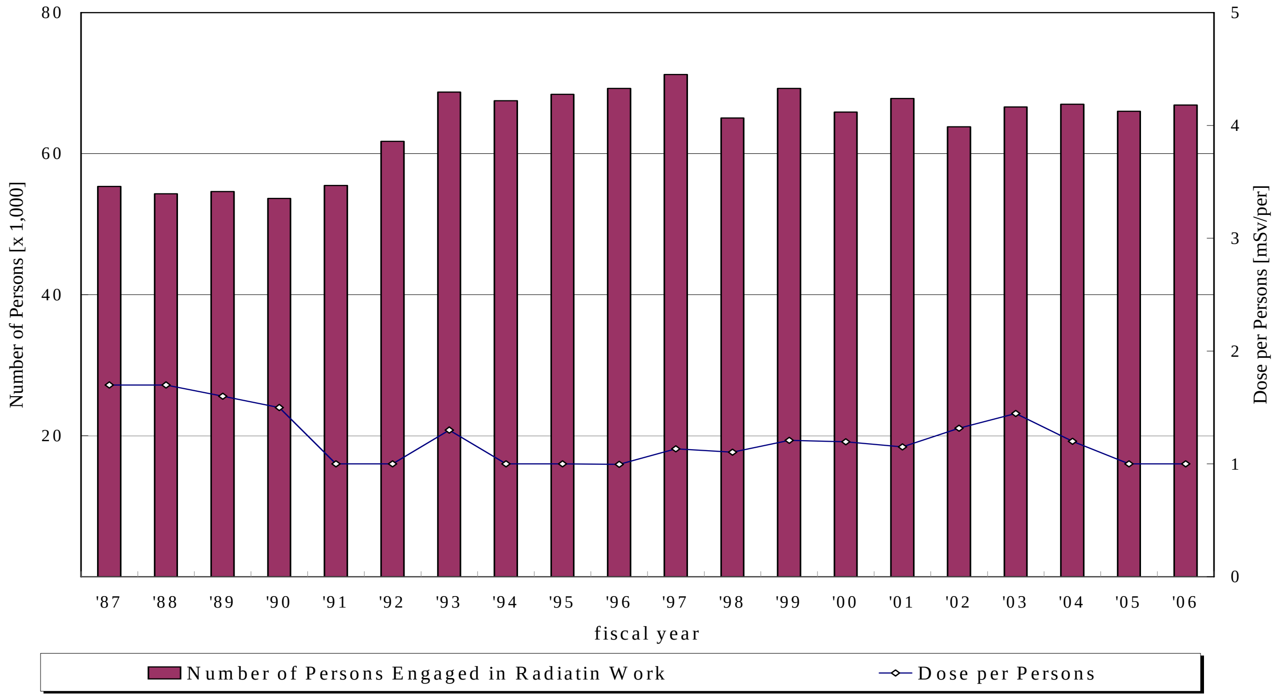
2.5 Assessment of Events by INES for Commercial Nuclear Power Reactors



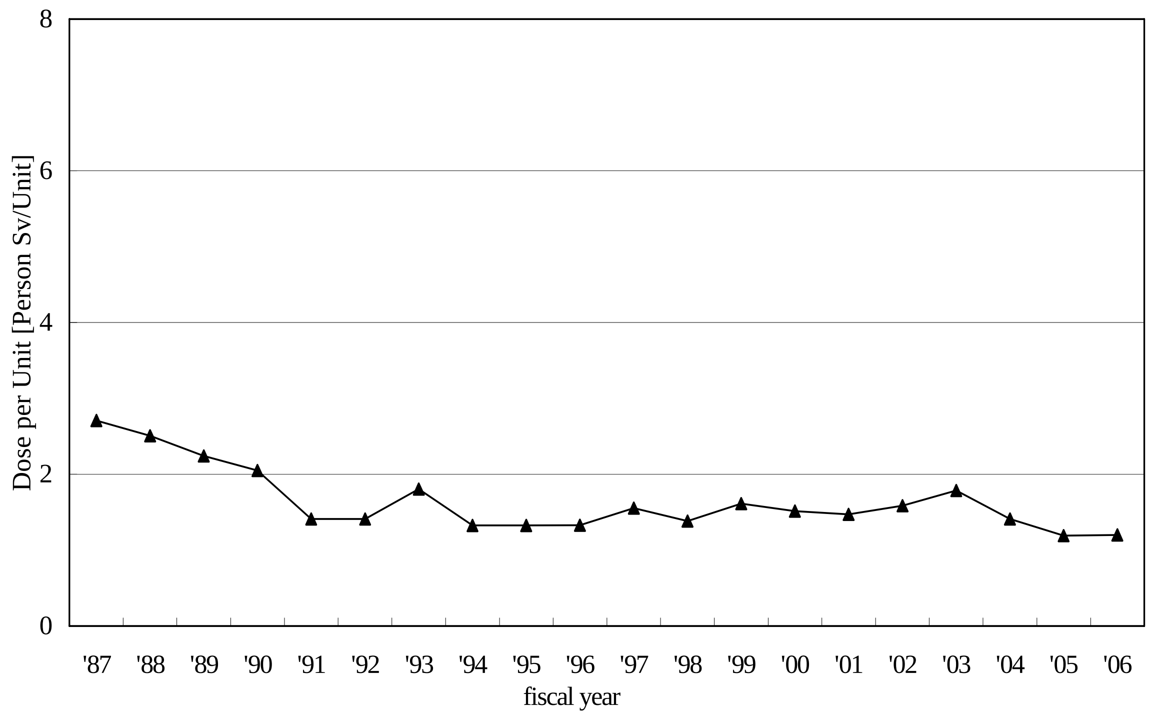
2.6 Human Error Induced Events Reported



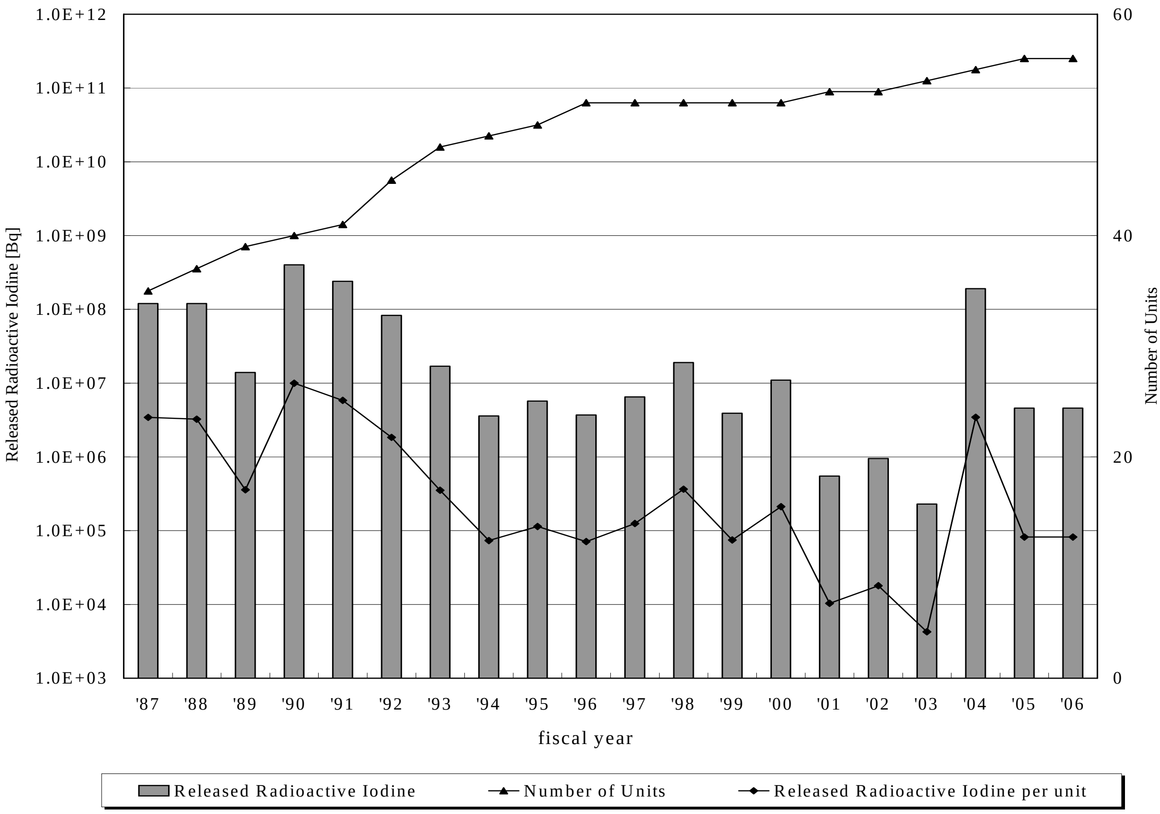
2.7 Dose per Persons at Commercial Nuclear Power Reactors



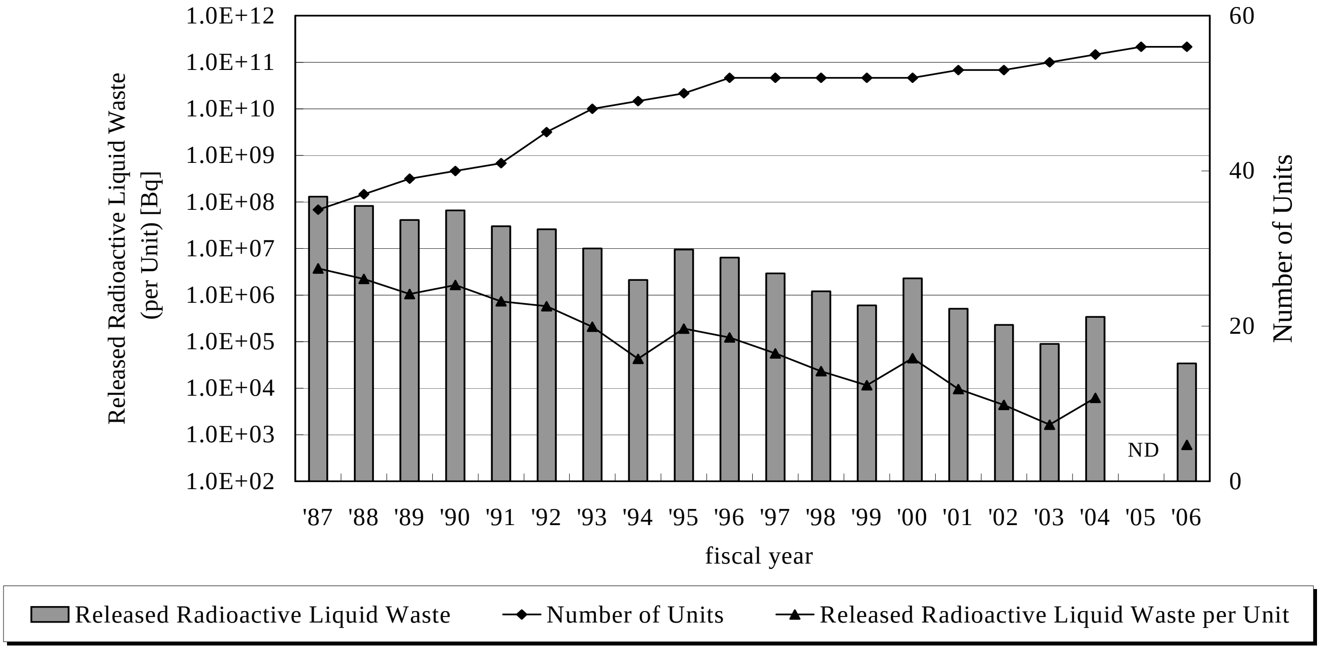
2.8 Averaged Dose at Commercial Nucear Power Reactors



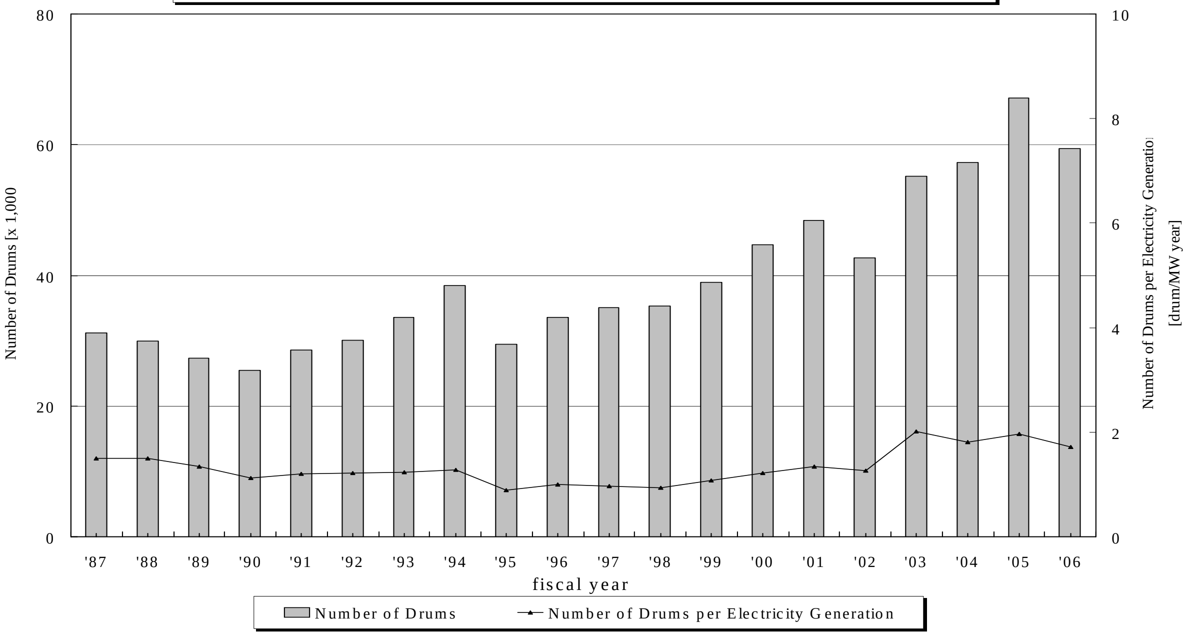
2.9 Radioactive Gaseous Waste (I-131) Released
from Commercial Nucear Power Reactors
(Number of units is summed from their initial criticality.)



2.10 Radioactive Liquid Waste (except H-3) Released
from Commercial Nuclear Power Reactors



2.11 Radioactive Solid Waste Generation per Electricity Generation of Commercial Nucear Power Reactors
(Total quantity of radioactive solid waste is converted to the drum of 200-liter capacity.)



Annex 3 Legislation and Guidelines

(1) The Rule for the Installation, Operation, etc. of Commercial Nuclear Power Reactors (Excerpt)

(Ordinance No.77 of the Ministry of International Trade and Industry, dated December 28, 1978)
Latest Revision: Ordinance of the Ministry of Economy, Trade and Industry dated June 19, 2007

In accordance with the Law on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (Law No. 166, 1957) and provisions for installation, operation etc. of commercial nuclear power reactors of the Ordinance for the Enforcement of the Law on the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors (Ordinance No. 324, 1957), and in order to enforce the said provisions, the Rules for Installation, Operation etc. of Commercial Power Reactors is established as follows.

(Periodic Evaluation of Nuclear Facility)
Article 15-2. The reactor establisher shall take actions provided for in the followings for every reactor and every period not exceeding 10 years in accordance with Article 35, Paragraph 1 of the Law:
(i) Evaluation of the implementing situation of fitness-for-safety activities at the nuclear facilities; and
(ii) Evaluation of the situation of reflection of state of the art technical information on fitness-for-safety activities at the nuclear facilities.
2. The licensee shall perform a technical assessment for deteriorations due to aging on essential components and structures in order to ensure safety of the reactor facility that have been defined by the Minister of Economy, Trade and Industry (hereinafter called as “important safety-related equipment etc.”) and those described in the following by the day elapsed 30 years from the date the reactor operation started, and shall develop a ten years plan concerning actions to be taken for maintenance of the reactor facility based on the said assessment. However, it shall not apply when the condition of the deterioration accompanying the service of the reactor facility can be adequately comprehended for portions of the components and structures that have active functions:
(i) The components and structures that have functions to generate actuation signals to the engineered safety features and the reactor shutdown system;
(ii) The components and structures that have functions for monitoring condition of the reactor facility in an accident;
(iii) The components and structures that have functions to safely shut down the reactor facility from other places than the central control room;
(iv) The components and structures that have functions to retain the reactor coolant, which are not the important safety related equipment etc.;
(v) The components and structures that have functions to circulate the reactor coolant;
(vi) The components and structures that have functions to clean-up the reactor coolant;
(vii) The components and structures that have functions to store radioactive materials;
(viii) The components and structures that have functions to supply electric power, which are not the important safety related equipment etc.;
(ix) The components and structures that have functions to measure and control the reactor facility (excluding those described in Item (i));
(x) The components and structures that have functions to support operation of reactor facilities;
(xi) The components and structures that have functions to prevent release of fission products into the reactor coolant;
(xii) The components and structures that have functions to mitigate the reactor pressure increase;
(xiii) The components and structures that have functions to suppress the power escalation;
(xiv) The components and structures that have functions to make up the reactor coolant; and,
(xv) Essential components and structures for taking emergency measures, and the components and structures that have functions for monitoring an abnormal situation.
3. The licensee shall review the assessment set forth in the preceding paragraph for every period not exceeding 10 years after the day on which the assessment and the plan set forth in the preceding paragraph were performed and developed, and shall develop a ten years plan concerning actions to be taken for maintenance of reactor facilities on the basis of the said review.
4. Provisions set forth in the preceding three paragraphs shall not apply to the reactor that received the license set forth in Article 43-3-2, Paragraph 2 of the Law.

(Report on Accidents and Troubles, etc.)
Article 19-17. Pursuant to the provision of the Law Article 62-3, any reactor establisher (including previous reactor establisher, etc. The same shall apply hereinafter in the immediately following Article and Article 24) shall, when any of the following subparagraphs is applicable, report to the Minister of Economy, Trade and Industry immediately to that effect, and within ten (10) days on the situation and the measures against it:
(i) When nuclear fuel material is stolen or its whereabouts is unknown;
(ii) When a reactor is shut down by failure of a reactor facility or when it become necessary to shut down a reactor during operation, or when reactor power output changes more than 5%, or when reactor power output change of more than 5% is required. Except when it is one of the following and the establisher announced officially about the situation of the concerned failure;
A. When it occurs in the term of the periodic inspection provided in Article 54-1 of the Electricity Utilities Industry Law (Law No. 170, 1964) (limited to those equipment concerned to the failure, which functional and operational

situation cannot be checked under the reactor shutdown condition),

B. When the failure does not cause deviation from the limit of operation (it is a requirement defined in the operational safety program for operation of the reactor facility, and when it cause any deviation from the concerned conditions the measure that the establisher should take is also defined in the operational safety program, the same in this paragraph), and there is no change observed related with the concerned failure, and when the establisher performs inspection of the failed equipment concerned, or

C. When the reactor output is required to follow the limit of operation.

(iii) When a reactor establisher has checked the equipment etc. important to the safety, and when the equipment etc. important to safety is considered that it does not satisfy the standard described in Article 9 or in Article 9-2 of the Order of Establishing Technical Standards for Nuclear Power Generation Equipment (Order No. 62 of Ministry of International Trade and Industry, 1965), or when it is considered that it does not have function to secure safety of the nuclear reactor facility;

(iv) When there is a failure of equipment etc. important to safety by the fire. Excluding the concerned failure is due to the measure of fire extinguishing or prevention of the spread of fire;

(v) Except for the preceding three subparagraphs, when deviation from the limit of operation by the failure of a the nuclear reactor facility (except those minor troubles which impact on operation of nuclear reactor is insignificant) is caused, or when the measure for the concerned deviation defined in the operational safety program is not implemented at the time of deviation from the limit of operation;

(vi) When the failure of a reactor facility or occurrence of other unexpected situation is considered to have caused any trouble in the situation of discharge of gaseous radioactive wastes through the ventilation facility or in the situation of discharge of liquid radioactive wastes through the drainage facility;

(vii) When the concentration of radioactive materials in the air outside the environment monitoring area exceeds the allowable limit in the case of discharge of gaseous radioactive wastes through the ventilation facility;

(viii) When the concentration of radioactive materials in the water outside the environment monitoring area exceeds the allowable limit in the case of the discharge of liquid radioactive wastes through the drainage facility;

(ix) When nuclear fuel materials or materials contaminated with nuclear fuel materials (hereinafter referred to as "nuclear fuel materials etc.") leak out of the control zone;

(x) When nuclear fuel materials etc. leak within the control zone due to failure of a nuclear reactor facility or occurrence of other unexpected situation. However, this is not the case in any of the followings (except the case when new measures such as human entry restriction into the leakage-related place and key control have been taken or when the leaked substances have spread outside the control zone);

A. When revealed liquid nuclear fuel materials etc. do not spread out of the floodgate that is installed in the circumference of the equipment of the concerned leakage for prevention of leakage enlargement,

B. When the function of ventilation facility of the concerned area of the leakage is maintained properly at the time when gaseous nuclear fuel materials etc. leak, or

C. When the amount of radioactivity of the leaked nuclear fuel materials etc. is very little and when the degree of the leakage is minor.

(xi) When the person who enters into the control zone suffered radiation exposure due to the failure of a nuclear reactor facility or occurrence of other unexpected situation, and when the effective dosage of concerned exposure exceeds or could exceeds five mSv for a personnel engaged in radiation work or 0.5 mSv for a person other than the personnel engaged in radiation work;

(xii) When the dosage of personnel engaged in radiation work exceeds or could exceed the allowable dose limit; or

(xiii) When a control rod that is not actually operating for insertion into or withdrawal from the original control rod position (it is one of control rod positions that the procedure for operating control rods, which is established by the licensee based on the Operational Safety Program, defines to set up based on a certain interval in order to control the control rod, the same shall apply hereinafter) moves to another control rod position or moves passing through the said another control rod position, or when the control rod that is at the full insertion position (it is a control rod position where the control rod is to be inserted to the maximum extent, the same shall apply hereinafter) which is not actually operating for insertion or withdrawal moves passing through the full insertion position into the further insertion direction. However, this is not the case when fuels are not loaded in the core.

(xiv) Other than those above subparagraphs, when persons are injured or could be injured in the nuclear reactor facility (except for what hospitalization medical treatment is not needed for other than radiation hazard).

(2) Ordinance of the Ministry for Establishing Technical Standards for Nuclear Power Generation Systems

(Ordinance of the Ministry of International Trade and Industry No. 62, June 15, 1965)
Latest Revision: Ordinance No.121 of the Ministry of Economy, Trade and Industry dated December 22, 2005

Pursuant to the provision of Article 48, paragraph 1 of the Electric Utilities Industry Law (Act No. 170, 1964) (including the cases when Article 74, paragraph 2 is applied mutatis mutandis), the Ordinance of the Ministry for Establishing Technical Standards for Nuclear Power Generation Systems was enacted as follows.

Article 1 (Scope of Application)

This Ordinance of the Ministry applies to the electric structures to be established for electricity production by utilizing nuclear energy as its energy source.

Article 2 (Definitions)

In this Ordinance of the Ministry, the meanings of the terms listed in the following items shall be as prescribed respectively in those items:

(i) The term "Radiation" means the radiation specified in Article 3, item 5 of the Atomic Energy Basic Act (Act No. 186, 1955) or electron beams or X-rays with energy less than one mega electron volts (eV) excluding natural radiation;

(ii) The term "Reactor Facility" means a reactor and its associated systems;

(iii) The term "Primary Coolant" means the fluid of which main function is to take out the heat generated in reactor core directly from the reactor;

(iv) The term "Secondary Coolant" means the fluid of which main function is to take out the heat of the Primary Coolant using heat exchangers with the main purpose to drive turbine;

(v) The term "Primary Cooling System" means a circuit through which the Primary Coolant circulates;

(vi) The term "Abnormal Transient During Operation" means the abnormal condition due to a single failure or malfunction of equipment, or a single error of operator’s action and external disturbances expected to occur with a similar frequency to these during operation of the Reactor Facility;

(vii) The term "Engineered Safety Features" means facilities with functions to suppress or prevent the potential release of a large amount of radioactive materials due to failure of fuel in a reactor, etc. resulted from a failure, damage etc. of the Reactor Facility;

(viii) The term "Safety Systems" means systems described in the followings, which failure, damage etc. directly or indirectly could cause possible Radiation hazards to the public;

- (a) Systems related to the Primary Cooling System and their associated systems,
- (b) Reactivity control system (system to adjust the reactivity during normal operation; hereinafter the same shall apply), and facilities related to reactor shutdown system (system to bring a reactor to subcritical condition and shut it down to maintain the non-critical state; hereinafter the same shall apply) and their associated systems,
- (c) Safety protection device (it means devices to actuate the reactor shutdown system automatically in such cases as an abnormal transient during operation, a difficulty in reactor operation due to an earthquake etc., or a loss of primary coolant accident, and the Engineered Safety Features automatically in the case of possible release of a large amount of radioactive materials due to failure of fuel in a reactor etc.; hereinafter the same shall apply), the emergency core cooling system (it means systems to remove the heat generated in the reactor pressure vessel in the case of the functional loss of the facility to remove the heat generated in the reactor pressure vessel during normal operation; hereinafter the same shall apply) and other systems required to ensure safety of the reactor in an emergency and their associated systems,
- (d) Reactor containment and their isolating valves, and
- (e) Emergency power supply systems and their associated systems.

(ix) The term "Controlled Area" means the area within a nuclear power plant where there is a hazard that the dose from external Radiation would exceed the limit specified separately by a Public Notice, and that the concentration of radioactive material in the air (excluding those included in the natural air or water; hereinafter the same shall apply) would exceed the level specified separately by a Public Notice or that the density of radioactive material on the surface of objects contaminated by radioactive materials would exceed the level specified separately by a Public Notice;

(x) The term "Peripheral Monitoring Area" means the area surrounding the Controlled Area, the outside of which there would be no possibility of the dose exceeding the limit of the dose specified separately by a Public Notice;

(xi) The term "Reactor Coolant Pressure Boundary" means the part that becomes a pressure barrier by automatic closure of valves actuated by a damage etc. of the facilities related to the Primary Cooling System;

(xii) The term "Allowable Fuel Damage Limit" means a limit on the extent of fuel cladding tube damage, which extent is acceptable by the safety design and enables the reactor to operate safely;

(xiii) The term "Reactivity Worth" means a variation in reactivity of a reactor produced by insertion or withdrawal of control rods, or injection of liquid control materials, etc.;

(xiv) The term "Maximum Reactivity Worth of A Control Rod" means the maximum value of the Reactivity Worth produced in a reactor core by withdrawal of one control rod, when the reactor is in a critical condition (including near-to-critical condition);

(xv) The term "Reactivity Insertion Rate" means the amount of reactivity added in a core per unit time by withdrawal of control rods, etc.;

(xvi) The terms "Class 1 Vessel", "Class 1 Piping", "Class 1 Pump" or "Class 1 Valve" (hereinafter referred to as "Class 1 Component") mean the components that compose the Reactor Coolant Pressure Boundary,

(xvii) The terms "Class 2 Vessel ", "Class 2 Piping", "Class 2 Pump" or "Class 2 Valve" (hereinafter referred to as "Class 2 Component") mean the following components;

- (a) The components belonging to systems necessary to shutdown the reactor safely or to ensure safety in an emergency, which failure, damage etc. would indirectly cause Radiation hazards to the public (for ducts belonging to the Radiation management systems, only their parts from the penetrations of the Reactor Containment to the outboard isolating valves are Class 2 Components),
- (b) The component belonging to systems of the circuit through which fluids (i.e., steam and feed water) circulates mainly for the purpose to drive a turbine, located between the Class 1 Component and its nearest stop valve in the steam system located in the downstream of the Class 1 Component, and located between the Class 1 Component and its nearest stop valve in the feet water system located in the upstream of the Class 1 Component, and
- (c) The component other than that listed in the items 1 and 2 located between a penetration of the Reactor Containment and an inboard isolation valve or outboard isolating valve,

(xviii) The terms "Class 3 Vessel", "Class 3 Piping", (hereinafter referred to as "Class 3 Component") mean the vessel or piping other than the Class 1 Component, Class 2 Component, Reactor Containment and the duct belonging to the Radiation management systems (limited to the piping which contains a fluid of radioactive material concentration of 37 mBq/cm^3 (for liquid fluid, 37 kBq/cm^3) or more, or the piping where its Maximum Operating Pressure exceeds zero MPa);
(xix) The term "Class 4 Piping" means the duct belonging to the Radiation management system, which contains a fluid of radioactive material concentration of 37 mBq/cm^3 or more (excluding the part belonging to the Class 2 Piping);
(xx) The term " Reactor Containment" means a vessel provided to prevent a leak of hazardous materials such as radioactive materials that are released from the mechanical equipment in the vessel;
(xxi) The term "Concrete Reactor Containment" means the Reactor Containment that has the steel-plate lined Concrete Part;
(xxii) The term "Concrete Part" means the part of reinforced-concrete structure or pre-stressed-concrete structure of the Concrete Reactor Containment;
(xxiii) The term " Steel-Plate Lined Part etc." means the steel plate that is lined on the Concrete Part (hereinafter referred to as "Liner Plate"), the steel plate to connect the Liner Plates of the shell and the bottom (hereinafter referred to as "Knuckle"), the penetration sleeve, and the metal fitting to the Concrete Part to prevent a leak of hazardous materials such as radioactive materials released from the mechanical equipment in the Concrete Reactor Containment;
(xxiv) The terms "Class 1 Support Structure", "Class 2 Support Structure" or " Reactor Containment Support Structure" mean the structures that support the Class 1 Component, Class 2 Component or Reactor Containment, respectively;
(xxv) The term "Operating Condition I" means the normal operation condition of a Reactor Facility;
(xxvi) The term "Operating Condition II" means the condition other than the Operating Condition I, Operating Condition III, Operating Condition IV, and the Testing Condition;
(xxvii) The term "Operating Condition III" means the condition that a shutdown of reactor operation is urgently required due to a failure or malfunction, etc. of the reactor facility;
(xxviii) The term "Operating Condition IV" means the condition that the abnormal condition assumed in the safety design of the Reactor Facility exists;
(xxix) The term "Testing Condition" means the condition that a pressure exceeding the Maximum Operating Pressure is applied to the Reactor Facility for a pressure test;
(xxx) The term "Load Condition I" means the condition where the Concrete Reactor Containment receives the load assumed in the Operating Condition I (excluding the condition during snow accumulation and storm.);
(xxxi) The term "Load Condition II" means the condition where the Concrete Reactor Containment receives the load assumed in either of the following conditions;

(a) The condition during SRV actuation (excluding the condition during snow accumulation and storm)
(b) The condition during a pressure test of the Reactor Containment (excluding the condition during snow accumulation and storm), and
(c) The condition during snow accumulation in the Operating Condition I (excluding the condition during storm).

(xxxii) The term "Load Condition III" means the condition of Concrete Reactor Containment other than the condition during storm in the Operating Condition I or the Load Condition IV in the Operating Condition IV;
(xxxiii) The term "Load Condition IV" means the condition that an abnormal event assumed for the safety of the Reactor Containment exists in the Concrete Reactor Containment during the Operating Condition IV (including the time during snow accumulation or storm);
(xxxiv) The term "Maximum Operating Pressure" means the pressure equal to or greater than the highest pressure to be received by the subject component or core support structure under the operating condition in which their main functions should be achieved and defined by the design;
(xxxv) The term "Maximum Operating Temperature" means the temperature equal to or greater than the highest temperature arising to the subject component, support structure or core support structure under the operating condition in which their main functions should be achieved and defined by the design;
(xxxvi) The term "Minimum Operating Temperature" means the temperature equal to or less than the lowest temperature arising to the subject component, support structure or core support structure under the operating condition or Testing Condition in which their main functions should be achieved and defined by the design; and
(xxxvii) The term "Mechanical Load" means the load by the dead weight, reactive force by piping or support structure, and other added loads, but excluding the seismic load and defined by the design.

Article 3 (Facility by Special Design)
When approved by the Minister of Economy Trade and Industry for a special ground, reactors, steam turbines, and the associated systems can be established not pursuant to the provisions of this Ordinance of the Ministry.
2 Entities intending to obtain an approval set forth in the preceding paragraph shall apply with the written application attached with related drawings, which describes the ground and the establishing method.

Article 4 (Protection Measures etc.)
When the Reactor Facility or the steam turbine driven by the Reactor Coolant or Secondary Coolant or its associated systems would impair the safety of the reactor due to the assumed natural phenomena (land sliding, fault, snow slide, flooding, tidal wave, high tide, uneven subsidence of the foundation ground, etc., however, excluding an earthquake), protection measures, improvement of the foundation ground or other appropriate measures shall be taken.
(2) When there is a place of business, railway, road etc. in the region adjacent to the Peripheral Monitoring Area, protection measures and other appropriate measures shall be taken in order not to impair the reactor safety by fire or explosion accident at the place of business, or an accident etc. of a vehicle carrying dangerous goods.
(3) When crash of an airplane would impair the reactor safety, protection measures and other appropriate measures shall be taken.

Article 4-2 (Prevention of Damage by Fire)

The Reactor Facility, steam turbine or its associated systems shall be provided with the measure appropriately combining the measures listed in the following each item in order not to impair the reactor safety by fire:

(i) In order to prevent an outbreak of fire, the following measures shall be taken;
  (a) Leak prevention and other measures for the system containing incendiary or inflammable materials shall be taken,
  (b) Noncombustible or flame retarding materials shall be used for the cable, reactor control room and other associated systems of the reactor according to the amount of combustible material, etc.,
  (c) Lightning protection systems etc. to prevent the outbreak of a fire due to lightning strike or other natural phenomena shall be provided,
  (d) The hydrogen supply facility etc. shall be installed such that it would not impair the reactor safety even in the event of hydrogen combustion, and
  (e) When rapid combustion of hydrogen generated and accumulated due to radiolytic decomposition would impair the reactor safety, preventive measures against buildup of hydrogen shall be taken.

(ii) The following measures for detection and extinguishing of fire shall be taken;
  (a) Fire detection and extinguishing systems that can extinguish the fire at an early stage shall be provided, and
  (b) Capabilities of the fire detection and extinguishing systems provided in the above (a) shall not be impaired by the natural phenomena expected to occur simultaneously with the fire.

(iii) Firewalls and other measures shall be taken to mitigate the consequence of a fire.

(2) Fire detection and extinguishing systems provided in Item 2, (a) of the preceding paragraph shall be those not to impair the functions of the Safety Systems due to a failure, damage, inadvertent actuation etc.

Article 5 (Earthquake Resistance)

The Reactor Facility or steam turbine driven by the Reactor Coolant or Secondary Coolant or its associated systems shall be provided not to cause Radiation hazards to the public due to the damage by the seismic force working on these.

(2) The seismic force described in the preceding paragraph shall be derived based on the condition of the foundation ground, and the extent of the earthquake damage and situation of seismic activities, etc. based on the past seismic records in the region, corresponding to the structure of the Reactor Facility or steam turbine driven by the Reactor Coolant or Secondary Coolant or its associated systems and the extent of the disaster when these are damaged.

Article 6 (Prevention of Damage by Flow Induced Vibration etc.)

The fuel assembly and reflector, the structures supporting these components, the thermal shield, and the vessel, piping, pump and valve belonging to the systems related to the Primary Cooling System shall be provided so as not to suffer a damage by the flow-induced vibration due to circulation, boiling etc. of the Reactor Coolant or Secondary Coolant or the temperature fluctuation due to mixing of fluids with a temperature difference, etc.

Article 7 (Facilities such as Fences)

The Controlled Area of a nuclear power plant shall be provided with walls, fences, barriers, etc. not to allow free access to the area, and it shall be indicated that the area is the Controlled Area.

(2) The boundary between the access-controlled areas (an area that especially requires control for maintaining the integrity of the Reactor Facility other than the Controlled Area; hereinafter the same shall apply) and the Controlled Areas, and other areas of a nuclear power plant shall be provided with boundaries, fences etc. in order to distinguish them from other areas, or it shall be indicated that the area is the access-controlled area.

(3) The Peripheral Monitoring Area of a nuclear power plant shall be provided with boundaries, fences etc. in order to restrict unnecessary access to the area except those who have authorized access for job-related reasons, or it shall be indicated that it is the Peripheral Monitoring Area. However, this shall not apply when it is obvious that there would be no possibility of access to the area concerned.

Article 7-2 (Prevention of Unlawful Entry)

A nuclear power plant shall be provided with appropriate intrusion preventive measures to a facility where the Safety Systems are provided with in order to prevent unlawful intrusions.

Article 7-3 (Prevention of a Collapse of Steep Sloping Ground)

The electric structures to be constructed at the designated dangerous area for a collapse of steep sloping ground pursuant to the provision of Article 3, paragraph 1 of the Act for Preventing a Disaster by Collapse of a Steep Sloping Ground (1969, Act No. 57) shall be installed in such a manner that they would not contribute to or induce a collapse of the steep sloping ground in the area concerned (a sloping ground specified in Article 2, paragraph 1 of the sama Act).

Article 8 (Reactor Facility)

The Reactor Facility shall have the capability to control the chain reaction of the nuclear fission by controlling the reactivity of the reactor as well as the capability to control the reactivity of the reactor safely and stably during its normal operation and the power suppression characteristics inherent to the reactor even in the event of an Abnormal Transient During Operation.

(2) The Reactor Facility (excluding an auxiliary boiler) shall be provided in such a manner that the maintenance and inspection (including tests and inspections) of necessary parts can be carried out during operation or shutdown of the reactor in order to confirm its integrity and capability.

(3) The Reactor Facility shall be provided in such a manner that, in the event of significant leakage of the fluid containing radioactive materials from a vessel, piping, pump, valve, or other mechanical equipment during normal operation, the leakage could be processed safely by the system to process the liquid radioactive waste.

(4) A system belonging to the Reactor Facility, assumed to be damaged by a missile following a damage of a steam turbine, pump etc. and then impair safety of the Reactor Facility shall be provided with damage preventive measures such as installation of protection component or others

(5) When a system belonging to the Reactor Facility is shared with other Reactor Facility, it shall be provided so that the reactor safety would not be compromised.

Article 8-2 (Safety Systems)

The Safety Systems listed in Article 2, item 8, (c) and (e) shall be provided such that they have redundancy or diversity, and independency taking into consideration of the function, structure and principle of operation of their mechanical equipment in order to enable them to function even when the external power is not available in the event of a single failure (it means that one mechanical equipment loses its intended safety function due to a single cause; hereinafter the same shall apply) of the mechanical equipment composing the Safety Systems concerned.
(2) Safety Systems shall be provided such that those can accomplish their function under all environmental conditions assumed.

Article 9 (Material and Structure)

The materials and structures of the vessel, piping, pump or valve (hereinafter referred to as "component") that belongs to the Reactor Facility (excluding a compressor and an auxiliary boiler,) or their support structures, or core support structures shall be in accordance with the following each item, and, in this case, the provisions from Items (i) to (vii) and Item (xv) shall be applied prior to their use:

(i) Materials to be used for the Class 1 Component and Class 1 Support Structure shall be pursuant to the followings;
  (a) The Class 1 Component or Class 1 Support Structure shall have an appropriate mechanical strength and chemical composition (including appropriate corrosion resistance under the in-service stress etc.) for its in-service conditions such as pressure, temperature, water quality, Radiation and load,
  (b) The material to be used for the Class 1 Vessel shall be one confirmed by a mechanical test etc. to have an appropriate fracture toughness under the conditions for the vessel concerned to be used, such as pressure, temperature, Radiation and load,
  (c) The material to be used for the Class 1 Component (excluding the Class 1 Vessel) or Class 1 Support Structure (excluding the support structure for the Class 1 Piping and valve) shall be one confirmed by a mechanical test etc. to have appropriate fracture toughness under the Minimum Operating Temperature of the component or support structure concerned, and
  (d) The material to be used for the Class 1 Component or Class 1 Support Structure (limited to the rod and bolt) shall be one confirmed by nondestructive examinations to have no harmful flaw.

(ii) Materials to be used for the Class 2 Component and Class 2 Support Structure shall be pursuant to the followings;
  (a) The Class 2 Component or Class 2 Support Structure shall have appropriate mechanical strength and chemical composition for its in-service conditions such as pressure, temperature and load,
  (b) The material to be used for the Class 2 Component shall be one confirmed by a mechanical test etc. to have appropriate fracture toughness under the Minimum Operating Temperature of the component concerned, and
  (c) The casting belonging to the Class 2 Component shall be one confirmed by nondestructive examinations to have no harmful flaw.

(iii) Materials to be used for the Class 3 Component shall be pursuant to the followings;
  (a) The Class 3 Component shall have an appropriate mechanical strength and chemical composition for its in-service conditions such as pressure, temperature and load, and
  (b) The material to be used for the Class 4 component belonging to the Engineered Safety Features shall be one confirmed by a mechanical test etc. to have appropriate fracture toughness under the Minimum Operating Temperature of the component concerned.

(iv) Materials to be used for the Class 4 Piping shall have an appropriate mechanical strength and chemical composition for its in-service conditions such as pressure, temperature and load;

(v) Materials to be used for the Reactor Containment (excluding the Concrete Reactor Containment; hereinafter the same shall apply in this item) and the Reactor Containment Support Structure shall be pursuant to the followings;
  (a) The Reactor Containment or Reactor Containment support structure shall have an appropriate mechanical strength and chemical composition for its in-service conditions such as pressure, temperature, humidity and load, and
  (b) The material confirmed by a mechanical test etc. to have appropriate fracture toughness under the Minimum Operating Temperature for the Reactor Containment or Reactor Containment support structure.

(vi) Materials to be used for the concrete and steel lining parts, etc. of the Concrete Reactor Containment shall be pursuant to the followings;
  (a) The concrete shall have an appropriate compressive strength under the conditions such as pressure, temperature and load under which the Reactor Containment concerned is to be used,
  (b) The concrete shall have long-term durability not to cause the harmful expansion and the corrosion of the reinforcement rebar,
  (c) The reinforcement rebar, tendon and fixing device (hereinafter referred to as "reinforcement rebar etc.") to be used as a structural member for strength of the Concrete Part shall have an appropriate mechanical strength, chemical composition and geometry/dimension for its in-service conditions such as pressure, temperature and load of the Reactor Containment concerned, and
  (d) The provisions of (a) and (b) of the preceding item shall apply mutatis mutandis to the materials to be used for the Steel Lining Part etc.

(vii) The provisions of item 1, (a) and (c) shall apply mutatis mutandis to the materials to be used for a core support structure;

(viii) The structure and strength of Class 1 Component and Class 1 Support Structure shall be pursuant to the followings;
  (a) The overall strain of the Class 1 Component shall be kept within the elastic range under the conditions that the Maximum Operating Temperature, Maximum Operating Pressure and mechanical load are applied (hereinafter referred to as "conditions defined by the design"),
  (b) The overall strain of the Class 1 Support Structure shall be kept within the elastic range under the Operating Condition I and the Operating Condition II,

(c) The Class 1 Vessel (excluding the omega seal etc.), Class 1 Piping, Class 1 Valve and Class 1 Support Structure shall not produce an overall plastic strain under the Operating Condition III. However, this clause shall not apply to a local plastic strain at structural discontinuities,
(d) The Class 1 Vessel (excluding the omega seal etc.), Class 1 Piping and Class 1 Support Structure shall not produce a plastic strain that results in a ductile fracture under the Operating Condition IV,
(e) The Class 1 Vessel (excluding the bolt, omega seal etc.) shall not produce an overall plastic strain under the Testing Condition. However, this clause shall not apply to a local plastic strain at structural discontinuities,
(f) The Class 1 Vessel (excluding the bolt etc.), Class 1 Piping, Class 1 Valve (limited to the valve body) and Class 1 Support Structure shall not produce a progressive strain under the Operating Condition I and Operating Condition II,
(g) The Class 1 Vessel, Class 1 Piping, Class 1 Valve (limited to the valve body) and Class 1 Support Structure shall not produce a fatigue fracture under the Operating Condition I and Operating Condition II,
(h) The Class 1 Vessel (limited to the shell, end plate etc.) shall not buckle under the Operating Condition I, Operating Condition II, Operating Condition III and Operating Condition IV and Testing Condition,
(i) The Class 1 Piping shall not buckle under the conditions defined by the design,
(j) The Class 1 Support Structure shall not buckle under the Operating Condition I, Operating Condition II, Operating Condition III and Operating Condition IV, and
(k) Notwithstanding the provisions of the preceding (b), (c), (d), (f), (g) and (k), the provisions for the Class 1 Vessel shall apply mutatis mutandis to the Class 1 Support Structure welded to a Class 1 Vessel, which failure would damage the Class 1 Vessel,

(ix) The structure and strength of Class 2 Component and Class 2 Support Structure shall be pursuant to the followings;
(a) The overall strain of the Class 2 Component shall be kept within the elastic range under the conditions defined by the design,
(b) The expansion joint belonging to the Class 2 Component shall not produce a fatigue fracture under repeated stress added under the conditions defined by the design,
(c) The Class 2 Piping (excluding the expansion joint) shall not produce a fatigue fracture under the Operating Condition I and Operating Condition II,
(d) The Class 2 Vessel and Class 2 Piping shall not buckle under the conditions defined by the design, and
(e) The Class 2 Support Structure welded to the Class 2 Component, which failure would damage the Class 2 Component, shall not produce a ductile fracture or buckle under the Operating Condition I and Operating Condition II.

(x) The structure and strength of Class 3 Component shall be pursuant to the followings;
(a) The overall strain shall be kept within the elastic range under the conditions defined by the design,
(b) The expansion joint belonging to the Class 3 Component shall not produce a fatigue fracture under repeated stress added under the conditions defined by the design, and
(c) A buckling shall not occur under the conditions defined by the design.

(xi) The structure and strength of the Class 4 Piping shall not produce a plastic strain that results in a ductile fracture under the conditions defined by the design;

(xii) The structure and strength of the Reactor Containment (excluding the Concrete Reactor Containment) and Reactor Containment Support Structure shall be pursuant to the followings;
(a) The overall strain of the Reactor Containment (excluding the parts described in (b)) shall be kept within the elastic range under the conditions defined by the design,
(b) The provisions of item 8, (a), (c), (d) and (e) for the Class 1 Vessel shall apply mutatis mutandis to the extremely stressed and specially formed parts of the Reactor Containment,
(c) The provisions of item 8, (b), (c) and (d) for the Class 1 Support Structure hall apply mutatis mutandis to the support structure of the Reactor Containment,
(d) The extremely stressed and specially formed parts and the support structure of the Reactor Containment shall not break due to progressive strain under the Operating Condition I and Operating Condition II,
(e) The expansion joint of the Reactor Containment shall not produce a fatigue fracture under repeated stress added under the conditions defined by the design case,
(f) The extremely stressed and specially formed parts and the support structure of the Reactor Containment shall not produce a fatigue fracture under the Operating Condition I and Operating Condition II,
(g) The Reactor Containment shall not buckle under the conditions defined by the design and the Operating Condition III and Operating Condition IV, and
(h) The support structure of the Reactor Containment shall not buckle under the conditions defined by the design and the Operating Condition I, Operating Condition II, Operating Condition III and Operating Condition IV.

(xiii) The structure and strength of the Concrete Reactor Containment shall be pursuant to the followings;
(a) The concrete shall not suffer a compression fracture under the Load Condition I, Load Condition II and Load Condition III, and, the Concrete Reactor Containment shall not suffer a compression fracture leading to a large plastic strain under the Load Condition IV,
(b) The reinforcement rebar etc. shall not yield under the Load Condition I, Load Condition II, and Load Condition III and produce a strain leading to a rapture under the Load Condition IV,
(c) The Concrete Part shall not suffer a shear fracture under the Load Condition I, Load Condition II and Load Condition III, and the Concrete Reactor Containment shall not suffer a shear fracture leading to a large plastic strain under the Load Condition IV,

(d) The Liner Plate (excluding the part to which a penetration sleeve is installed) shall not produce an extreme residual strain under the Load Condition I and Load Condition II, and shall not result in a rapture under the Load Condition III and Load Condition IV,

(e) The provision of item 12 for the Reactor Containment in addition to the provision of (d) shall apply mutatis mutandis to the Liner Plate (excluding the part to which a penetration sleeve is installed),

(f) The provisions of item 12, (c), (d), (f) and (h) for the Reactor Containment Support Structure shall apply mutatis mutandis to the Liner Plate (limited to the part to which a penetration sleeve is installed), penetration sleeve and fixing metal fittings (excluding fixing metal fittings to be attached to a Liner Plate, which overall strain can be kept within the elastic range under any load conditions). In this case, the terms "Operating Condition I and Operating Condition II" of item 12 shall be deemed to be replaced with the "Load Condition I and Load Condition II", and the terms "Operating Condition I, Operating Condition II, Operating Condition III and Operating Condition IV" with the "Load Condition I, Load Condition II, Load Condition III, and Load Condition IV", and

(g) The provisions of item 12, (b), (d) and (f) for the extremely stressed and specially formed parts of the Reactor Containment shall apply mutatis mutandis to the Knuckle.

(xiv) The structure and strength of a core support structure shall be pursuant to the followings;

(a) An overall strain shall be kept within the elastic range under the conditions defined by the design,

(b) An overall plastic strain shall not be produced under the Operating Condition III. However, this clause does not apply to a local plastic strain at structural discontinuities,

(c) The plastic strain leading to a ductility fracture shall not be produced under the Operating Condition IV,

(d) The core support structure shall not be destroyed due to progressive strain under the Operating Condition I and Operating Condition II,

(e) A fatigue fracture shall not be produced under the Operating Condition I and Operating Condition II, and

(f) A buckling shall not occur under the Operating Condition I, Operating Condition II, Operating Condition III and Operating Condition IV.

(xv) The weld (weld metal and heat affected zone) of major pressure parts of the Class 1 Vessel, Class 1 Piping, Class 2 Vessel, Class 2 Piping, Class 3 Vessel, Class 3 Piping, Class 4 Piping and Reactor Containment shall be pursuant to the followings;

(a) It shall not be a discontinuous and unusual shape,

(b) The weld would not produce a crack due to welding, and shall be one confirmed by nondestructive examinations not to have incomplete penetrations and other defects harmful for ensuring a sound weld,

(c) It shall have an appropriate strength, and

(d) It shall be welded with the welding procedure etc. confirmed to be appropriate one beforehand by mechanical tests etc.

Article 9-2 (Prevention of Destruction during Service due to a Crack etc.)

The Class 1 Component, Class 1 Support Structure, Class 2 Component, Class 2 Support Structure, Class 4 component, Class 4 Piping, Reactor Containment Support Structure of the Reactor Containment and core support structure in use shall not have any cracks and other defects that could cause their destruction.

(2) The pressure part of the Class 1 Component in use shall not have any cracks and other defects that penetrate the pressure part.

Article 10 (Safety Valve etc.)

The Reactor Facility shall be provided with the safety valve or relief valve (hereinafter referred to as "safety valve etc."; hereinafter the same shall apply in this Article) pursuant to the following each item:

(i) The safety valve etc. shall have a mechanism that functions without fail;

(ii) The valve stem of safety valve etc. shall have a structure that can prevent the leak from the valve seat surface appropriately;

(iii) Materials of the safety valve etc. shall be pursuant to the followings;

(a) The provision of Article 9, item 1 shall apply mutatis mutandis to the materials of safety valve etc. attached to the Class 1 Vessel and Class 1 Piping, and

(b) The provisions of Article 9, item 2 shall apply mutatis mutandis to the materials of safety valve etc. attached to the Class 2 Vessel and Class 2 Piping.

(iv) The safety valve etc. with an auxiliary actuation device shall have a structure that can have a required blow-down capacity even in the event of a failure of the auxiliary actuation device concerned;

(v) The reactor pressure vessel (pressurizer when it is provided; hereinafter the same shall apply in this item) shall be pursuant to the followings;

(a) Two or more safety valves with bellows (referred to as "safety valve with bellows" in item 7) shall be provided at appropriate locations to prevent malfunction of its actuation due to the effect of backpressure, and

(b) The total of safety valve capacity shall be equal to or greater than the capacity required to prevent overpressure of the reactor pressure vessel concerned by appropriate combination of the blow-off pressure and the installed number of the safety valve concerned. However, for the reactor pressure vessel provided with the device to have overpressure preventive effects other than the safety valve, the value equivalent to the overpressure preventive capability of the device concerned could be subtracted.

(vi) The steam generator shall be pursuant to the followings;

(a) Two or more safety valves shall be provided at appropriate locations,

(b) The total capacity of safety valves shall be equal to or greater than the capacity required to prevent overpressure of the steam generator concerned by appropriate combination of the blow-off pressure and the installed number of the safety valve concerned, and

(c) The safety valve shall cease to blow-off promptly after the pressure goes down below its blow-off pressure.

(vii) The piping provided with the pressure reducing valve, of which low pressure portion or the component connected to it is not designed so as to withstand the pressure of the high pressure portion, shall be pursuant to the followings;

(a) For the Class I piping, two or more safety valve with bellows shall be provided closely to the low-pressure side of the pressure-reducing valve,

(b) For piping other than that described in (a), one or more safety valve etc. shall be provided closely to the low-pressure side of the pressure-reducing valve,

(c) The total capacity of safety valves etc. shall be equal to or greater than the capacity required to prevent overpressure of the low pressure side of the piping or the components connected to it when the pressure reducing valve is fully open by appropriate combination of the blow-off pressure and the installed number of the safety valve concerned etc., and

(d) The safety valve shall cease to blow-off promptly after the pressure goes down below its blow-off pressure.

(viii) The vessel (excluding those described in item 5, item 6 and paragraph 3, an auxiliary boiler and the Reactor Containment) or piping (excluding those described in the preceding item) belonging to the Reactor Facility that would be over-pressurized inside shall be provided with safety valves etc. at appropriate locations in the same manner as the provisions of item 6, (b) and (a), (b) and (d) of the preceding item;

(2) In the case of the preceding paragraph, when a rapture disk is provided at the inlet-side or outlet-side of the safety valve etc., it shall be provided pursuant to the following each item:

(i) When provided at the inlet-side of the safety valve etc., it shall be provided pursuant to the followings;

(a) The blow-off pressure of rapture disk shall not be more than the Maximum Operating Pressure of the vessel concerned, and

(b) Destruction of the rapture disk shall not compromise the function of the safety valve etc.

(ii) When provided at the outlet-side of the safety valve etc., it shall be provided pursuant to the followings;

(a) The rapture disk shall break at a low pressure not to prevent actuation of the safety valve etc.,

(b) The pressure that is the sum of the blow-off pressure of a rapture disk and that of the safety valve etc. shall be lower than a blow-off pressure required to prevent overpressure,

(c) The support mechanism of a rapture disk shall be such that the passage area for fluid discharge is equal to or more than the discharge area of the safety valve etc., and

(d) Destruction of a rapture disk shall not compromise the function of the discharge piping.

(3) The vessel which belongs to the Reactor Facility and contains materials such as liquid carbon dioxide gas etc. that would disable actuation of safety valve etc., shall be provided with rapture disks pursuant to the following each item;

(i) One or more rapture disk shall be installed at appropriate locations to obtain the capacity required for overpressure prevention of the vessel concerned by appropriately combining the blow-off pressure and the number of the installed rapture disk, and

(ii) The cross section of the connecting piping of the vessel and the rapture disk shall be equal to or more than the cross section of the rapture disk.

(4) If stop valve is to be provided at the inlet or outlet side of the safety valve etc. or rapture disk for the case of paragraph 1 or the preceding paragraph, a device capable to confirm that the stop valve is fully open during startup and operation of the reactor shall be provided.

(5) The vessel or piping belonging to the Reactor Facility, which would receive a pressure exceeding the pressure defined by the design on the external surface due to the internal pressure getting less than the atmospheric pressure, shall be provided with a vacuum breaker in order to ensure the capacity equal to or more than one required for overpressure prevention pursuant to the following each item:

(j) The material of vacuum breaker shall be pursuant to the followings;

(a) The provision of Article 9, item 1 shall apply mutatis mutandis to the material of vacuum breaker to be provided to the Class 1 Vessel and Class 1 Piping, and

(b) The provision of Article 9, item 2 shall apply mutatis mutandis to the material of vacuum breaker to be provided to the Reactor Containment, Class 2 Vessel and Class 2 Piping.

(ii) The Reactor Containment shall be provided with two or more vacuum breakers at appropriate locations; and

(iii) The vessel or piping other than one described in the preceding item shall be provided with one vacuum breaker at appropriate location.

(6) In the case that the fluid released from safety valves, relief valves, rapture disks or vacuum breakers contains radioactive materials, the Reactor Facility shall be provided such that they can be safely processed.

Article 11 (Pressure Test etc.)

The Class 1 Component, Class 2 Component, Class 3 Component, Class 4 Piping and Reactor Containment shall withstand the pressure proof test under the pressure of the following each item, and have no significant leakage. However, when it has been confirmed to withstand the pressure concerned in the case that the test is performed using air pressure, it can be confirmed that there is no significant leakage at the pressure concerned reduced to the Maximum Operating Pressure (0.9 times of the Maximum Operating Pressure for the Reactor Containment);

(i) The pressure of pressure test for a component that receives internal pressure shall exceed the Maximum Operating Pressure of the component and be one at which the overall strain produced is within the elastic range. However, the test pressure for the Class 1 Component, Class 2 Piping or Class 3 Piping which pressure test is conducted together with the

reactor pressure vessel as a unit, can be the pressure exceeding its operating pressure during normal operation after the test has been conducted prior to the fuel loading, and
(ii) The test pressure of the pressure test of the component that receives external atmospheric pressure as the inside gets below atmospheric pressure shall be the pressure exceeding the largest difference between atmospheric pressure and the internal pressure. In this case, the test pressure of the pressure test can be applied from the inside of the component.
(2) The Class 1 Component, Class 2 Component, Class 3 Component and Class 4 Piping shall have no significant leakage when the leak test is conducted at its normal operating pressure.
(3) The Reactor Containment shall have no significant leakage when the air tightness test is conducted at an air pressure equal to 0.9 times of the Maximum Operating Pressure.
Article 12 (Surveillance Test Piece)
The vessel belonging to the Reactor Facility, which material would significantly deteriorate when irradiated by neutron with one mega electron-volt or more, shall be provided in its inside with the surveillance test piece defined in the following each item for verification of the effects of the irradiation so that the vessel concerned would not have a brittle fracture under its expected operating conditions;
(i) The material from which the surveillance test piece is made shall have the production history equivalent to that of the vessel material subject to the neutron irradiation,
(ii) The number of surveillance test pieces shall be appropriate to confirm the change in the mechanical strength and fracture toughness of the vessel material by taking out and testing them after the vessel is placed in service, and
(iii) The surveillance test pieces shall be located such that they would be under the conditions equivalent to those of neutron spectrum, amount of neutron irradiation and temperature history that the vessel material is subject to.
Article 13 (Reactor Core etc.)
The material of fuel, moderator, reflector and structures supporting these shall maintain the required physical and chemical properties under the severest conditions that are produced with the pressure, temperature and Radiation during normal operation.
(2) The material of fuel, moderator, reflector and structures supporting these shall withstand the Maximum Operating Pressure, dead weight, load to be applied etc.
Article 14 (Thermal Shield)
The reactor pressure vessel of which material would significantly deteriorate due to Radiation shall be provided with the thermal shield to prevent it.
(2) The thermal shield set forth in the preceding paragraph shall be provided in a manner not to compromise operation of the reactor due to its deformation caused by thermal stress.
Article 15 (Primary Coolant)
The Primary Coolant shall maintain the required physical and chemical properties under the severest conditions that are produced with the pressure, temperature and Radiation during normal operation.
Article 16 (Circulation Systems etc.)
The nuclear power plant shall be provided with the system described in the following each item;
(i) The system to circulate the Primary Coolant with the capacity that can transfer the heat generated in the reactor pressure vessel,
(ii) The system to automatically control the pressure fluctuation in the reactor pressure vessel due to a load change etc.,
(iii) The system to automatically make up the loss of the Primary Coolant during normal operation or during small leakage of the Primary Coolant, etc.,
(vi) The system to maintain concentrations of impurities and radioactive materials in the Primary Coolant below the values that would not compromise operation of the nuclear power plant,
(v) The system capable of removing the residual heat generated in the reactor pressure vessel during reactor shutdown (including the time of total AC power loss of short duration, and
(vi) The system that can transfer the heat removed by the system of the preceding item to an ultimate heat sink.
Article 16-2 (Reactor Coolant Pressure Boundary)
Components that compose the Reactor Coolant Pressure Boundary shall be provided in such a way that they can withstand a shock associated with a damage of the Primary Cooling System etc. and increase in their load due to change of the reactor core reactivity, etc.
Article 16-3 (Leakage of Reactor Coolant Pressure Boundary etc.)
The Reactor Coolant Pressure Boundary shall be provided with isolation devices in order to restrict the runoff of the Reactor Coolant.
(2) The Reactor Facility shall be provided with devices to detect the leakage of the Reactor Coolant from the Reactor Coolant Pressure Boundary.
Article 17 (Emergency Core Cooling System)
A nuclear power plant shall be provided with the emergency core cooling system.
(2) The emergency core cooling system shall have the following functions;
(i) The system shall be capable of preventing the temperature of cladding tube from increasing above the temperature that causes a melt down of fuel material or significant failure of the fuel assembly, and
(ii) The system shall prevent generation of a significant amount of hydrogen by the reaction of the cladding tube with the coolant.
(3) The pump of emergency core cooling system shall have the capability to function normally even under the severest condition expected of the pressure and temperature in the reactor pressure vessel or the Reactor Containment.
(4) The emergency core cooling system shall be designed and constructed so that it can be tested during operation of the reactor in order to confirm its operability.

Article 18 (Discharge of the Primary Coolant)
When discharging the Primary Coolant containing radioactive materials (including the fluid containing radioactive materials discharged from the equipment of Article 16, item 4) to the outside of the Primary Cooling System during normal operation, the device to process the fluid safely shall be provided.
Article 19 (Installation of Check Valve)
The piping delivering the fluid not containing radioactive materials to the vessel or piping that contains the Primary Coolant containing radioactive materials or the piping, or the system that processes radioactive wastes (excluding a stack and those facilities provided in Articles 28 and 31; hereinafter the same shall apply in Article 21) shall be provided with a check valve. However, when there would be no back-flow of the fluid containing radioactive materials to the piping providing the fluid not containing radioactive materials, this shall not apply.
Article 20 (Instrumentation Devices)
The nuclear power plant shall be provided with the devices to measure the matters described in the following each item, and in this case, when it is difficult to directly measure them, the said devices may be replaced with devices to indirectly measure the matters concerned;
(i) Neutron flux density in the reactor core,
(ii) Reactor period,
(iii) Position of control rods and the concentration of the liquid control material, in the case when it is used,
(iv) Following matters on the Primary Coolant,
(a) The concentration of radioactive materials and impurities, and
(b) The pressure, temperature and flow rate at the inlet and outlet of the reactor pressure vessel,
(v) The water Level in the reactor pressure vessel (pressurizer, in the case when a pressurizer is provided) and steam generator,
(vi) The pressure, temperature, concentration of combustible gases, concentration of radioactive materials and dose equivalent rate in the Reactor Containment,
(vii) The concentration of radioactive materials in the main steam piping and the air-ejector exhaust gas, etc.,
(viii) The pressure, temperature and flow rate of the Secondary Coolant and the concentration of radioactive materials in the Secondary Coolant at the outlet of the steam generator,
(ix) The concentration of radioactive materials at the stack outlet or at the locations close to it during the stack in service,
(x) The concentration of radioactive materials in discharge water at the discharge point or at the locations close to it,
(xi) The concentration of radioactive materials in discharge water at the outlet or at the locations close to it, of an drainage line provided with an opening in the Controlled Area (the Controlled Area excluding the area where only the dosage related to external Radiation would exceed the dosage specified in a public notice pursuant to the provision of Article 2, item 9; hereinafter the same shall apply) that would be contaminated with radioactive materials,
(xii) The dose equivalent rate in the area of the Controlled Area that is always occupied and other area that especially requires the Radiation control (it means the fuel handling area etc.),
(xiii) The air dose rate and concentration of radioactive materials in the region adjacent to the Peripheral Monitoring Area, and
(xiv) The wind direction and wind velocity at the nuclear power plant.
(2) The device described in item 6 of the preceding paragraph, to measure the dose equivalent rate shall have redundancy and independency.
(3) The devices to measure the matters listed in Paragraph 1, item 1 and from item 3 to 14 shall be able to display and record the measurement results.
Article 21 (Alarm Devices etc.)
The nuclear power plant shall be provided with devices to reliably detect and automatically sound the alarm about the following events: operation of reactor would be significantly compromised due to loss of function of its mechanical equipment or erroneous operator action; remarkable increase in the concentration of radioactive materials described in item 9 of the preceding Article or in the dose equivalent rate described in items 12 and 13 of the said Article; liquid radioactive waste would significantly leak from facilities process or store the liquid radioactive waste (excluding gaseous one; hereinafter the same shall apply).
(2) The nuclear power plant shall be provided with devices to display the operating state of the major mechanical equipment for the reactor, the Primary Cooling System and facilities processing or storing radioactive wastes
(3) The nuclear power plant shall be provided with the appropriate communication system to give necessary instructions to persons in the nuclear power plant concerned, in the event of failure or damage etc. of a facility for the Primary Cooling System.
Article 22 (Safety Protection System)
The nuclear power plant shall be provided with the safety protection system pursuant to the following each item;
(i) One capable of not letting the Allowable Fuel Damage Limit be exceeded, working together with the reactor shutdown system and the Engineered Safety Features, in the event of the Abnormal Transient During Operation or when operation of the reactor is compromised due to an earthquake etc.,
(ii) The mechanical equipment or the trains that compose the system shall have redundancy not to loose their safety protection functions in the event of a single failure or a single removal from its in-service state,
(iii) The trains that compose the system shall be separated from each other, and be independent of each other not to loose their safety protection functions,
(iv) Even in the event of a loss of its driving sources, a system cutoff or other adverse conditions, the safety protection system shall be able to maintain the states that would not compromise the safety of the Reactor Facility by bringing it to more safe states or keeping the states concerned,

(v) When a part of the instrumentation and control system is shared with the safety protection system, the system shall be functionally separated from the instrumentation and control system in order not to loose its safety protection function,
(vi) The system shall enable the conduct of the required tests to confirm its capability during reactor operation, and
(vii) The safety protection system shall enable the change of its actuation setpoints according to the operation conditions.

Article 23 (Reactivity Control System and Reactor Shutdown System)

The nuclear power plant shall be provided with the reactivity control system and reactor shutdown system. In this case, the reactivity control system and the reactor shutdown system may be provided not independently from each other.

(2) The reactivity control system shall have the capability to control a reactivity change associated with a planned power change without exceeding the Allowable Fuel Damage Limit.

(3) The reactor shutdown system shall consist of two or more independent systems such as control rods, liquid control material etc., and shall have the following capabilities;

(i) Each of the two or more independent systems shall be able to bring the reactor to a non-critical state and maintain the state under the high temperature condition during normal operation, and also, at least one of the reactor shutdown systems shall be able to bring the reactor to a non-critical state and maintain the state under the high temperature condition during the Abnormal Transient During Operation without exceeding the Allowable Fuel Damage Limit. In this case, the Reactivity Worth produced by injecting the liquid control material along with an actuation of the emergency core cooling system etc. can be added,

(ii) At least one of the systems shall be able to bring the reactor to a non-critical state and maintain the state under the cold condition during normal operation and the Abnormal Transient During Operation,

(iii) In an accident such as a loss of the Primary Coolant etc., at least one of the systems shall be able to bring the reactor to a non-critical state, and at least one of the systems is able to maintain the reactor in a non-critical state. In this case, the Reactivity Worth produced by injecting the liquid control material along with an actuation of the emergency core cooling system etc. can be added, and

(iv) In the case of using control rods, the provisions from item 1 to 3 shall be satisfied, even when one control rod with the largest Reactivity Worth is stuck.

(4) The Maximum Reactivity Worth of Control Rod and the Reactivity Insertion Rate shall not cause a damage of the Reactor Coolant Pressure Boundary and the reactor core that compromises core cooling in the event of the anticipated reactivity insertion event (it means an event where a reactivity is abnormally inserted to the reactor).

(5) The control rod, liquid control material etc. shall be able to maintain the required physical and chemical properties under the severest conditions of the pressure, temperature, and Radiation during normal operation.

Article 24 (Control Material Drive Mechanism)

The mechanism to drive control material shall be provided pursuant to the following each item;

(i) It shall be able to drive the control material at the velocity appropriate to the characteristics of the reactor,

(ii) It shall not be able to drive control rods at the velocity that exceeds the Allowable Fuel Damage Limit, even in the event of an abnormal withdrawal of control rods during normal operation of the reactor,

(iii) It shall not drive control rods in a direction to increase the reactivity of the reactor in the event of a loss of the power source for driving control rods, and

(iv) The mechanism to drive control rods shall not damage the control rods, fuel assemblies, reflectors etc. by insertion of the control rods or other shocks.

Article 24-2 (Nuclear Reactor Control Room etc.)

The nuclear power plant shall be provided with the reactor control room.

(2) The reactor control room shall be centralized with the facility to operate the equipment for the reactivity control system and reactor shutdown system, the one to operate the equipment for ensuring safety of the reactor in an emergency such as the emergency core cooling system, the one to display operating status of major mechanical equipment for the reactor and the Primary Cooling System, the one to display measurement results of major instrumentation, and other major equipment for safe operation of the reactor (including the equipment specified in Article 21, paragraph 1) and also, shall be capable of operating them properly without an inadvertent action.

(3) The reactor control room and the passageway etc. leading to it shall be provided with the shielding and other appropriate Radiation protection measures, and isolation of the ventilation system against the toxic gas that will be generated with the fire etc. outside the control room, and other appropriate protection measures, so that personnel etc. can enter the reactor control room without any problem and remain there for a certain period of time to take measures such as shutdown of the reactor in the event of a failure or damage of the facilities for the Primary Cooling System.

(4) The nuclear power plant shall be provided with the device capable of shutting down the reactor and maintaining it in a safe state from a place other than the reactor control room when it is not available due to a fire etc.

Article 24-3 (Emergency Management Office of A Nuclear Power Plant)

The nuclear power plant shall be provided with the emergency management office of the nuclear power plant at a place other than the reactor control room to take appropriate measures in the event of a damage etc. of the facilities for the Primary Cooling System.

Article 25 (Fuel Storage Facility)

The storage facility for fuel assemblies necessary for normal operation or the spent fuel (hereinafter referred to as "fuel") shall be provided pursuant to the following each item:

(i) The structure shall not allow the fuel to attain the criticality;
(ii) The fuel shall not melt with the decay heat;
(iii) It shall have a capacity large enough to store the fuel, as required;
(iv) The water pool to store the spent fuel and other highly radioactive fuel shall be pursuant to the followings;
  (a) The structure shall not allow the water to overflow or leak,

(b) It shall have the amount of water required to shield the Radiation from the fuel,
(c) When the fuel cladding tube would corrode significantly, it shall be prevented,
(d) It shall be able to detect a leakage of the water and anomalies of the water temperature in the water pool, and
(e) Its functions shall not be compromised at the time of the anticipated fuel drop during fuel handling.

(v) In the case that the radioactive materials released by a fuel damage due to a fuel drop would cause Radiation hazards to the public, a facility to contain the fuel storage facility and a facility to reduce the release of radioactive materials shall be provided in order to mitigate the consequences of the radioactive materials to the outside of the nuclear power plant;

(vi) The dry cask (hereinafter referred to as "cask") that stores the spent fuel in a nuclear power plant shall be pursuant to the followings;

(a) It shall be able to contain the radioactive materials contained in the spent fuel, and to monitor its functions appropriately,
(b) It shall have an appropriate shielding capability for the Radiation from the spent fuel,
(c) It shall be able to prevent the significant corrosion or deformation of the spent fuel cladding tube, and
(d) The main body of the cask, etc. shall be of the appropriate materials and structures for the temperature, Radiation, load, and other conditions during its service.

(vii) Persons other than persons in charge shall not have unnecessary access.

Article 26 (Fuel Handling Facilities)

The fuel handling facilities shall be provided pursuant to the following each item;

(i) It shall have the capability to handle the fuel to be used for normal operation,

(ii) It shall have the structures that do not allow the fuel to attain the criticality,

(iii) The fuel shall not melt with the decay heat,

(iv) It shall not damage the fuel during handling,

(v) The container to enclose the fuel shall withstand the shock, heat etc. during its handling, and shall not be damaged easily,

(vi) The dose equivalent rate on the surface and the dose equivalent rate at a distance of one (1) meter from the surface of the said container in the preceding item with fuel loaded shall not exceed the dose equivalent rates specified in a separate public notice, respectively to prevent its Radiation hazards. However, This shall not apply to the container only used in the Controlled Area, and

(vii) In the even of a loss of the power source for handling the fuel during fuel handling, fuel drop shall be prevented by providing a mechanism to retain the fuel, etc.

Article 27 (Biological Shield etc.)

The biological shield shall be provided at places inside the nuclear power plant and required to prevent Radiation hazards due to external Radiation, pursuant to the following items;

(i) It shall have a shielding capability required for preventing Radiation hazards,

(ii) Places with openings or penetrations such as piping shall be provided with measures for Radiation leak, as required, and

(iii) It shall be able to withstand its dead weight, additional load and thermal stress.

(2) The Reactor Facility, and steam turbine driven by the Reactor Coolant and its associated systems shall be provided such that the air dose rate in the vicinity of the site by the direct gamma ray and sky-shine gamma ray from the facility or systems concerned during normal operation is less than the dose limit specified in a separate public notice.

Article 28 (Ventilation System)

The ventilation system shall be provided at places inside the nuclear power plant, required to prevent Radiation hazards due to the air contaminated with radioactive materials, pursuant to the following items;

(i) It shall have a ventilation capacity required for preventing Radiation hazards,

(ii) It shall be of a structure that could make it hard for the air contaminated with radioactive materials to leak and would not produce backflow,

(iii) In the case of installing the device to clean the air to be discharged, it shall be of the structure easy for the filter to be decontaminated from radioactive materials or to be replaced, and

(iv) An intake port shall be constructed so as to make it hard for the air contaminated with radioactive materials to be taken in.

Article 29 (Prevention of Contamination by Radioactive Materials)

The surface of walls, floors and other portions inside of buildings frequently occupied at the nuclear power plant, which would be contaminated with radioactive materials and touched by people, shall be easy to remove the contamination with radioactive materials.

(2) The nuclear power plant shall be provided with the system to remove the contamination with radioactive materials of the matters that would be touched by people.

Article 29-2

A drainage line with an opening in the Controlled Area to be possibly contaminated with radioactive materials, which discharges the wastewater outside a nuclear power plant, shall be provided with discharge water monitoring equipment and the system to safely process the wastewater containing radioactive materials.

Article 30 (Waste Treatment System, etc.)

The nuclear power plant shall be provided with the system to process radioactive wastes（including a stack, and excluding those provided in Article 28 and the following Article） pursuant to the following items;

(i) It shall have the capability to process radioactive wastes generated in a nuclear power plant such that the concentration of radioactive materials in the air outside the Peripheral Monitoring Area and the water at the boundary of the Peripheral Monitoring Area is below the values specified in a separate Public Notice, respectively,

(ii) It shall be provided separately from the facility that processes wastes other than radioactive wastes. However, in the case where liquid waste other than radioactive wastes is delivered to the facility that processes liquid radioactive wastes and the

liquid radioactive wastes would not flow back to the facility that deals with wastes other than radioactive wastes, this shall not apply,
(iii) It shall be of a structure that could make it hard for radioactive wastes to leak and chemical agents etc. would not significantly corrode,
(iv) The provision of Article 28, item 3 shall apply mutatis mutandis to the facility that processes gaseous wastes, and it shall not discharge gaseous wastes at the locations other than the outlet of a stack,
(v) The container for transport of the liquid radioactive wastes, and the highly radioactive solid wastes which are generated from what is provided within the Reactor Coolant Pressure Boundary, within the site of a nuclear power plant shall withstand the shock, heat etc. during its handling, and shall not be damaged easily. However, for the container only used in the Controlled Area, this shall not apply, and
(vi) The said container in the preceding item with radioactive wastes loaded shall be able to be shielded so that the dose equivalent rate at the surface and the dose equivalent rate at a distance of one (1) meter from the surface of the container do not exceed the dose equivalent rates specified in a separate Public Notice to prevent Radiation hazards. However, this shall not apply to the container only used in the Controlled Area.
(2) The facility in which the system to process liquid radioactive wastes (limited to a portion where a leakage of liquid radioactive wastes would spread, and hereinafter; hereinafter the same shall apply in this paragraph) is provided shall be provided pursuant to the following each item;
(i) The surface of floors and walls inside of the facility should be of a structure that could make it hard for liquid radioactive wastes to leak,
(ii) The floor of the facility shall be of a structure such that liquid radioactive wastes is led to effluent receiving ports by means of the slope of troughs provided at the slope of floor or the floor, and the periphery of the facility that processes liquid radioactive wastes shall be provided with an embankment to prevent leakage of liquid radioactive wastes from spreading,
(iii) The entrance leading to the outside the facility or its periphery shall be provided with an embankment to prevent liquid radioactive wastes from leaking outside the facility. However, in the case where the floors of the facility are lower than the floors of the adjacent facilities or the ground surface and there would be no leakage to the outside of the facility, this shall not apply, and
(iv) It shall be provided such that no floor of the facility exists above the discharge water channel that discharges wastewater outside a nuclear power plant (excluding those for storm water, which have no opening in the Controlled Area which would be contaminated with radioactive materials, and those provided with discharge monitoring equipment and the system that safely processes the wastewater containing radioactive materials.)
(3) The container for transporting the liquid radioactive wastes of Paragraph 1, item 5 shall be provided in the same manner as provided in Paragraph 2, item 3, to prevent the spread of a leakage of liquid radioactive wastes. However, this shall not apply to the container only used in the Controlled Area and of a structure that would not cause leakage.

Article 31 (Waste Storage Facility etc.)
The facility to store radioactive wastes shall be provided pursuant to the following each item;
(i) It shall have the capacity to store the radioactive wastes to be generated during normal operation,
(ii) It shall be of a structure that could make it hard for the radioactive wastes to leak, and
(iii) It shall withstand the heat generated by the decay heat and irradiation, and shall not significantly corrode by chemical agents etc.
(2) The facility provided with the facility to store solid radioactive waste shall be provided such that the contamination with radioactive wastes would not spread.
(3) The provision of paragraph 2 of the preceding Article shall apply mutatis mutandis to the facility provided with the facility to store liquid radioactive wastes. In this case, "the facility that processes liquid radioactive wastes" shall be deemed to be replaced with "the facility that stores liquid radioactive wastes".

Article 32 (Reactor Containment Facility)
The nuclear power plant shall be provided with a Reactor Containment facility pursuant to the following each item, in order that the leak rate in the event of a failure or damage of the components for the Primary Cooling System would not cause Radiation hazards to the public:
(i) The reactor Containment shall be pursuant to the followings;
- (a) It shall withstand the maximum pressure and maximum temperature anticipated in the event of a failure or damage of the components for the Primary Cooling System,
- (b) In the case of providing an opening to the Reactor Containment, the airtightness shall be ensured, and
- (c) The penetrations of the Reactor Containment and the entrance shall be able to undergo the leak test according to the leakage anticipated.

(ii) The piping to be installed penetrating the Reactor Containment shall be provided with isolation valves (shutoff isolation valve (limited to that with a locking device) or automatic isolation valve (excluding the check valve without isolation function); hereinafter the same shall apply) pursuant to the followings;
- (a) The piping to be installed to the Reactor Containment and penetrate the Reactor Containment shall be provided with one isolation valve at the locations close to the inside and outside of the penetration concerned;
- (b) Irrespective of the preceding item (a), it may be pursuant to the followings;
    1. The piping that does not have an opening within a facility for the Primary Cooling System and the Reactor Containment and would not be damaged in the event of a damage of a facility for the Primary Cooling System or the piping that would not cause a leakage of radioactive materials contained in the Reactor Containment due to the remaining liquid inside because of its structure in the event of a damage of a facility for the Primary Cooling System shall be provided with one isolation valve at the location close to the inside or outside of the penetration concerned, and

2. When an isolation valve is to be provided at the inside or outside of the penetration, for the piping at one side of installation location of which isolation valve function could be considered to significantly deteriorate due to humidity etc., two isolation valves shall be provided at the location close to the other side of the penetration.

(c) Notwithstanding the provisions of the preceding (a) and (b), it is not required to provide an isolation valve in the following cases;

1. In the case that providing an isolation valve to the piping of a system required for terminating an accident would compromise safety and the isolation function of the Reactor Containment could not be lost because of the piping of the system concerned, and
2. In the case of the piping related to the measurement or control rod drive mechanism, through which the leakage from the piping concerned is controlled to the fully acceptable extent.

(d) The isolation valve shall not lose its isolation function, even in the event of a loss of its driving power source after it has been closed; and

(e) The isolation valve shall be able to undergo the leak test according to the leakage anticipated.

(iii) When the safety of the Reactor Containment would be compromised due to hydrogen and oxygen generated in the event of a failure or damage of a facility for the Primary Cooling System, a system to reduce hydrogen or oxygen concentration shall be provided;

(iv) In the case that the leakage of gaseous radioactive materials from the Reactor Containment would cause Radiation hazards to the public in the event of a failure or damage of a facility for the Primary Cooling System, a system to reduce the concentration of radioactive materials concerned (including the facility to contain the radioactive materials concerned) shall be provided; and

(v) In order to prevent an increase in the pressure and temperature in the Reactor Containment made in the event of a failure or damage of a facility for Primary Cooling System from compromising safety of the Reactor Containment, a system to remove the heat generated in the Reactor Containment (hereinafter referred to as the " Reactor Containment heat-removal system ") shall be provided pursuant to the followings;

(a) The pump of the Reactor Containment heat-removal system shall be able to function normally even under the severest condition of the pressure and temperature anticipated in the Reactor Containment, and

(b) The Reactor Containment heat-removal system shall be able to undergo the test to confirm its capability during reactor operation.

Article 33 (Emergency Power Supply System)

At least two lines of the electric lines connected to the nuclear power plant shall be able to be accepted at the nuclear power plant concerned, be of special high operating voltage exceeding 60,000 volts, and be provided such that the nuclear power plant concerned could be networked to a power transmission system.

(2) The nuclear power plant shall be provided with power generating equipment with an internal combustion engine as its generative power or emergency standby power unit with the function equivalent to or more than it, in order to maintain the function of the equipment required for ensuring safety in the event of stoppage of deliveries of the electricity from the electric lines described in the preceding paragraph or a generator usually used at the nuclear power plant concerned.

(3) Equipment especially required to ensure the safety of a nuclear power plant shall be provided with a uninterruptible power supply device or the one with the function equivalent to or more than it.

(4) The emergency power supply system and its associated systems shall have redundancy or diversity and independency, and sufficient capacity for ensuring the functions of facilities such as the Engineered Safety Features during an Abnormal Transient During Operation or an accident such as a loss of the Primary Coolant, even in the event of a single failure of the mechanical equipment composing the system.

(5) The nuclear power plant shall be provided with batteries that have capacity required to ensure that the facility to safely shutdown the reactor and cool it down after its shutdown can function even in the event of a loss of all AC power for a short period of time.

Article 34 (Application, Mutatis Mutandis)

The provision of Article 8, paragraph 3 shall apply mutatis mutandis to the steam turbine driven with the Reactor Coolant and its associated systems to be provided at a nuclear power plant,.

(2) The provision of Article 9, item 15 and the provisions of the Chapter 2 of the Ordinance of the Ministry for Establishing Technical Standards on Thermal Power Generation Equipment (Ordinance of the Ministry of International Trade and Industry No. 51 of 1997) shall apply mutatis mutandis to the auxiliary boiler to be provided at a nuclear power plant.

(3) The provision of Article 9, item 15 and the provisions of Chapter 3 of the Ordinance of the Ministry for Establishing Technical Standards on Thermal Power Generation Equipment shall apply mutatis mutandis to the steam turbine and its associated systems to be provided at a nuclear power plant.

(4) The provisions of Article 25 and Articles 26 to 29 of the Ordinance of the Ministry for Establishing Technical Standards on Thermal Power Generation Equipment shall apply mutatis mutandis to the internal combustion engine to be provided at a nuclear power plant.

(5) The provision of Article 4 of the Ordinance of the Ministry for Establishing Technical Standards on Thermal Power Generation Equipment shall apply mutatis mutandis to the electric structures to be provided at a nuclear power plant.

(3) The Law for the Independent Administrative Agency, Japan Atomic Energy Agency, (Excerpt)

(Law No. 155, December 3, 2005)
Latest Revision: Law No. 28, April 20, 2007;
Law No.28, April 20, 2007 (unenforced)

(Mission of the Agency)
Article 4. The incorporated administration agency, Japan Atomic Energy Agency (hereinafter called as the " Agency"), in accordance with the basic policies provided in Article 2 of the Atomic Energy Basic Law, comprehensively, systematically and efficiently conducts fundamental and applied researches on the nuclear power, develops fast breeder reactors and the required nuclear fuel materials for the nuclear fuel cycle, and develops technologies for reprocessing nuclear fuel materials and disposal of high level radioactive wastes, etc., and performs dissemination of these outcomes etc. so as to contribute in promoting research, development, and utilization of atomic energy, which helps the welfare of human society and improvement of the life of the people.
(Scope of Duties)
Article 17. The Agency performs the following duties in order to achieve the missions provided in Article 4;
1 To carry out basic research of nuclear energy,
2 To carry out applied research of nuclear energy,
7 To collect, organize and provide information about nuclear energy, and
8 As other duties than ones described in Item 1 to 3, the Agency performs experiments, researches, surveys, analysis and judgment on the nuclear energy, when requested by heads of related administrative agencies or local governments who authorize that they are necessary.

(4) Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities

September 19, 2006
The Nuclear Safety Commission

1. Introduction
This guide is provided to show the basis of the judgment for adequacy of the seismic design policy in the standpoints to ensure seismic 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 power reactor.
The former 'Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities (decided by the Nuclear Safety Commission "NSC" on 20 July 1981 and revised on 29 March 2001, hereinafter referred to as "Former Guide") ' was the guide which was revised based on the state of arts of evaluating methods of static seismic force etc. by the NSC in July 1981, which had been provided in September 1978 by the Atomic Energy Commission. And it was partially revised in March 2001.
This time, overall revision of Former Guide has been conducted by reflecting accumulated new seismological and earthquake engineering knowledge and remarkable improvement and development of seismic design technology of nuclear power reactor facilities.
Incidentally, this guide shall be revised to reflect the coming new knowledge and experiences suitably according to accumulation of new findings.
2. Scope of Application
This guide shall be applied to the nuclear power reactor facilities (hereinafter referred to as "Facilities").
Nevertheless, basic concept of this guide could be referred to other type nuclear reactor facilities as well as other nuclear related facilities.
Incidentally, if some part of application contents could not comply with this guide, it would not be excluded if it reflected technological improvements or developments and seismic safety could be ensured farther than satisfying this guide.
3. Basic Policy
A part of Facilities designated as important ones from the seismic design points shall be designed to bear seismic force exerted from earthquake ground motion and to maintain their safety function, which could be postulated appropriately to occur but very scarcely in the operational period of Facilities from the seismological and earthquake engineering standpoints such as geological features, geological structure, seismicity, etc. in the vicinity of the proposed site.
Moreover, any Facilities shall be designed to bear the design seismic force sufficiently which is assumed appropriately for every classification in the seismic design from the standpoint of radiological effects to the environment which could be caused by earthquake.
Besides, buildings and structures shall be settled on the grounds which have sufficient supporting capacity.

(Commentary)
I. Regarding Basic Policy
(1) Regarding determination of earthquake ground motion in the seismic design
In the seismic design, it shall be based on the principle that ' the ground motion which could be postulated appropriately to occur but very scarcely in the operational period of Facilities and are feared affecting severely to Facilities' shall be determined adequately, and that, on the premise of this ground motion, the seismic design shall be

conducted not to give any risk of serious radiological exposure to the public in the vicinity of Facilities from the external disturbance initiated by an earthquake.

This policy is equal to the ‘ basic policy’ in Former Guide which is required to the seismic design with the provision of ‘ nuclear power reactor facilities shall maintain seismic integrity against any postulated seismic force assumed so sufficiently that no earthquake would induce significant accidents’.

(2) Regarding existence of “Residual Risk”

From the seismological standpoint, the possibility of occurrence of stronger earthquake ground motion which exceeds one determined on the above-mentioned (1) can not be denied. This means, in determination of seismic design earthquake ground motion, the existence of “Residual Risk”(defined as such a risk that, by extension of the effect of the ground motion which exceeds the determined design ground motion of Facilities, impairing events would occur to Facilities and the event in which massive radioactive materials diffuse from Facilities would break out, or the result of these events would cause radiological exposure hazards to the public in the vicinity of Facilities).

Therefore, at the design of Facilities, appropriate attention should be paid to possibility of occurrence of the exceeding ground motion to the determined one and, recognizing the existence of this “Residual Risk”, every effort should be made to minimize it as low as practically possible not only in the stage of design basis but also in the following stages.

4. Classification of Importance in Seismic Design

Importance in seismic design of Facilities shall be classified into the followings from the standpoints of the possible impact of radiation to the environment caused by earthquake corresponding to the categories of Facilities.

(1) Classification on Function

S Class : Facilities containing radioactive materials by themselves or related directly to Facilities containing radioactive materials, whose loss of function might lead to the diffusion of radioactive materials to the environment, Facilities required to prevent the occurrence of those events and Facilities required to mitigate the consequences resulting from the diffusion of radioactive materials in the occurrences of those accidents, and also whose influences are very significant,

B Class : Facilities of the same functional categories as above S Class, however whose influences are relatively small,

C Class : Facilities except for S or B Class, and ones required to ensure equal safety as general industrial facilities.

(2) Facilities of Classes

Facilities of Classes are shown as follows by the above classification of the importance in the seismic design,

1) S Class Facilities :
   - i) Equipment/piping system composing of the ‘reactor coolant pressure boundary’ (the definition is the same that is described in other Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities),
   - ii) Spent fuel storage pool,
   - iii) Facilities to add the negative reactivity rapidly to shutdown the reactor and Facilities to preserve the shutdown mode of the reactor,
   - iv) Facilities to remove the decay heat from the reactor core after reactor shutdown,
   - v) Facilities to remove the decay heat from the reactor core after the failure accident of reactor coolant pressure boundary,
   - vi) Facilities to prevent the propagation of radioactive materials directly as the pressure barrier at the failure accident of reactor coolant pressure boundary,
   - vii) Facilities, except for those in the category ⅵ) above, to mitigate the diffusion of radioactive materials to the environment at the accident which involves the release of radioactive materials.

2) B Class Facilities :
   - i) Facilities connected directly to reactor coolant pressure boundary and containing radioactive materials by themselves or have possibility to contain radioactive materials,
   - ii) Facilities containing radioactive materials. Except for those whose effect of radiological exposure to the public due to their break is smaller enough to compare with annual exposure limit at the outside of the peripheral observation area, because of its small inventory of containing radioactive materials or of the difference of the type of storage system,
   - iii) Facilities related to the radioactive materials except radioactive wastes and have possibility to give excessive radiological exposure to the public and the operational personnel from their break,
   - iv) Facilities to cool the spent fuels,
   - v) Facilities except for those of S Class, to mitigate diffusion of radioactive materials to the environment at an accident which involves the release of radioactive materials.

3) C Class Facilities :

   Those Facilities not belong to above S or B Class.

5. Determination of Design Basis Earthquake Ground Motion

The ground motion to be established as the basis of the seismic design of the Facilities shall be determined adequately as the ground motion to be postulated to occur but very scarcely in the operational period of Facilities from the seismological and earthquake engineering point of view relating to geology, geological structure, seismicity, etc. in the vicinity of the proposed site, and to be feared making a serious impact to Facilities. (Hereinafter this ground motion is referred to as ”Design Basis Earthquake Ground Motion Ss” or “DBGM Ss”.)

DBGM Ss shall be determined on the following principles.

(1) DBGM Ss shall be determined as following two types of earthquake ground motions in horizontal direction and vertical direction on the free surface of the base stratum at the proposed site, relating to (2)”Site specific earthquakes ground motion whose source to be identified with the proposed site” and (3) “Earthquake ground motion whose source not to be identified” mentioned below.

(2) Site specific earthquakes ground motion whose source to be identified with the proposed site shall be determined on the following principles.

1) Taking account of the characteristics of active faults and the situation of earthquake occurrences in the past and at present in the vicinity of the proposed site, and classifying the earthquakes by the pattern of earthquake occurrence etc. plural number of earthquakes which are feared making severe impact to the proposed site shall be selected (hereinafter referred to as “Investigation Earthquakes”).

2) Following items shall be taken into account concerning the ‘characteristics of the active faults around the proposed site’ in above-mentioned 1).

i) The active faults considered in the seismic design shall be identified as the one of which activities since the late Pleistocene epoch can not be denied. Incidentally, judgment of the faults can depended upon whether the displacement and deformation by the faults exist or not in the stratum or on the geomorphic surface formed during the last interglacial period.

ii) The active faults shall be investigated sufficiently by integrating geomorphological, geological and geophysical methods, etc. to make clear the location, shape, activity of the active faults, etc. according to the distance from the proposed site.

3) For any Investigation Earthquakes selected in above-mentioned 1), following evaluations of earthquake ground motion both ⅰ) with response spectra and ⅱ) by the method with fault models shall be conducted, and DBGM Ss shall be determined from respective Investigation Earthquakes.

Incidentally, in evaluating the earthquake ground motion various characteristics (include the regional peculiarity) according to the pattern of earthquake occurrences, seismic wave propagation channel, etc. shall be taken into account sufficiently.

i) Evaluation of earthquake ground motion with response spectra

For respective Investigation Earthquakes, response spectra shall be appraised by applying appropriate methods and the design response spectra shall be evaluated on these spectra, and earthquake ground motions shall be evaluated in considering the earthquake ground motion characteristics such as duration time, time depending change of amplitude-enveloping curve suitably.

ii) Evaluation of earthquake ground motion by the method with fault model

For respective Investigation Earthquakes, earthquake ground motions shall be evaluated by settling the seismic source characteristics parameters with appropriate methods.

4) Uncertainty (dispersion) concerned with the evaluation process of the DBGM Ss in above-mentioned 3) shall be considered by applying the appropriate methods.

(3) Earthquake ground motion whose source not to be identified shall be determined on the following principle.

Design Earthquake Ground Motions shall be determined by collecting the observation records near the source which are obtained from past earthquakes inside the inland earth’s crust, of which the source can not be related directly to any active faults, settling the response spectra based on those records by taking account of the ground material characteristics of the proposed site, and adding consideration of the earthquake ground motion characteristics such as the duration time, time dependent change of amplitude-enveloping curve, etc. suitably to these results.

(Commentary)

II. Regarding to determination of DBGM Ss.

(1) Regarding the characteristics of DBGM Ss.

In Former Guide, regarding design basis earthquake ground motion two categories of “Earthquake Ground Motion S1” and “Earthquake Ground Motion S2” were required to be determined, however in this revision both these motions were integrated, and enhancement of selection of Investigation Earthquakes, evaluation of ground motion etc. were strived for DBGM Ss.

This DBGM Ss is the premise ground motion of the seismic design to ensure seismic safety of Facilities and, in determining it , it’s adequacy should be checked sufficiently according to the latest knowledge in the specific examination.

(2) The interpretation of the terminology regarding determination of DBGM Ss are as follows.

1) ‘Free surface of the base stratum’ is defined as the free surface settled hypothetically without any surface layer or structure and as the surface of base stratum postulated to be nearly flat with considerable expanse and without eminent unevenness to plan out design basis earthquake ground Motion. ‘Base stratum’ mentioned here is defined as a solid foundation of which sear wave velocity Vs exceeds 700m/s, and which has not been weathered significantly.

2) ‘Active faults’ are defined as faults which moved repeatedly in recent geological age and have also possibility to move in the future.

(3) Regarding the principle of determination DBGM Ss

1) In selecting Investigation Earthquakes, the characteristics of active faults and the situation of earthquake

occurrence in the past and at present should be investigated carefully, and furthermore existing research results concerned with distribution of middle, small and fine size of earthquakes in the vicinity of the proposed site, stress field, pattern of earthquake occurrence (including shape, movement and mutual interaction of the plate) shall be examined comprehensively.

2) Investigation Earthquakes shall be selected depending on the classification considering the pattern of earthquake occurrence etc. as follows.

i) Inside Inland Earth's Crust Earthquake

'Inside inland earth's crust earthquake' is defined as the earthquake which occurs in the upper crust earthquake generation layer and includes one which occurs in the rather offshore coast.

ii) Inter-plates Earthquake

'Inter-plates earthquake' is defined as one which occurs in the interfacial plane of two mutually contacting plates.

iii) Inside Oceanic Plate Earthquake

'Inside oceanic plate earthquake' is defined as one which occurs inside a subducting (subducted) oceanic plate, and is classified into two types,

'Inside subducting oceanic plate earthquake' which occurs near the axis of sea trench or in it's rather offshore area, and 'Inside subducted oceanic plate earthquake (Inside slab earthquake) 'which occurs in the land side area from the vicinity of the axis of sea trench.

3) The evaluation method using fault model should be regarded as important in the case of earthquake whose source is near the proposed site and process of its failure could be supposed to make large impact to evaluation of the ground motion.

4) In consideration of 'uncertainty (dispersion) concerned with the determination process of DBGM Ss', appropriate method should be applied considering the cause of uncertainty (dispersion) and it's extent which are supposed to make large impact directly to plan out DBGM Ss.

5) The principle of determination of 'Earthquake ground motion whose source not to be identified' is implied that, if the detailed investigation would be conducted sufficiently considering the situation etc. in the vicinity of the proposed site, it could not be asserted to evaluate all earthquakes inside inland earth's crust in advance which could have still the possibility to occur near the proposed site, therefore this earthquake should be considered commonly in all applications in spite of the results of the detailed investigation around the proposed site.

The validity of DBGM Ss determined by materializing this principle should be confirmed specifically in checking on the latest information at the time of each application. Incidentally, on that occasion, probabilistic evaluation could be referred as the needs arise regarding the ground motion near the source generated from the source fault which does not indicate any clear trace on the ground surface.

6) Regarding 'Site specific earthquakes ground motion whose source to be identified with the proposed site' and 'Earthquake ground motion whose source not to be identified', the exceedance probability of respective earthquakes should be referred in each safety examination from the standpoint that it is desirable to grasp that the response spectra of each seismic ground motion planed out correspond to what extent of the exceedance probability.

7) In the case that the necessary investigation and evaluation are implemented in selection of Investigation Earthquakes and determination of DBGM Ss, existing materials etc. should be referred in considering the accuracy of them sufficiently. If different result would be obtained compared with the existing evaluation results, its reason should be shown clear.

8) Regarding the ground which supports the structures of Facilities and Facilities themselves, if the peculiar frequency characteristics could be found in the seismic response, it should be reflected to determination of DBGM Ss as the needs arise.

(4) Regarding evaluation of the faults which assumed as the source of earthquake

1) As investigation of the active faults is the basis of the evaluation concerning the faults which is assumed as the source of earthquake, appropriate investigation should be implemented combining adequately the survey of existing materials, tectonic geomorphologic examination, the earth's surface geological feature examination, geophysical examination, etc. according to the distance from the proposed site. Especially in the area near the proposed site, precise and detailed investigation should be applied. Incidentally extent of the area near the proposed site should be decided suitably considering the relation etc. with DBGM Ss determined as 'Earthquake ground motion whose source not to be identified'.

2) Regarding active folds, active flexures, etc. these should also be the object of investigation in above-mentioned 1) as well as the active faults and should be considered in the evaluation of the faults assumed to be the source in accordance with their dispositions.

3) The dispositions of the faults should be evaluated appropriately grasping the under ground structure etc. depending on the regional situation. Incidentally, the special consideration should be required if the earthquake should be assumed from the dispositions of faults in the area where the faults are indistinct.

4) In the case, the scale of earthquake shall be postulated from the length of the fault etc. by applying the empirical formula, the scale should be evaluated adequately considering the special features etc. of the empirical formula.

5) Uncertainty shall be considered appropriately in assumption of the characteristics of the source, in the case that sufficient information could not be obtained to settle the source characteristics parameter including the shape evaluation of the fault to be assumed as the source even by implementing

investigation of the active faults.

6. Principle of Seismic Design

(1) Primal Policy

Facilities shall be designed to fulfill the following primal policies of the seismic design for respective categories of Class.

1) Respective Facilities of S Class shall maintain their safety functions under the seismic force caused by DBGM Ss. And also shall bear the larger seismic force loading of those caused by “Elastically Dynamic Design Earthquake Ground Motion Sd” or the static seismic force shown below. (Hereinafter Elastically Dynamic Design Earthquake Ground Motion Sd is referred to as “EDGM Sd”.)
2) Respective Facilities of B Class shall bear the static seismic force shown below. And, as for the Facilities those are feared of resonating with earthquake, the influence shall be evaluated.
3) Respective Facilities of C Class shall bear the static seismic force shown below.
4) In respective items shown above, the integrity of upper Class Facilities shall not be impaired by the damage of the lower Class Facilities.

(2) Computation Method for Seismic Force

The seismic force for seismic design of Facilities shall be obtained by using the methods shown below.

1) Seismic forces caused by DBGM Ss

Seismic force caused by DBGM Ss shall be computed by applying DBGM Ss in combining horizontal seismic force with the vertical seismic force appropriately.

2) Seismic forces caused by EDGM Sd

EDGM Sd shall be established based on DBGM Ss with the technological judgments. And the seismic forces caused by EDGM Sd shall be also evaluated in combining horizontal seismic forces with the vertical seismic force appropriately.

3) Static seismic force

Evaluation of the Static seismic force shall be based on the followings.

i) Buildings and structures

Horizontal seismic force shall be evaluated by multiplying the seismic story shear coefficient Ci by the coefficient corresponding to the importance classification of the facilities as shown below, and multiplying the weight at the above height of the story concerned.

S Class	3.0
B Class	1.5
C Class	1.0

Here, Ci of the seismic story shear coefficient shall be obtained in putting the standard shear coefficient Co to be 0.2, considering the vibration characteristics of the buildings and structures, categories of the ground, etc.

As for the facilities of S Class, both horizontal and vertical seismic forces shall be combined simultaneously in the most adverse fashion. The vertical seismic force shall be evaluated with the vertical seismic intensity which is obtained by putting the seismic intensity 0.3 as a standard, and by considering the vibration characteristics of buildings and structures, categories of the ground, etc. However the vertical seismic coefficient shall be constant in the height direction.

ii) Components and piping system

The seismic force of respective Classes shall be evaluated with the seismic intensities which are obtained by multiplying the seismic story shear coefficient Ci in above-mentioned i) by the coefficient corresponding to the importance classification of the Facilities as the horizontal seismic intensity, and by increasing the horizontal seismic intensity concerned and the vertical seismic intensity in above-mentioned i) by 20% respectively.

Incidentally, horizontal seismic force shall be combined with the vertical seismic force simultaneously in the most adverse fashion. However, vertical seismic forces shall be assumed to be constant in the height direction.

(Commentary)

III. Regarding the Design Principle

(1) Regarding the necessity of establishment of EDGM Sd

In Former Guide, the design basis earthquake ground motion should have been determined classified as two categories of Earthquake Ground Motion S1 and Earthquake Ground Motion S2 corresponding to the seismic importance classification of the buildings, structures, components and piping system, however in this revision, the determination of DBGM Ss shall only be required. In the seismic design concept to ensure seismic safety of Facilities, it is the basic principle that the safety function s of the seismically important Facilities shall be maintained under the seismic forces by this DBGM Ss.

In addition to confirm maintenance of seismic safety function s of the Facilities under this DBGM Ss with higher precision, establishment of EDGM Sd, which is closely related with DBGM Ss from technical standpoint, is also required to be prescribed.

(2) Regarding establishment of EDGM Sd

The concept of ‘to bear the seismic force’ which prescribed in the Article 6. in this Guide means that Facilities as a whole are designed in the elastic range on the whole to a certain seismic force.

In this case, design in the elastic range means to retain the stress of respective parts of the Facilities under the allowable limits by implementing stress analysis supposing the facilities as the elastic body.

Incidentally, the allowable limits shown here, does not require strict elastic limits and requires the situation that the Facilities as a whole should retain in elastic range on the whole even though the case in which the Facilities partially exceeds the elastic range could be accepted.

Although respective S Class Facilities are required 'to bear the seismic force' by EDEGM Sd, this EDGM Sd is established based on the technological judgment.

The elastic limits condition is the condition that the impact which the Earthquake Ground Motion makes to the Facilities and the situation of the Facilities can be evaluated clearly, and that it makes a grasp of maintenance of seismic safety functions as a whole of the Facilities under the seismic force by DBGM Ss more reliable by confirming that the Facilities as a whole retains in elastic limits condition on the whole under the seismic force by EDEGM Sd.

Namely EDEGM Sd assumes a part of the roles which the Design Earthquake Ground Motion S1 of Former Guide used to be attained in the seismic design.

EDGM Sd should be established by multiplying DBGM Ss by coefficients obtained on the technological judgment in considering the ratio of input seismic loads for the safety functional limits and the elastic limits for the respective Facilities and their composing elements. Here, in evaluating the coefficient, the exceedance probability which is referred in the determination of DBGM Ss would be consulted.

The concrete established value and reason of establishment of EDGM Sd should be made clear sufficiently in respective specific application. Incidentally, the ratio of EDGM Sd and DBGM Ss (Sd/Ss) should be expected larger than a certain extent in considering the characteristics required to EDGM Sd, and should be obtained not to be less than 0.5 as an aimed value.

In addition, EDGM Sd would be established specifically to respective elements which compose the Facilities depending on the difference of their characteristics to be considered in seismic design.

Incidentally, regarding to B Class Facilities, 'as for Facilities that are feared resonating with seismic force loading, the influence shall be evaluated', the earthquake ground motion applied to this evaluation would be established with multiplying EDGM Sd by 0.5.

(3) Regarding the evaluation of the seismic force by DBGM Ss and EDGM Sd

In case that the seismic force by DBGM Ss and EDGM Sd are evaluated based the seismic response analysis, the appropriate analytical methods should be selected and suitable analytical consideration should be settled based on the sufficient investigation in considering to the applicable range of response analysis methods, applicable limits, etc.

Incidentally, in the case 'free surface of the base stratum' is very deep compared with the ground level on which Facilities would be settled, amplification characteristics of the ground motion on the ground level above free surface of the base stratum should be investigated sufficiently and be reflected to the evaluation of the seismic response as the needs arise.

(4) Regarding Static seismic force

Evaluation of the static seismic force should be depended upon 1) and 2) shown below.

In addition, regarding to the buildings and structures, the adequate safety margin of retained horizontal strength of buildings and structures concerned should be checked to maintain the retained horizontal strength required relating to the importance of Facilities, and the evaluation of retained horizontal strength required should be complied to the 3) shown below.

1) Horizontal seismic force

i) The datum plane for evaluation of horizontal seismic force should be the ground surface in principle. However, if it is needed to consider the characteristics such as the constitution of the building and the structures and the relation to the surrounding ground around Facilities, the datum plane should be provided appropriately and be reflected to the evaluation.

ii) Horizontal seismic force applied to aboveground part from the datum plane should be obtained to be the total of the seismic forces acted on the part concerned in accordance with the height of the building and the structure and be calculated with the following formula,

$Qi = n \cdot Ci \cdot Wi$

where,

Qi : Horizontal seismic force acting on the part in question,

n : Coefficient in accordance with importance classification of facilities (Earthquake-proof S Class 3.0, Earthquake-proof B Class 1.5, Earthquake-proof C Class 1.0).

Ci : Seismic story shear coefficient, it depends on the following formula,

$Ci = Z \cdot Rt \cdot Ai \cdot Co$

where,

Z : Zoning factor (to be 1.0, the regional difference is not considered),

Rt: A value representing vibration characteristics of building to be obtained by the appropriate calculation methods specified in standards and criteria which are assumed to be adequate for safety. Here, 'the appropriate calculation methods in standards and criteria which are assumed to be adequate for safety' corresponds to the Building Standard Law etc.

However, if the value which expresses the vibration characteristics and is evaluated considering the structural characteristics of buildings and structures, and the response characteristics and situation of the ground in the seismic condition would be confirmed to fall short of the value calculated by the methods in the Building Standard Law etc. it could be reduced to the evaluated value by this method (but equal to or not less than 0.7).

Ai: A value representing a vertical distribution of seismic story shear coefficient according to the vibration characteristics of building, to be calculated by the appropriate methods specified in standards, criteria and the other appropriate methods as is like Rt,

Co: Standard shear coefficient (to be 0.2),

Wi : Total of fixed loads and live loads supported by the part in question.

iii) Horizontal seismic force which acts on the parts of the buildings and structures under the datum plane should be evaluated by following formula,

Pk =n・k・Wk

where,

Pk : Horizontal seismic force acting on the part in question.

n : Coefficient in accordance with importance Classification of Facilities (Earthquake-proof S Class 3.0, Earthquake-proof B Class 1.5, Earthquake-proof C Class 1.0).

k : Horizontal seismic coefficient by the following formula,

$$k \geqq 0.1 \cdot \left[1-\frac{H}{40}\right] \cdot Z$$

where,

H : Depth of each under part from the datum plane; 20 (m) at depths of >20 m,

Z : Zoning factor (to be 1.0, the regional difference is not considered),

Wk : Summation of dead loads and live loads of the part concerned.

Incidentally, in the case if the value would be calculated in evaluating the vibration characteristics suitably by considering the structural characteristics of buildings and structures, and the response characteristics and situation of the ground in the seismic condition, it would be the value calculated by this method.

2) Vertical seismic force

The vertical seismic force in the evaluation of the static force to Earthquake-proof S Class Facilities should be evaluated with the vertical seismic intensity by the following formula,

Cv = Rv・0.3

where,

Cv : Vertical seismic intensity,

Rv : A value representing the vertical vibration characteristics of the building , to be 1.0. However, based on special investigation or study, if it would be confirmed to fall short of 1.0, it would be reduced to be the value based on the results of investigation or study (but equal to or not less than 0.7).

3) Retained horizontal strength required

Retained horizontal strength required should be evaluated specified in the method in standards and criteria which are accepted to be adequate for safety.

Here, the standards and criteria which are accepted to be adequate for safety corresponds to the Building Standard Law etc.

Incidentally, in evaluation of retained horizontal strength required, the coefficient regarding the importance classification of the facilities which is multiplied by the seismic story shear coefficient should be settled to be 1.0 in all the case of Earthquake-proof S, B, C Class and standard shear force coefficient Co which is used in this case should be provided to 1.0.

7. Load Combination and Allowable Limit

The basic concept about combination of loads and allowable limits which shall be considered in assessing adequacy of design principle regarding seismic safety is as follows.

(1) Buildings and Structures

1) Earthquake-proof S Class Buildings and Structures

i) Combination with DBGM Ss and allowable limit

Regarding the combination of normal loads and operating loads with the seismic forces caused by DBGM Ss, the buildings and structures concerned shall have sufficient margin of deformation acceptability (deformation at ultimate strength)as a whole, and adequate safety margin compared to the ultimate strength of buildings and structures.

ii) Combination with EDGM Sd and allowable limit

Regarding resulted stress in combining the normal loads and operating loads imposed with the seismic loads caused by EDGM Sd or Static seismic force, allowable unit stress specified in standards and criteria assumed to be adequate for safety shall be established as the allowable limits.

2) Earthquake-proof B, C Class Buildings and Structures

Regarding resulted stress in combining the normal loads and operating loads imposed with Static seismic forces, allowable unit stress in above-mentioned 1) ⅱ) shall be established as the allowable limits.

(2) Components and Piping System

1) Earthquake-proof S Class Components and Piping System

i) Combination with DBGM Ss and allowable limits

The functions of Facilities shall not be affected by the occurrence of excessive deformations, crack and failure, even if the most part of structures would reach yield condition and the plastic deformation would occur, with respective resultant stress due to combined respective loads which occur in the normal operating condition, unusual transient condition in operation and accident condition with the seismic loads caused by DBGM Ss.

As for the active components etc., acceleration limit etc. for retaining of function shall be established as the allowable limit, which is confirmed by the verification test etc. regarding the response acceleration caused by the DBGM Ss.

ii) Combination of EDGM Sd with allowable limits

The yield stress or the stress with equivalent safety to this shall be established as allowable limits to respective resultant loads due to combined loads at normal operating condition, unusual transient condition in operation and accident condition imposed with the seismic loads caused by EDGM Sd or Static seismic force.

2) Earthquake-proof B, C Class Components and Piping System

The yield stress or the stress with equivalent safety to this shall be established as allowable limits to respective resultant loads due to combined loads in normal operating condition and unusual transient condition in operation imposed with the seismic loads caused by Static seismic force.

(Commentary)

IV. Regarding Load Combination and Allowable Limit

The interpretation of the combination of loads and allowable limits should be based on the followings.

(1) Regarding 'respective loads which occur in unusual transient operation and accident', if the load acted on by the events which are feared being caused by the earthquake and the loads, even if which are not feared being caused by the earthquake but being caused by the events which continue in long term if they would occur once, should be considered to be combined with the seismic load.

However, even if the load is 'a load which occurs in accident', considering the relation between occurrence probability of this accidental event and the duration time, and the exceedance probability of the earthquake, the load caused by this event needs not be considered to be combined with the seismic loads if the probability that the both of them occur simultaneously is extremely small.

(2) Regarding the allowable limits for combination of buildings and structures with EDGM Sd etc. though it was required to be established as the 'allowable unit stress specified in standards and criteria assumed to be adequate for safety', this standards and criteria correspond concretely to the Building Standard Law etc.

(3) 'Ultimate strength' in the terms regarding combination of the buildings and structures with DBGM Ss means the bounding maximum bearing load in reaching the condition, which is considered as the ultimate condition of the structures, where deformation and strain of the structure would increase remarkably by adding the load to the structure gradually.

(4) Regarding the allowable limit of components and piping system, though the basic principle requires to maintain the resulted stress under the ' yield stress or equivalent safety situation', this situation corresponds concretely to the situation specified in the 'Technical Standards on Structures etc. of Nuclear Power Generation Facilities etc.' which is prescribed in the Electricity Utilities Industry Law.

8. Consideration of the accompanying events of earthquake

Facilities shall be designed regarding the accompanying events of earthquake with sufficient consideration to the following terms.

(1) Safety functions of Facilities shall not be significantly affected by the collapses of the inclined planes around Facilities which could be postulated in the seismic events.

(2) Safety functions of Facilities shall not be significantly affected by the tsunami which could be postulated appropriately to attack but very scarcely in the operational period of Facilities.

(5) Regulatory Guide for Reviewing Technical Competence of Nuclear Operators

16 NSC Decision No. 6
May 27, 2004
Decision of Nuclear Safety Commission

Introduction

This guide is established in order to be used in the review of capability of a person who applies for license for business of nuclear fuel fabrication, spent fuel storage, spent fuel reprocessing, radioactive waste storage/disposal or establishment and operation of a reactor (hereinafter referred to as "business, etc") as provided for by the Law on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (hereinafter referred to as "the Reactor Regulation Law"), and to confirm whether the applicant has adequate technical capability to appropriately perform its business mentioned above so that the use of nuclear fuel material or a reactor cannot cause any disaster.

Establishment of the guide was prompted by the criticality accident which occurred at a nuclear fuel fabrication facility on September 30, 1999. The Nuclear Safety Commission decided to start preparation of a guide for examining technical capability of license holders, based on the interim and the final reports on the investigation of the accident. The reports are "The immediate measures for ensuring safety of nuclear power" decided by the Nuclear Safety Commission on November 11, 1999, and "A basic policy on the immediate measures of the Nuclear Safety Commission" decided by the Nuclear Safety Commission on January 17, 2000. The Special Committee on General Safety of Nuclear Power of the Nuclear Safety Commission started drafting the guide and compiled a report titled "A study on establishing a guide for technical capability (June, 2003)". The Special Committee for Nuclear Safety Standards and Guides of the Nuclear Safety Commission continued drafting the guide and reported to the Nuclear Safety Commission on March 24, 2004. The Nuclear Safety Commission invited public opinion on the report, discussed it, and decided "The Examination Guide for Technical Capability of License holders of Nuclear Power" on May 27, 2004.

The technical capability mentioned in this guide includes knowledge, technology and expertise of engineers of the organization as well as its organizational capability to ensure safety. This guide provides for the basic requirements to be met in obtaining license for the businesses mentioned above, license for establishment and operation of a reactor or license for change of the licenses (hereinafter referred to as "license, etc. for business").

In review of an application, it is necessary to confirm that contents of the application for license for business, etc. meet this guide. However, even in the case a part of the application does not comply with this guide, the application shall not be denied if the reason for non-compliance is adequate.

Furthermore, this guide shall appropriately be revised, as experience in the review of, and relevant knowledge of, technical capability accumulate.

I. Scope of application

This guide shall be applied to a person who applies for license for following businesses, etc. as provided for by the law:

i) nuclear fuel fabrication
ii) establishment and operation of a reactor
iii) spent fuel storage
iv) spent fuel reprocessing
v) radioactive waste storage/disposal

The fundamentals of this guide may be referred to in reviewing an application for license for a nuclear facility other than those mentioned above.

As the requirements in this guide are broadly divided into "design and construction" and "operation and maintenance", characteristics of the individual business shall be taken into consideration when applying each division of the requirements to the business.

II. Requirements

Guideline 1. Organization for design and construction

The license holder shall have established an appropriate organization which has clear assignment of performing design and construction adequately.

Guideline 2. Assignment of engineers for design and construction

The license holder shall appropriately have assigned engineers who have professional knowledge, technology, and expertise required to carry out design and construction.

Guideline 3. Experience in design and construction

The license holder shall have sufficient experience in design and construction of equivalent or similar facilities in the said business.

Guideline 4. Quality assurance activities for design and construction

The license holder shall appropriately have established a system for quality assurance activities required to adequately perform design and construction.

Guideline 5. Organization for operation and maintenance

The license holder shall have established an appropriate organization which has clear assignment of performing operation and maintenance adequately, or shall appropriately present its policy to establish such an organization.

Guideline 6. Assignment of engineers for operation and maintenance

The license holder shall appropriately have assigned engineers who have professional knowledge, technology, and expertise required to carry out operation and maintenance, or shall appropriately present its policy to assign such engineers.

Guideline 7. Experience in operation and maintenance

The license holder shall have sufficient experience in operation and maintenance of equivalent or similar facilities in the said business, or shall appropriately present its policy to obtain such experience.

Guideline 8. Quality assurance activities for operation and maintenance

The license holder shall appropriately have established a system for quality assurance activities required to adequately perform operation and maintenance, or shall appropriately present its policy to establish such a system.

Guideline 9. Education and training of engineers

The license holder shall appropriately present its policy for carrying out education and training of engineers for keeping and improving their professional knowledge, technology, and expertise.

Guideline 10. Selection and assignment of qualified persons

The license holder shall, when required by a law or regulations based on it, have selected and assigned qualified engineers for carrying out the said business, or shall appropriately present its policy to select and assign such qualified engineers.

(Annotation)

Guideline 1. Organization for design and construction

1) The scope of "design and construction" mentioned herein means stages until the license holder passes pre-service inspection relating the license, etc. for the said business. However, the scope of "design and construction" for radioactive

waste disposal means stages before the first waste package is received by the receiving facility, as pre-service inspection is not applicable to the business.

2) "The license holder shall have established an organization…" includes the case where a policy for establishing an organization in accordance with progress in design and construction is appropriately presented.

Guideline 2. Assignment of engineers for design and construction

1) "Professional knowledge" mentioned herein may include the knowledge required by the national qualification system, related to this business, such as chief engineer of reactor, chief nuclear fuel engineer, supervisor of radiation protection, chief engineer of boiler and turbine, chief electrical engineer, and professional engineer.

2) "The license holder shall have assigned engineers…" includes the case where a policy for assigning engineers in accordance with progress in design and construction is appropriately presented.

Guideline 3. Experience in design and construction

"The license holder shall have sufficient experience…" includes the case where experience and technology have sufficiently accumulated through dispatching engineers to equivalent or similar facilities in the said business at home or abroad or through training of engineers at related facilities, or the case where a policy for obtaining experiences in accordance with progress in design and construction is appropriately presented.

Guideline 4. Quality assurance activities related to design and construction

1) "The license holder shall establish a system…" includes the case where a policy for establishing a system for quality assurance activities in accordance with the progress in design and construction is appropriately presented.

2) "Quality assurance activities" mentioned herein include a scheme where a policy of the top management for ensuring safety in design and construction is decided, planning, implementation, evaluation, and improvement of the activities are conducted in accordance with the quality assurance plan, and continuous improvement of the activities is secured through auditing. Also, these activities shall be documented and kept.

3) The term "system" may include committees for comprehensive deliberation of the quality assurance activities, if necessary.

Guideline 5. Organization for operation and maintenance

1) The scope of "operation and maintenance" mentioned herein means stages at and after the start of operation of the facility following passing of the pre-service inspection relating the license, etc. for the said business. However, the scope of "operation and maintenance" for radioactive waste disposal means stages at and after the first waste package is received by the receiving facility, as pre-service inspection is not applicable to the business.

2) The term "organization" may include committees for deliberating matters related to operational safety, if necessary.

Guideline 6. Assignment of engineers for operation and maintenance

"Professional knowledge" may include the knowledge required by national qualification system related to the said business, such as chief engineer of reactors, chief nuclear fuel engineer, supervisor of radiation protection, chief engineer of boiler and turbine, chief electrical engineer, and professional engineer.

Guideline 7. Experience in operation and maintenance

"The license holder shall have sufficient experience …" includes the case where experience and technology have sufficiently accumulated through dispatching engineers to equivalent or similar facilities in the said business at home or abroad or through training of engineers at related facilities.

Guideline 8. Quality assurance activities related to operation and maintenance

1) "Quality assurance activities" mentioned herein include a scheme where a policy of the top management for ensuring safety in operation and maintenance is decided, planning, implementation, evaluation, and improvement of the activities are conducted in accordance with the quality assurance plan, and continuous improvement of the activities is secured through auditing. Also, these activities shall be documented and kept.

2) The term "system" may include committees for comprehensive deliberation of the quality assurance activities, if necessary.

Guideline 10. Selection and assignment of qualified persons

"Qualified persons" mean those who have a license for chief engineer of reactor or chief nuclear fuel engineer, or those who meet the criterion as person responsible for operation.

(6) Emergency Preparedness Guide, "Emergency Preparedness for Nuclear Facilities" (Excerpt)

(Decision of the Nuclear Safety Commission, June 1980)
(Latest Revision: May 2007)

The main points of the latest revision are shown in the following.
In line with international trends, such as International Atomic Energy Agency (IAEA), objectives of the guideline and the subject facilities etc. were clarified, and the effectiveness of the preventive protective actions was provided. And the overlaps with the Special Law of Emergency Preparedness for Nuclear Disaster and other guidelines relating to the Nuclear Safety Commission were arranged. The provisions of main revisions are provided below.

Chapter 1 Preface

1-1 Position of the Guideline

After the accident of the Three Mile Island (TMI) nuclear power station in U.S. that occurred in March 1979, the Nuclear

Safety Commission (hereinafter called as “the NSC”) studied technical and specialized matters focusing on events unique to nuclear emergency so that the emergency preparedness activities in the surrounding area of a nuclear power station etc. can be implemented more smoothly, and decided the "Emergency Preparedness of the Surrounding Area of Nuclear Power Stations etc." in June 1980 (after a partial revision in May 2000 referred to as the "Emergency Preparedness for Nuclear Installations", (hereinafter called as the "Emergency Preparedness Guideline").

The Emergency Preparedness Guideline is defined in the 10th volume, Nuclear Emergency Preparedness of the Basic Plan for Emergency Preparedness that the guideline shall be sufficiently considered on specialized and technical matters. NSC established the guideline on the specialized and technical matters concerning emergency measures, to support the national government, local governments and operators when they develop a plan of the Nuclear Emergency Preparedness and implement the measures in an emergency.

The Emergency Preparedness Guideline provide basic ideas for the environmental radiation monitoring and emergency exposure medical treatment in an emergency, and the details are specified in separate guidelines etc. issued by the NSC.

1-2 Scope of the Guideline

The scope of the Emergency Preparedness Guideline shall be the nuclear emergency of the following nuclear installations provided in the Reactor Regulation Law (limited to the subjects of the Special Law of Nuclear Emergency) and the nuclear emergency during transportation of nuclear fuel materials etc.

- Reactor facilities (excluding the reactors which are installed in a ship),
- Reprocessing facilities,
- Fuel fabrication facilities,
- Utilization facilities (limited to facilities that use nuclear fuel equal to or exceeding the critical mass), and
- Waste disposal facilities and waste storage facilities

1-3 Objectives of Protective Measures

The protective measures described in the Emergency Preparedness Guideline are implemented for the following four purposes. In addition, in implementing the measures, the principles of justification*[1] and optimization*[2] provided by the International Committee on Radiation Protection (ICRP) etc. should be followed, and it is important to take into consideration the results of the implementation of the protective measures concerned sufficiently.

- To prevent occurrence of deterministic health effect*[3] in residents in the vicinity, in nuclear installation workers, in those relevant in emergency preparedness, etc.,
- To render first aid and to manage the treatment of radiation injuries,
- To prevent, to the extent practicable, the occurrence of stochastic health effects in the population, and
- Mitigating the anxiety for health of residents in the vicinity, nuclear installation workers, relevant person on emergency preparedness, etc.

*[1] Justification: Implementing a protective action is justified when the benefit is larger than a damage due to a risk or other impact caused by implementing the action.

*[2] Optimizations: The radiation hazard avoided by each protective action should be balanced to the expense of the action and other damages so that the net benefit attained by the action becomes the maximum.

*[3] ICRP, IAEA etc. documents describe health effects of radiation, which are the deterministic effect, although the effect does not necessarily occur in all cases, which appears by an exposure of the levels more than a certain dose, and the stochastic effect that appears by an exposure of small dose. In addition, the IAEA document describes objectives of protective measures, which are prevention of deterministic effects and reduction of stochastic effects, which are fundamental principles of radiation protection.

5-3 Indices for Protective Measures

Indices to take protective measures are expressed as the dose (projected dose) expected to receive for individuals if certain measures are not taken, or measured values as concentration of radioactive materials in foods and drinks.

The projected dose will be presumed from the mode of an abnormal situation, the release situation of radioactive materials or radiations, emergency monitoring information, weather information, SPEEDI network system, etc.*[4]

*[4] IAEA documents etc. define indices for protective measures (sheltering / evacuation) with the dose (avertable dose) avoidable by taking measures. On the other hand, the Emergency Preparedness Guideline applies the projected doses.

This is because it becomes more safe side decision to implement protective measures by comparing the projected doses that are obtained assuming a certain period of time than the avertable doses which are obtained defining the period of implementing the protective measures, with the indices of protective measures, at the time of nuclear-emergency generation.

Annex 4 The NSC views on, and future actions to take for, the impacts due to the Niigata-ken Chuetsu-oki Earthquake in 2007

NSC Decision No. 17, 2007
30 July 2007

The Niigata-ken Chuetsu-oki Earthquake in 2007[1] (the Earthquake) on 16 July 2007 has strongly shaken the Kashiwazaki-Kariwa Nuclear Power Station of the Tokyo Electric Power Company, Inc. (TEPCO). The impacts caused include a fire breakout of the Unit 3 transformer, and a release of spilled water containing small amount of radioactive materials to the non-radiation control area and subsequently to the environment at Unit 6. A joint of the driving shaft of the overhead crane of the Unit 6 reactor building has been also found to have damaged.

No serious concerns about the environmental impacts have been identified so far. Nevertheless the Nuclear Safety Commission (NSC) of Japan believes that the damages suffered by the systems and equipment at the station leave us with big lessons in ensuring seismic safety of nuclear power plants.

Thorough investigations of the impacts on the nuclear power plants due to the Earthquake are intensively underway, whereas the growing interests are being raised among stakeholders overseas and in the country. Following are the NSC views, as of today, on the Earthquake impacts and future actions to take.

1. Immediate impacts of the Earthquake

(1) Ensuring major safety functions such as automatic reactor shutdowns

The Earthquake shook the plants with the maximum seismic accelerations exceeding the values assumed in the design, but all the units in operation or in power ascension (Units 2, 3, 4 and 7) were automatically shut down under control. Together with other units in the maintenance mode (Units 1, 5 and 6), all seven units at the Kashiwazaki-Kariwa Nuclear Power Station are now brought to the stable cold standby mode. Therefore, the emergency requirements of “shut-down, cool and contain” have been successfully met for ensuring nuclear safety.

(2) Investigation of impacts due to the Earthquake, and future actions

The incidents caused by the Earthquake are now under in-depth investigation and the total 64 cases on Units 1 to 7 (except four cases of automatic shutdowns due to the Earthquake) have been

[1] On July 16, 2007, at 10:13 (JST) there was a M6.8 (preliminary) earthquake at a depth of approximately 15km in the off-shore Chuetsu region, Niigata prefecture. Source: http://www.jishin.go.jp/main/chousa/07jul_chuetsu_oki/index.htm (in Japanese)

reported. Fifteen cases out of them are reported to be relevant to radioactive materials, but no cases have caused concerns of environmental impacts.

Thorough investigations are due for the reactor pressure vessel internals and other major safety-related components. Comprehensive evaluation of the impacts due to the Earthquake should follow incorporating such investigation results. The NSC will keep its awareness of the progress for necessary evaluation, receiving the relevant reports from NISA as their evaluation develops.

2. Ensuring seismic safety

(1) Requirements of the revised Seismic Safety Design Examination Guide and safety checks of existing nuclear power plants

a) Requirements of the revised Seismic Safety Design Examination Guide

The NSC revised in September last year (2006) the "Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities (Seismic Guide)." The revised Seismic Guide (the New Seismic Guide) requires the operators: (1) to investigate in detail the conditions of the active geological faults using latest technologies; (2) to analyze the ground motions with latest methods; and (3) to upgrade the formulation of the "Earthquake ground motions without the site specific epi-center." In doing so, the New Seismic Guide requires the operators to define the ground motions bigger than those by the earlier version of the Seismic Guide, using the latest knowledge and data, and to ensure the safety functions thereto.

b) Seismic safety checks of existing nuclear power plants (back checks)

Upon authorization of the New Seismic Guide in September last year (2006), the NSC requested through NISA all existing nuclear power plants, which had been designed based on the earlier version of the Seismic Guide, be reevaluated for seismic safety, referring to the New Seismic Guide (the so-called "Back checks") by the operators. In response, the back-check work is ongoing by the operators, and some of their results are being reviewed by NISA.

In this reevaluation processes, the complete and earliest checks are important, pursuant to the New Seismic Guide, on the formulation of the Design Basis Earthquake Ground Motion (DBEGM), reliabilities of the analytical models used in the design, and the use of latest knowledge made available after the original design. The operators' evaluation results are subject to the review by NISA and further by the NSC.

c) The effectiveness of the New Seismic Guide

It is important to make concluding judgment in the seismic safety, without any prejudges, based upon the scientific knowledge and facts. The necessity of revising again the New Seismic Guide should be judged after the new DBEGM be formulated in the back-check process and verified against the actual impacts due to the Earthquake. It is not the time to contend its necessity now.

The NSC will judge its necessity appropriately, in view of these verification results as well as the external experts' opinions.

(2) Detail understanding of shakes due to the Earthquake and additional investigation of geological faults around the plant site

The Earthquake has recorded at the Kashiwazaki-Kariwa Nuclear Power Station the inexperienced tremors far exceeding the values assumed in the original design. The operator should disclose the detail data, as early as possible, relevant to the Earthquake (seismometer recordings, etc.). The NSC will receive the report at its "Project Team on Seismic Safety Investigation (Seismic Safety PT, founded on 5 July 2007)" for necessary evaluation, as soon as the relevant information is made public.

Detailed investigation is needed concerning the active fault(s), which broke out the Earthquake at the Kashiwazaki-Kariwa Nuclear Power Station. As soon as the TEPCO investigation plan is fixed on the geological faults and seabed formations around the site, the NSC will receive the report at its Seismic Safety PT for its evaluation.

(3) Actions to take at all existing nuclear power stations

a) Confirmation of supporting capabilities of structures and systems

The New Seismic Guide prescribes in its basic policy "to install structures and systems on the ground with sufficient supporting capabilities." Namely, it requires them to be installed on the ground with sufficient supporting capabilities for design loads depending on their classes of importance, contrary to the earlier Seismic Guide, which had the similar requirements only to the safety important structures and systems. However, the uneven ground settlement due to the Earthquake has damaged quite a number of systems, components, piping and ducts, etc.

The NSC requests TEPCO to take necessary measures to meet the requirements of the New Seismic Guide, including the possible foundation improvement or reinforcement, for the structures and systems of Seismic Classes B and C in addition to those of Class S, after identifying the real damages due to the Earthquake.

This request should be applied to all existing nuclear power units in the back-check processes, not limited to those of the Kashiwazaki-Kariwa Nuclear Power Station.

b) Earliest back-checks and the disclosure of the results

The TEPCO back-check program made available in October last year (2006) foresees the completion of the work for the Kashiwazaki-Kariwa Nuclear Power Station by December next year (2008). However, the NSC requests the operators review their back-check programs and advance their geological investigation and formulation of the DBEGM. The NSC will receive the report on the results at its Seismic Safety PT for evaluation.

Further, for the Kashiwazaki-Kariwa Nuclear Power Station, the NSC thinks it important to disclose the results at the earliest practicality. The NSC requested NISA at its extraordinary session on 17 July 2007 to instruct the operator to submit the evaluation results as soon as individual work packages are complete.

c) Setting up of seismometers and record-taking of seismic data

Seismometers installed at every reactor building at the Kashiwazaki-Kariwa Nuclear Power Station after the“Niigata Chuetsu Earthquake in 2004 (Mid Niigata Prefecture Earthquake in 2004[2])” could collect a lot of valuable data. In the meantime, some of collected data were lost regrettably despite the experience at the Noto Hanto Earthquake in 2007[3]. The seismic data are extremely valuable in improving seismic safety of all nuclear power stations. It should not be limited to the safety checks for the subject specific earthquake. Adequate measures by all the operators are requested thereto. The NSC Seismic Safety PT will also check the status of seismometer installations at each nuclear facility and data keeping measures to prevent their losses.

d) Preparation of revising the safety examination guidelines concerning geologic and ground conditions

The “Safety Examination Guidelines Concerning Geologic and Ground Conditions (Geologic Guideline)” identifies the items to examine, concerning the geologic and ground conditions of the site for locating the nuclear reactor, when the safety examination is conducted pursuant to the Seismic Guide. The Geologic Guideline is being prepared for revision, as specified in the “Revision of relevant clauses on seismic guidelines for power generating nuclear installations (an earlier NSC decision on 19 September 2006). To this end, relevant information is being collected and analyzed. The work will be advanced and the preparatory revision work will be initiated at an appropriate timing for incorporating latest knowledge.

e) Earliest incorporation of latest knowledge

New knowledge obtained from the Earthquake should be evaluated at the earliest practicality and be incorporated in the back-check processes, as needed, including the lateral development to other existing nuclear power units.

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[2] On October 23, 2004, at 17:56 (JST) there was a M6.8 earthquake with a maximum seismic intensity 7, at a depth of approximately 10km in the Chuetsu region, Niigata prefecture. Source: http://www.jishin.go.jp/main/index-e.html

[3] On March 25, 2007, at 09:42 (JST) there was a M6.9 (preliminary) earthquake at a depth of approximately 10km near the west coast of the Noto Peninsula, Ishikawa prefecture. Source: http://www.jishin.go.jp/main/index-e.html

(4) Evaluation of “Residual risks”

The New Seismic Guide requests, in its commentary to the basic policy, the operators to pay due attention to the “residual risks (the risks of system damages causing dispersion of radioactive materials and radiation exposure of the public, due to the ground motions exceeding the assumed DBGM)” and to make efforts to minimize them to the maximum practical extent. Upon request, the operators are currently evaluating them, as part of their back-check processes.

Probabilistic risk assessment of the “residual risks” still has rooms for future development for applications. But the quantitative evaluation is requested to the operators on the test trial, which will advance and facilitate future full applications.

(5) Strengthening of safety research relevant to seismic safety

Operators, regulatory bodies and other research institutions are requested to strengthen and reinforce their research programs on seismic safety. In particular, precision improvement in the fault investigation in the ocean and on the land, or that in predicting the magnitudes of earthquakes should be advanced. Collaboration with the Headquarters for Earthquake Research Promotion[4] would be more than important.

The NSC will hold a nuclear safety research forum “Seismic safety and safety research” for the exchange of relevant information and knowledge as well as opinions on future research needs for seismic safety.

3. Trouble shooting of fires, etc. at earthquakes

(1) Trouble shooting frameworks for fires, etc. at earthquakes

The fire of the Unit 3 transformer at the Earthquake developed safety concerns among the public because of insufficient effectiveness of the private fire brigade, unavailability of fire control systems, and delayed notification to the external fire station, and as its consequence a lot of time needed to bring the fire under control. The operators are requested to establish a system, in which any necessary equipment and manpower are available at any time including holidays and nights, in preparation for unavailable assistance from external sources. Such systems should be prescribed in the respective safety rules of the operators. The NSC will conduct relevant subsequent regulation reviews in due course.

(2) Strengthening of fire protection measures at earthquakes

The Regulatory Guide for Reviewing Fire Protection of Light Water Nuclear Power Reactor

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[4] A special governmental organization, attached to the Ministry of Education, Culture, Sports, Science and Technology, was established in accordance with the Special Measure Law on Earthquake Disaster Prevention in July 1995 in the wake of the Great Hanshin-Awaji Earthquake Disaster.
http://www.jishin.go.jp/main/index-e.html

Facilities (the Fire Protection Guide) requires the fire control systems to be designed so as not to lose seriously its capabilities under concurrent earthquake conditions, depending on the classes of safety importance of reactor facilities. The NSC takes the preparatory actions to revise the Fire Protection Guide, taking note that the fire control systems did not function properly at the Earthquake.

4. Reporting system and information dissemination upon troubles

(1) Notification to the central and local governments and publicity

People point out that the notifications from the operator to the central and local governments were delayed, and the publicized contents were not easily understandable to the public. The NSC requests the operators and NISA to reconsider the notification and publicity system for improvement.

(2) Accountability to the nation

In order to mitigate nuclear safety concerns among the public, most important is for the operators and NISA to recover the public trust and foster their understanding. Their activities in this regard contribute to the further improvement of nuclear safety. The constant activities by the operators and NISA are requested to ensure the transparency of information, and to disseminate to the public relevant information on radiation safety. The NSC also accounts for proactively to the public on the ensuring of seismic safety.

(3) Information sharing internationally

NISA as well as the NSC have been promoting the information sharing with the IAEA and other nations. It is our nation’s responsibility as one of the most earthquake-ridden countries to disseminate the new knowledge learned from the Earthquake to the world for improving safety. Japan shares the lessons with the IAEA delegation of investigation. NSC also takes actions to disseminate relevant information internationally.

5. Conclusions

The NSC is in a position to avoid any prejudges on nuclear safety, especially in the seismic safety. The NSC prioritizes the open-mind and learning attitudes of placing primary importance in the scientific knowledge and facts. The NSC adheres to its philosophy.

Annex 5

Answers to the Synopsis of the Relevant IAEA Safety Requirements

INTRODUCTION

In the third Review Meeting held in April 2005, the IAEA Secretariat was requested to relate their Safety Requirements to each article of the Convention. In response to this, the Secretariat formulated a synopsis in the Q & A style by referencing to the Safety Requirements Publications listed below and allocating the requirements to the corresponding articles. This document's purpose consists of our reply to the synopsis based on the Safety Requirements, which reflects its current status in Japan.

The original objective of the synopsis is to be used as a reference when preparing the National Report and for the Review Meeting, and may be used at the discretion of each State.

IAEA SAFETY REQUIREMENTS PUBLICATIONS

Reference Number	Title
GS-R-1	Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety
GS-R-2	Preparedness and Response for a Nuclear or Radiological Emergency
GS-R-3 (DS338)	The Management System for Facilities and Activities
NS-R-1	Safety of Nuclear Power Plants: Design
NS-R-2	Safety of Nuclear Power Plants: Operation
NS-R-3	Site Evaluation for Nuclear Installations
SS115	International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources

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Article 7: Legislative and Regulatory Framework

	Question	Reply
GS-R-1, 2.2	(1) What legislative and governmental mechanisms are in place that define national requirements for the regulation of the safety of facilities and activities?	Concerning commercial power reactors, which are the subject of this IRRS review, the safety of their facilities and activities is regulated by the Reactor Regulation Law, the Electricity Utilities Industry Law, and other laws. Other facilities and activities are also regulated by the Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors, and other laws. For an overview of these regulatory measures, please refer to the Article 7-related part of Japan's National Report under the Convention on Nuclear Safety. The details of the substance and procedures are determined by cabinet order, ministerial order, ministerial notice and ministerial instruction.
	(2) How is the Regulatory Body established?	The Ministry of Economy, Trade and Industry(METI), one of whose assigned duties is the safety regulation of nuclear power, is established under Law for Establishment of the Ministry of Economy, Trade and Industry (hereinafter referred to as the "METI Establishment Law") (cf. Article 2), and the Nuclear and Industrial Safety Agency undertakes the assigned duties (cf. Article 20). Concerning the other Ministries' jurisdiction, they are defined in each Ministry's Establishment Law.
	(3) How is the responsibility for licensing nuclear installations; regulatory review and assessment; inspection and enforcement assigned to the regulatory body?	The METI Establishment Law states the mission of the Ministry of Economy, Trade and Industry (METI) as follows: "The Ministry of Economy, Trade and Industry shall engage in the growth of [a] economy and industries with emphasis on the enhancement of the economic vitality of the private sector and the harmonized development in Japan's external economic relations and [b] shall engage in ensuring stable and efficient supply of mineral resources and energy" (Article 3). To fulfill this mission, METI is assigned various duties (Article 4). One of these assigned duties is: "[a] matters relating to the regulations for [i] the refining, fabrication, storage, reprocessing and waste disposal business in nuclear fuel cycle and [ii] the nuclear power installations and [b] matters relating to ensure the safety of these businesses and installations"(Article 4, paragraph 1, subparagraph 57). According to the Article 20, paragraph 3, NISA is responsible for the duties.
	(5) How is it ensured that there are no responsibilities assigned to the regulatory body that may jeopardize or conflict with its responsibility for regulating safety?	According to the METI Establishment Law, Article 20, paragraph 3, NISA is responsible for the duties listed in Article 4, paragraph 1, subparagraphs 57 through 59, 62 and 64. As duties other than nuclear safety regulation, securing industrial safety (mining safety; gas safety; thermal, hydro and other power generation safety; and so on) is included. However, the duties for securing industrial safety is executed under separate laws, and the divisions responsible for the execution of such duties are separated from the divisions responsible for nuclear power related matters. Thus, executing such duties does not jeopardize nor conflict with the responsibilities of NISA concerning nuclear power safety regulation. Moreover, NISA has established common institutional objectives (securing the safety of the lives of the Japanese people and protecting the environment) and code of conduct (strong sense of duty, scientific and rational judgment, transparency in execution of duties, neutrality and fairness) and fulfills its duties accordingly.
GS-R-1, 2.4	How does the legislation: (1) Set out objectives for protecting individuals, society and the	 The Reactor Regulation Law, in Article 1, stipulates that the objective of said law is to conduct the necessary regulation,

	environment from radiation hazards, both for present and in the future?	etc., in order to prevent hazards due to nuclear source material, nuclear fuel material and reactors and to ensure public safety and so on.
	(3) Establish an authorization process?	The Reactor Regulation Law (including the Electricity Utilities Industry Law in the case of commercial power reactors) sets forth regulation, taking into consideration the potential magnitude and nature of the hazard, for each of the following categories: nuclear facilities and refining, fabricating, storage, reprocessing, and waste management and disposal. For example, in the case of commercial power reactors, the Reactor Regulation Law and the Electricity Utilities Industry Law sets forth regulation explicitly for each step along the way, from licensing of establishment, approval of construction plan, approval of operational safety program, approval of decommissioning program, and confirmation of the completion of decommissioning.
	(7) Establish a procedure for review of, and appeal against, regulatory decisions (without compromising safety)?	According to the Administrative Complaint Investigation Law, a disposition under the Reactor Regulation Law and the Electricity Utilities Industry Law can be appealed against the Minister of METI, who has made the disposition. Also, according to the Code of Administrative Procedure, a lawsuit to rescind the disposition can be lodged. The lodging of an appeal or a lawsuit does not in principle impede the effect, the execution, or the continuation of the disposition, so the lodging of an appeal or a lawsuit will not endanger nuclear safety.
	(8) Provide for continuity of responsibility when several successive licence holders carry out activities?	Concerning a commercial power reactor, someone who intends to take over such a reactor from the reactor establisher, according to the stipulation by the Reactor Regulation Law, must receive a license from the Minister of METI. The one who has received such license and taken over the reactor assumes the legal status of the reactor establisher. Also, when the reactor establisher enters into a merger, upon receiving a license from the Minister of METI, the legal entity surviving the merger or the legal entity to be incorporated upon the merger shall assume the legal status of the reactor establisher. In these cases, the transfer of the responsibilities of the reactor establisher shall be recorded as part of the licensing or approval procedures.
	(9) Allow for the creation of independent advisory bodies to provide expert opinion to, and for consultation by, the government and regulatory body?	The METI Establishment Law stipulates that the Advisory Committee for Natural Resources and Energy shall be placed in the Agency for Natural Resources and Energy. The Advisory Committee for Natural Resources and Energy surveys and deliberates important matters on consultation by the METI Minster and gives its opinion to the Minister of METI. Concerning nuclear safety, there is a Nuclear and Industrial Safety Subcommittee under the Advisory Committee for Natural Resources and Energy. The Advisory Committee and its Subcommittee are state agencies. Therefore, their independence from undertakers and their associations is assured.
	(10) Set up a means whereby research and development work can be undertaken in important areas of safety?	In order to enhance safety regulations, regarding scientific and rational aspects, and utilizing most advanced knowledge, NISA newly established "Subcommittee on Fundamental Policies for Nuclear Safety Infrastructure" in 2006 as one of the branch subcommittees of "The Nuclear and Industry Safety Subcommittee". And this "Subcommittee on Fundamental Policies for Nuclear Safety Infrastructure" is expected to express expert opinions and advices to NISA through reviewing needs for nuclear safety researches and development (hereafter referred to as “nuclear safety research”), verifying agenda of nuclear safety researches and etc.. Considering such opinions and advices, NISA is to conduct nuclear safety researches in accordance with real needs and to feedback the results of nuclear safety researches to regulatory

		process. In addition, in order to conduct research and development in areas important to nuclear safety in an organized and prioritized manner, the NSC establishes approximately every five years a promotion program for nuclear safety research and circulates it among the relevant organizations. Most recently, the July 2004 "Prioritized Nuclear Safety Research Program" covers 2005-2009 and summarizes the research areas that deserve priority and the promotional measures thereof. Based in this program, NISA provides grants to the Japan Nuclear Energy Safety Organization (JNES; incorporated administrative agency) and commission fees to the Japan Atomic Energy Agency (JAEA; incorporated administrative agency).
	(14) Define what is an offence and the corresponding penalties?	Concerning commercial power reactors, the Reactor Regulation Law stipulates what kind of infractions can lead to the revocation of a license for the establishment of a nuclear power reactor or the suspension of use of a nuclear power reactor. The Law also stipulates what kind of infractions can lead to criminal penalties.
	(15) Implement any obligations under international treaties, conventions or agreements?	Concerning treaties, the Japanese Constitution states: "The treaties concluded by Japan and established laws of nations shall be faithfully observed." Also, the Vienna Convention on the Law of Treaties states: "Every treaty in force is binding upon the parties to it and must be performed by them in good faith." Therefore, Japan performs its obligations under treaties that it has entered into by adopting new laws or amending existing ones.
	(16) Defines how the public and other bodies are involved in the regulatory process?	When issuing orders and other rulings under a law, according to the Administrative Procedure Law, it is necessary to seek public comment from the general public. Concerning specific dispositions by regulatory agencies, such as license for establishment of an commercial power reactor, according to the Administrative Complaint Investigation Law or the Code of Administrative Procedures, a party with legitimate interests can lodge an appeal or law suit to rescind such disposition. Among other matters, the relationship with other administrative organizations, such as the NSC, to whom the Minister of METI reports the status of its regulatory activity, and the relationship with other institutions, such as JNES, which conducts Fuel Assembly Inspection, are stipulated in the Reactor Regulation Law and the Electricity Utilities Industry Law.
	(17) Specify the nature and extent of the application of newly established requirements to existing facilities and current activities?	When introducing new regulation, including laws concerning nuclear safety regulation, Japanese legislation in general enacts transitional measures in a "supplementary provisions" to stipulate how existing facilities and ongoing activities shall be treated. Also, when determining the date of enforcement, we make sure that there will be sufficient time to inform the public before the new regulation is enforced.
GS-R-1, 3.2	How does the regulatory body establish regulations and guides, and assessment principles and associated criteria upon which its regulatory actions are based?	Ordinances are issued in the name of the Minister of METI, and documents are issued in the name of the Director-General of NISA. NISA strives to make them know and understand thoroughly, and to apply them strictly. For details, see Article 7 of this report.
GS-R-1, 3.3	How does the Regulatory Body: (9) Ensure that its regulatory principles and criteria are adequate and take account of international standards and recommendations?	In deliberating the transition to performance-based technical standards for nuclear power generation equipment and other matters, NISA considers internationally endorsed standards and recommendations, for example by seeking to harmonize them with international standards, taking into consideration such standards as the IAEA Nuclear Safety Standards (NUSS) and the US NRC 10CFR. Also, in developing its regulatory principles and criteria, NISA

		follows the safety examination guidelines established by the NSC, and the regulatory principles and criteria of the safety examination guidelines are based on the standards and recommendations of ICRP etc. as needed. In addition, the Radiation Review Council, in the efforts toward the domestic assimilation of the International Commission on Radiological Protection (ICRP) Recommendation of 1990 (Pub.10), has issued the "Technical Standards Concerning the Assimilation of the New Recommendation (pub.60) of the International Commission on Radiological Protection", and otherwise takes international standards and recommendations into consideration when deliberating domestic countermeasures for radiation protection.

Article 8: Regulatory Body

8.1

	Question	Reply
GS-R-1, 3.1	How does the Regulatory Body define policies on its regulatory actions?	NISA, while recognizing that the primary responsibility of nuclear safety belongs to the operator, upholds the following three as regulatory ideals: 1. Safety regulation is clear and open to the public. 2. Safety regulation is effective regulation that reflects the latest scientific knowledge and experience. 3. Safety regulation works proactively with international trends.
GS-R-1, 4.1	How is the Regulatory Body structured to discharge its responsibilities? How is the structure and size matched to the extent and nature of facilities and activities it regulates?	Please refer to the Article 8 (8.3 (2)).
GS-R-1, 4.2	If the Regulatory Body consists of more than one authority, what arrangements are there to ensure that duplication or omissions are avoided and conflicting requirements are not placed on the license holder? How are its main functions organized to ensure consistency and to enable feedback and exchange of information?	The Nuclear and Industrial Safety Agency has sole responsibility for regulating commercial power reactors. In terms of the relationship with the other authorities concerned, see 8.7 of this report.
GS-R-1, 4.3	Is the Regulatory Body self-sufficient in all technical and functional expertise? If not, how does it seek advice or assistance that is independent of the license holder?	Regarding review and assessment, NISA is entirely self-sufficient, while it sometimes has JNES, established by the state based on the legislation, conduct computer analysis in order to confirm the appropriateness of computer analysis submitted by an applicant concerning anticipated transients during operation, accidents, radiation exposure, etc., as required. Regarding inspection, NISA is also entirely self-sufficient. Inspection is conducted by NISA or JNES, not consultants. NISA solicits views of experts in the Nuclear and Industrial Safety Subcommittee as necessary.
GS-R-1, 4.5	How has the Regulatory Body established and implemented arrangements for quality management?	NISA spares no pains to disclose information to the public, accept third-party judgment, and has continuously worked to improve its regulation activities by continuously reviewing its activities and institutional framework. Moreover, in January 2007, in order to ensure further improvement in the quality of its regulations and transparency and fairness thereof, NISA has have established the “Quality Management Manual” in compliance with the IAEA GS-R-3 management standards, and have begun executing its operations accordingly.
GS-R-1, 4.6	How does the Regulatory Body ensure that it employs a sufficient number of personnel with the necessary skills to undertake its functions and responsibilities?	NISA has approximately 800 officials, of whom about 330 are responsible for nuclear safety.(as of end of January 2007) NISA provides training so that they can acquire sufficient capabilities required to execute their duties.
GS-R-1, 4.7	How does the Regulatory Body ensure that its staff has the relevant competencies? What education and training program does the regulatory body have for its technical and professional staff?	As shown in Table 11-1 (Article 11), in order to have NISA officials acquire and maintain the knowledge and skills to appropriately execute examination and inspection activities, NSA has established a nuclear safety human resources development program. Under this program, officials undergo training according to their levels of knowledge and skill so that they can acquire the proper skills, knowledge about state-of-the-art technology, and new principles and concepts that are required in regulatory activities. Specifically, NISA

		actively promotes human resources development for nuclear safety administration through special, dedicated training by the Nuclear Safety Training Office in the Training Institute of Economy, Trade and Industry, the training institution for METI officials, as well as through on-the-job-training at 21 Nuclear Safety Inspector Offices around the country, long-term training at JAEA, overseas training at safety regulation agencies in the US, UK and France, and other means.
GS-R-1, 4.8	How does the Regulatory Body ensure that their staffs have sufficient expertise to either perform regulatory reviews directly, or evaluate the work of consultants?	Safety Examiners in the Nuclear Power Licensing Division and other divisions undertake examination and evaluation activities as their sole responsibilities. Senior experts are assigned to core examination positions such as Senior Safety Examiner and a leader of examiners. Meisters are assigned to positions that oversee the whole examinations and hold responsibility for them such as Director for Safety Examination. It requires in general approximately 10 years to become a senior expert.
GS-R-1, 4.9	Does the regulatory body or Government use advisory bodies to give independent advice? How is it ensured that the advice does not relieve the regulatory body of its responsibilities to make decisions and recommendations?	The Nuclear and Industrial Safety Subcommittee has been established in the Advisory Committee for Natural Resources and Energy as the institution to deliberate and make proposals concerning nuclear safety regulation. This is established to hear the high-level and specialized opinions of experts, and a Cabinet Decision states that ultimate policy-making decisions are made under the responsibility of administrative organs.
GS-R-1, 4.11	How are arrangements established for the exchange of safety related information, bi-laterally or regionally, with relevant intergovernmental organizations to fulfill safety obligations and promote cooperation?	Japan does the following to exchange safety-related information. I. Bilateral and regional level: Japan has bilateral nuclear agreements for the peaceful use of nuclear energy with the United States, France, Germany, Sweden, Republic of Korea, China, Australia, Canada, the UK and Russia. Japan also has bilateral agreements concerning nuclear power generation safety with the United States, France, the UK, Republic of Korea and China, and conduct periodical meetings for the exchange of safety-related information. II. Neighboring states and other interested states Japan is a party to the Convention on Early Notification of a Nuclear Accident and the Convention on Assistance in the Case of Nuclear Accident or Radiological Emergency. Japan and China shall promptly notify each other in the case of serious accidents concerning nuclear power generation plants under an agreement with the objective of promoting the safety level of commercial power reactors. Japan and the Republic of Korea shall establish an early notification network and conduct cooperation activities based on intergovernmental consultations. Moreover, Japan is consulting with China and the REPUBLIC OF Korea with the objective of further enhancing nuclear power generation safety in Northeast Asia as whole measures, such as the development of a framework for regional cooperation, to reinforce coordination between the nuclear power safety regulatory agencies in Northeast Asia nuclear power generating countries.

8.2

GS-R-1, 4.1	How does the Regulatory Body maintain its independence in the governmental infrastructure?	The Nuclear and Industrial Safety Agency (NISA) was established on January 1, 2001, within the reorganization of national administrative organs as a “special organization” at the Ministry of Economy, Trade and Industry, an administrative organ of the Japanese Government, to conduct safety regulation of nuclear. NISA is effectively independent from the Agency for Natural Resources and Energy, which is

		responsible for nuclear power technology development and facilities and so on, through mechanisms such as inter alia provisions in the decision-making mechanisms and supervision and auditing by the Nuclear Safety Commission (the NSC), and this independence of NISA is secured.
GS-R-1, 2.2 (2)	How is it ensured that the regulatory body is effectively independent of organizations or bodies charged with the promotion of nuclear technologies or responsible for facilities or activities?	Same as above.

Article 9: Responsibility of the Licensee

	Question	Reply
GS-R-1, 2.3	How is the prime responsibility for safety assigned to the license holder for siting; design; construction; commissioning and operation? How do legislative and governmental mechanisms ensure that compliance with the requirements imposed by the regulatory body does not relieve the operator [licensee] of its prime responsibility for safety? How does the operator [licensee] demonstrate to the regulator's satisfaction that this responsibility for safety has been and will continue to be discharged?	It is stipulated in laws and ordinances that the operator has the primary responsibility for the safety concerning a commercial power reactor. 1) Reactor Regulation Law When there is intent to establish a nuclear reactor, the operator must receive a license from the Minister of METI for matters including siting, and the securing of safety in basic design and basic design policies. The operator must establish operational safety program (including the details of the operation and control of nuclear reactor, operation limits, and safety education) and have them approved by the Minister of METI before starting operations. The operator has the obligation to take measures necessary to ensure safety concerning each of the following: (1) Maintenance of reactor facilities; (2) Operation of reactors; and (3) Transportation, storage, and disposal of nuclear fuel material or material contaminated by nuclear fuel material. Where there is a violation of these requirements, the Minister of METI can issue orders including suspending use of reactor facilities. 2) Electricity Utilities Industry Law • When one intends to conduct construction of an electric structure (includes nuclear reactor facility; same hereafter), one must receive the approval of the Minister of METI for the construction plan for electric structures including items concerning the securing safety in detailed design and during construction. Also, the operator may not use the electric structure until it has been inspected and certified by the Minister of METI. • The operator must maintain an electric structure so that it is in compliance with technical standards. If there is an infraction, the Minister of METI can issue orders including suspending use of that electric structure. In this way, the undertaker has been assigned the primary responsibility for safety regulation of a commercial power reactor. Beyond these measures, the system for the operator to regularly review the operator's organization for the operator's review of the main facilities (Electricity Utilities Industry Law) and the system for the review of the observance of operational safety program (Reactor Regulation Law) require that the primary obligation of the operator are ensured and that they will continue to be appropriately maintained in the future.
GS-R-1, 2.1	How does the operating organization as licensee retain prime responsibility for safety when it delegates authority to the plant management for the safe operation of the plant? In such cases what resources and support does the operating organization provide for the plant management?	The operator, in taking measures required for safe operation of reactor facilities, must establish a quality assurance program to comply with the national requirements and clarify the organization in charge of its implementation. The quality assurance program must include the provisions concerning "human resources", "nuclear installations" and "work environment".
GS-R-1,	How does the operating organization	The licensee has the primary responsibility for the safety of

3.1	with overall responsibility for safety, ensure that interfacing organizations engaged in activities important to safety meet their responsibility to ensure that safety matters are given the highest priority?	nuclear installations and must observe legislation, etc. provided by the regulatory body. For details, see 9.1 of this report.
GS-R-1, 2.14 - 2.18	How is the interface between the operator and the regulatory body with regard to the responsibility for safety organized?	NISA strives to have sufficient opportunities to exchange opinions with licensees in order to have them understand its views on safety regulation and also to see their views on it. For details, see 9.3 of this report.

Article 10: Basic Policy for Priority to Safety

	Question	Reply
NS-R-2, 2.2; DS338, 2.2	What are the policies of the operating organization giving safety matters the highest priority?	The operator, in taking measures required for safe operation of reactor facilities, must establish a quality assurance program to comply with the national requirements, give the highest priority to safety matters in the program and clarify the policy in "Commitments" in the quality assurance program.
NS-R-2, 2.6	How does the operating organization make sure that its safety policy is applied by all site personnel?	In order to ensure safety of nuclear installations, safety culture in organization is vital. Licensees take various efforts to establish safety culture. For details, see 10.2 of this report.
NS-R-2, 2.3	How are management objectives set and how are these objectives related to the policy for nuclear safety and quality?	The operator, according to the quality assurance program giving the highest priority to safety matters, must establish the quality goal to fulfill the work requirements in each division and level in the organization.
NS-R-2, 2.3	How is safety monitored and followed up on a regular basis, timely corrective actions taken and opportunities for improvements used?	The operator, according to the quality assurance program giving the highest priority to safety matters, must implement management review to monitor and identify non conformity and confirm the implementation of preventive measures and corrective measures.
NS-R-1, 3.1	How is it ensured that the design organization takes into account the current state of the art for safety, and that the safety of any design change is properly considered?	The operator, according to the quality assurance program giving the highest priority to safety matters, must implement review, validation and verification in an alteration management process of design and receive an approval prior to its implementation.
NS-R-2, 2.5	How are proposed changes to the management structure and associated arrangements which might be significant to safety systematically reviewed by the operating organization and submitted to the regulatory body for review?	The items concerning operations management are specified in the operational safety program and an alteration to them must be submitted to and reviewed by the regulatory body.
NS-R-2, 2.6	How are clear lines of authority established to deal with plant safety matters?	Operational Safety Program states matters relating to duties and organizations, and clarifies the authority to deal with plant safety matters.
NS-R-2, 2.10	How is it ensured that all activities that may affect safety are performed by suitably qualified and experienced persons?	When licensing for establishment, technical capabilities of licensees are reviewed. Duties, education and capabilities required are stipulated in Operational Safety Program and the quality assurance program.
NS-R-2, 2.11	How is it ensured that all activities that may affect safety and which can be planned in advance are conducted in accordance with established procedures?	As described in 19.3 of this report, the operational safety program covering such activities is established by the operator and reviewed and approved by the regulatory body. Then, to confirm observance of the operational safety program, the operational safety inspection is implemented by the regulatory body.
NS-R-2, 2.12	What are the procedures to deal with activities that are not included in the normal procedures?	As described in 19.3 of this report, a special review committee is established to examine any off-normal activity such as alteration of an operational safety program.
NS-R-2, 2.13	How is it ensured that an appropriate safety consciousness and safety culture prevail in plant operations?	Operational Safety Program states matters relating to duties and organizations, and clarifies the authority to deal with plant safety matters.

Article 11: Financial and Human Resources

11.1

	Question	Reply
DS338, 4.1	How does the licensee determine the necessary material and financial resources to carry out the activities of the organization? How are financial resources made available to perform safety improvements? How are financial resources made available to cope for any waste management activities resulting from the operation of the facility? How are financial resources made available to cope for decommissioning activities after the termination of the operation of the facility?	When issuing license of a nuclear installations, financial basis is reviewed. The operator, in taking measures required for safe operation of reactor facilities, must establish a quality assurance program to comply with the national requirements and clarify the organization in charge of its implementation. The quality assurance program must include the provisions concerning "human resources", "nuclear installations" and "work environment". The operator, according to the Electricity Utilities Industry Law, etc., deposit reserves for waste storage outside of reactor facilities and decommissioning activity, as described in 11.1 (2) of this report.
DS338, 4.3	How are competence requirements determined for individuals at all levels?	The licensee, according to the quality assurance program giving the highest priority to safety matters, must clarify the competence required for the personnel, implement education and training and assess them. The regulatory body classifies the competence requirements for staff into four levels (Entry, Expert, Senior Expert and Meister). The regulatory body specifies the contents of study and training required for the achievement of each level of competency, and implements it.

11.2

DS338, 4.3	How does the licensee provide training or takes other actions to achieve the required level of competence?	The licensee, according to the quality assurance program giving the highest priority to safety matters, must clarify the competence required for the personnel, implement education and training and assess them.
NS-R-2, 3.1, 3.4	How does the license holder define the qualifications and experience necessary for personnel performing duties that may affect safety? What provisions are in place to select suitably qualified personnel and given the necessary training and instruction to enable them to perform their duties correctly, including managerial and supervisory skills?	The licensee, according to the quality assurance program giving the highest priority to safety matters, must clarify the competence required for the personnel, implement education and training and assess them.
NS-R-2, 3.3	What programs are in place for training personnel before their assignment to safety related duties?	As described in 11.2 (2), 2) of this report, the items concerning the education for the operation and management personnel are specified in an operational safety program. The relevant legislation and regulation are as follows. 1. The Rules for the Installation, Operations, etc. of Commercial Power Reactors, Article 7-3, "Quality Assurance" stipulates that a licensee must establish a quality assurance program. To comply with the requirement, a licensee establishes a quality assurance program in the operational safety program. The quality assurance program specifies the operational management of resources including items such as required competence, education and

		training. 2. The Reactor Regulation Law, Article 37, “Operational Safety Program” and the Rules for the Installation, Operations, etc. of Commercial Power Reactors, Article 16, “Operational Safety Program” stipulate that the operational safety program must specify the items about an education on operational safety for the operating personnel and management of the reactor installation. To comply with the requirement, the operational safety program specifies that a plan for education on operational safety be established and implemented. The situation of the implementation is regularly confirmed in the operational safety inspection by the regulatory body.
NS-R-2, 3.4	How does the license holder ensure that all personnel who may be required to perform safety related duties have sufficient understanding of the plant and its safety features?	The licensee, according to the quality assurance program giving the highest priority to safety matters, must make the personnel understand the implication and importance of their activities and their contribution to the achievement of quality goal through education and training process.
NS-R-2, 3.5	How does the license holder ensure that the qualifications and training of external personnel performing safety related duties are adequate for the functions to be performed?	The licensee, according to the quality assurance program giving the highest priority to safety matters, must evaluate contractors based on established standards in a procurement process and conduct necessary inspection to ensure that the procurement requirements are satisfied.
NS-R-2, 3.6	What provisions are there for periodic confirmation of the competence of personnel and for refresher training?	The licensee, according to the quality assurance program giving the highest priority to safety matters, must evaluate the effectiveness of education and training.
NS-R-2, 3.7	Who provides the training organization with the necessary resources and facilities? Who determines the need for training, and ensures that operating experience is taken into account in the training? How is it ensured that production needs do not interfere with the conduct of the training program and the need for personnel to be trained?	The licensee, according to the quality assurance program giving the highest priority to safety matters, must clarify the resources required for nuclear safety and provide them. And, the operator must implement education and training to ensure the required competence.
NS-R-2, 3.9	How is it ensured that training instructors are competent in their assigned areas of responsibility and have the necessary instructional skills?	The licensee, according to the quality assurance program giving the highest priority to safety matters, must clarify the competence required for personnel with duties that may influence nuclear safety and evaluate its effectiveness.
NS-R-2, 3.11	What simulator facilities are used for the training operating personnel on operational states and for accidents?	Please refer to Tables 11-2 and 11-3 (Article 11) of the Main Report.
NS-R-2, 3.12	What instruction is given to plant staff on the management of accidents beyond the design basis?	The licensee must implement an education on accident management and provide a training course with a training simulator.
NS-R-2, 3.13	What is in place to assess and improve the training programs, and modify and update the training facilities and materials to ensure that they accurately reflect plant conditions?	The licensee, according to the quality assurance program giving the highest priority to safety matters, must identify duties throughout the whole process from work plan to its implementation including training, etc., and implement them in a controlled manner.
NS-R-2, 3.14	How is operating experience of events at the plant and relevant events at other plants factored into the training program?	The licensee shall, by in-house nonconformity management process for any abnormal event in own plant and by the nuclear information library (NUClear Information Archives: NUCIA) for any information in other plants, examine an action including any alteration to a training program and

		implement it.
NS-R-2, 2.4	What provisions did the license holder take to established liaison with organizations for design, construction, manufacturing and plant operation and with other organizations (national and international) as necessary to ensure the proper transfer of information, expertise and experience to respond to safety issues? Are the national resources for services and technical support adequate?	The licensee, according to the quality assurance program giving the highest priority to safety matters, acquires information, expertise and experience from suppliers as a part of procurement management. In addition, the operator can share necessary information through an owners' group organized in cooperation with plant manufacturers.
GS-R-1, 5.13, (3)	Are the competence requirements, the qualification, training and re-training activities of the licensee subject to regulatory inspection?	Specialists such as the Chief Engineer of Reactors, the Chief Electrical Engineer and the Chief Engineer of Boiler and Turbine are qualified by the competent authorities having a jurisdiction over respective license. While the operator implements education of its employees, the state of the education and training is confirmed by the regulatory body through the inspection on the observance of operational safety program.

Article 12: Human Factors

	Question	Reply
NS-R-1 5.48, 5.49	How are designs made ‘operator friendly’ and limit the effects of human errors by plant layout, work areas, working environment and procedures (administrative, operational and emergency), including maintenance and inspection, in order to facilitate the interface between the operating personnel and the plant?	As described in Article 12 of this report, human factors are taken into consideration in designing the main control room and in an operations management aspect.
NS-R-1, 5.50	How is consideration of human factors and the human–machine interface take into account, provided?	As described in Article 12 of this report, human factors are taken into consideration in designing the main control room and in an operations management aspect.
NS-R-1, 5.51	How does the human-machine interface provide the operators with comprehensive, easily manageable information, compatible with the necessary decision and action times? How are similar provisions made for the supplementary control room?	As described in Article 12 of this report, human factors are taken into consideration in designing the main control room and in an operations management aspect.
NS-R-1, 5.52	How are verification and validation of aspects of human factors included at appropriate stages to confirm that the design adequately accommodates all necessary operator actions?	As described in Article 12 of this report, human factors are taken into consideration in designing the main control room and in an operations management aspect.
NS-R-2, 3.2	Is there a program to ensure fitness for duty?	There is no specific regulatory requirement. The operator must take appropriate measures timely to assure compliance with related legislation and an operational safety program while giving priority to safety.

Article 13: Quality Assurance

	Question	Reply
DS338, 2.1	How does the management system bring together all the requirements for managing the nuclear installations actions to provide confidence that these requirements are satisfied and that quality requirements are not considered separately from safety requirements?”	Please refer to 7.3, 10.2, 13.1 and 19.3 of this report. As for the activities of the regulatory body, (1) i) NISA set forth the Management Manual, which determines NISA's management system in January 2007, and began administration of its management system. ii) The Manual adheres to the "Management Policy", which lays out NISA's objectives, etc. NISA's management system, which is based on the manual, also adheres to NISA's organizational objectives. (2) i) “Work Management Guidelines” which specifies NISA's management system, referring to objective evaluation standards such as IAEA (GS-R-3), etc., integrate all the requirements to operate and administrate the regulatory body in a consistent manner. (3) i) The “Work Management Guidelines” establish the planned and systematic measures to appropriately implement the requirements of IAEA (GS-R-3) etc. : for example, the establishment of “Work Management Committee” within NISA, with the Director-General as its head; the establishment of “Work Implementation Plan” for individual divisions and NISA as a whole; and the establishment of a mechanism to implement, inspect and assess the duties under the plan. (4) The management system of NISA is in the early stage of its operation, to be improved successively as the experience accumulates. The operator must establish the quality assurance in the operational safety program.
DS338, 2.6	How are management system requirements graded to deploy appropriate resources relative to the safety significance, hazards and risks, and possible consequences of failure?	As for the activities of the regulatory body, i) Under the Quality Management Manual, which determines the NISA management system, the “Quality Management Committee” is established within NISA, with the Director-General as its head. Also established under the Quality Management Manual are the “annual work plan” for individual divisions and NISA as a whole. A mechanism to implement, inspect and assess activities under the work plans has been established. ii) Under the “annual work plan”, items for priority operational improvement from a mid- to long-term point of view (operational improvement priority) have been selected in order to prioritize activities. (Quality Management Manual Chapter 10 Improvement of Work, (2) Priority for Improvement of Work) iii) Activities are prioritized at the individual officials’ level as well as through such means as the establishment of operational goals in conformance with the “annual work plan”. iv) Therefore, through the measures such as prioritization of works in the aforementioned “Work Implementation Plan”, activities of “Work Management Committee” to inspect and assess the plan and implementation of “work assessment”, a graded classification of duties, which contributes to the appropriate allocation of resources, can be realized. The operator shall, establishing a quality assurance program giving the highest priority to safety to comply

		with the national requirements, clarify and assure the resources required for nuclear safety.
DS338, 3.1	How does management demonstrate its commitment to implementation, assessment and continued improvement of management systems, including allocation of adequate resources?	As for the activities of the regulatory body, (1) i) Directors and the Director-General each creates an Annual work plan for his/her division and NISA, and executes, assesses, and makes activities improved under the Annual work plan, based on Quality Management Manual, Chapter 3 Duty of Organization, (3) Planning; and Chapter 5 Work Management. ii) As members of the Quality Management Committee, senior management and directors participate in the operation of the management system and execute their responsibilities, based on the responsibilities and authority stipulated by the Quality Management Manual, Chapter 3, Duty of Organization, (4) Responsibility and authority. (2) i) In order to facilitate the execution of NISA's duties, the Director-General also establishes NISA's institutional framework that includes the appropriate allocation of resources and exercises overall direction and supervision over the execution of NISA's duties. ii) Other senior management members and directors participate in the operation of the management system including the appropriate allocation of resources, and execute such responsibilities as members of the Quality Management Committee or by responsibilities and authority stipulated by the Quality Management Manual Chapter 3 Duty of Organization, (4) Responsibility and authority. The operator, according to a quality assurance program established to comply with the national requirements, must specify continual improvement of the effectiveness of a quality management system and the assured implementation of review to maintain appropriateness.
DS338, 3.8	How does management establish goals, strategies, plans and objectives (sometimes known as the business plan)?	As for the activities of the regulatory body, i) The Quality Management Manual, which defines the NISA management system cites the "Management Policy" and sets forth NISA objectives and code of conduct and other matters of institutional policy. ii) The Director-General determines the Management Policy. The Policy states NISA's fundamental policies such as "Fundamental Philosophy of NISA", "Basic view concerning Risks, Safety and Public Confidence", "Reforming Institutional Management", and ""How to perform Activities -commonly followed by all individuals". iii) Moreover, the "annual work plan" that the individual divisions and the Quality Management Committee each establishes contains the mid-term objectives and the annual work plan that the divisions and NISA as a whole each undertake. Each such annual work plan is in conformance with aforementioned institutional policies. iv) Concerning an annual work plan, the Quality Management Committee, with the Director-General as chairman, is established and examines and evaluates the contents of the annual work plan. The conformance between the institutional policy and the annual work plan can be confirmed through the Committee. The operator, as described in 10.2 (2) of this report, must clarify the organizational goal, policy, plan and objectives for safety by the top management in a quality assurance program.
DS338,	How are individuals given	As for the activities of the regulatory body,

3.12, and 3.13	responsibility and authority within the management system, including when external organizations are involved in the system?	Senior management fulfills its duties by participating in the management of the management system within NISA as members of the Quality Management Committee and through the responsibilities and authority. Each division implements work allocated to it. The director of each division is responsible for executing and managing the allocated work. The operator must specify organization and duties for the personnel in the operational safety program.
DS338, 4.1	How does the license holder determine the resources necessary to establish, implement, assess and continually improve the management system?	The operator, pursuant to a quality assurance program established to comply with the national requirements, must clarify and assure the resources required for nuclear safety.
DS338, 5.1	How are management system processes identified and their development planned, assessed and continually improved?	As for the activities of the regulatory body, The Quality Management Manual establishes the "Quality Management Committee" within NISA, with the Director-General as chairman, and inter alia mandates the creation of "annual work plan" for individual divisions and NISA as a whole, and establishes a mechanism to implement, inspect and assess activities under the work plans. In this manner, the process of the management system has been established. i) Directors and the Director-General each creates an Annual work plan for his/her division and NISA, and executes, assesses, and makes activities improved under the plan. ii) The "annual work plan" shall be accompanied by charts that show an overall outline and procedures on the respective division (organization and process charts). Processes and other measures shall be undertaken as appropriate. As for the licensee, refer to 13.3 of this report.
DS338, 5.11 - 5.28	What processes are covered by the management system?	As described in Article 13 of this report, the QMS is established. The QMS covers all the items including document management, product control, recording management, procurement control, communication and management of organizational alteration.
DS338, 6.3	What independent assessments of the management system are there?	As for the activities of the regulatory body, NISA activities are periodically evaluated by third parties (Nuclear and Safety Subcommittee). NISA is also supervised and audited by the NSC. The licensee, pursuant to the quality assurance program established to comply with the national requirements, must assure the objectivity and fairness of an overseeing process and let an auditor to audit duties other than its own duty. And, the operator must receive the operational safety inspections by the Nuclear and Industrial Safety Agency for the items specified in an operational safety program.
DS338, 6.7	How is the system reviewed to ensure continued suitability and effectiveness, and enable accomplishment of objectives?	As for the activities of the regulatory body, i) It is stipulated that the Quality Management Committee can establish and amend the Quality Management Manual, which determines the NISA Management System. ii) The Quality Management Manual is to be reassessed as necessary, taking into consideration appropriateness, effectiveness and other relevant factors, and following review by the Quality Management Committee. The operator, pursuant to a quality assurance program established to comply with the national requirements, must review a quality management system in prescribed interval to ensure that the quality management system of

		the organization continues to be appropriate, valid and effective.
DS338, 6.11	How are non-conformances and remedial actions dealt with?	As for the activities of the regulatory body, i) When a division becomes aware of a non-conformance, the division shall expeditiously report it to the Quality Management Committee. ii) When the Quality Management Committee receives the report, it shall expeditiously investigate the cause and implement corrective actions proportionate to the seriousness of the event. The operator, pursuant to a quality assurance program established to comply with the national requirements, must implement nonconformity management and corrective measures.
DS338, 6.17	How are opportunities for improvement of the management system identified and, where appropriate enacted?	As for the activities of the regulatory body, i) Concerning each division's annual work plan, it is stipulated that the division should conduct a self-assessment at the end of the fiscal year, whose results shall be subject to a hearing by the Quality Management Committee. ii) NISA's work is constantly reviewed, and shall be continuously improved, after review by the Quality Management Committee. The operator, pursuant to a quality assurance program established to comply with the national requirements, must investigate a possibility to improve the quality management system and also assess a need to alter the quality management system including policy on quality and quality goal, by management review process.
NS-R-1, 5.48, and 5.49	How are designs made 'operator friendly' and limit the effects of human errors by plant layout, work areas, working environment and procedures (administrative, operational and emergency), including maintenance and inspection, in order to facilitate the interface between the operating personnel and the plant?	As described in Article 12 of this report, human factors shall be taken account of when designing a main control room and human factors and human error prevention shall be taken account of in the operational management.

Article 14: Assessment and Verification of Safety

14(i)

	Question	Reply
NS-R-1, 3.10	How is a comprehensive safety assessment carried out to confirm that the design as delivered for fabrication, as for construction and as built meets the safety requirements set out at the beginning of the design process?	As described in 14.1 and 14.2 of this report, a safety assessment by the regulatory body is carried out at the review process of a license for establishment, an approval of construction plan and a pre-service inspection.
NS-R-1, 3.13	How does the operating organization ensure that an independent verification of the safety assessment is performed by individuals or groups separate from those carrying out the design, before the design is submitted to the regulatory body?	It is not a regulatory requirement that the operating organization have an independent group verify the design for establishment license. The operator, when preparing an application for establishment license, confirms that the design should comply with the requirements of the Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities, etc.
NS-R-2, 4.1	How is the approval granted by the regulatory body before starting normal operation?	As described in 7.2 (4) of this report, the regulatory body, begore the start of normal operation, approves the operational safety program of the operator, which specifies concrete methods for operations management. And, as described in 14.2 (1) of this report, the regulatory body verifies that reactor facilities fulfill assigned functions and performance through measures such as pre-service inspection and the review of start-up test.
NS-R-1, 3.12	How is it ensured that licensees in their safety assessments use data derived from the safety analysis, previous operational experience, results of supporting research and proven engineering practice?	Please refer to 18.7 of this report, which describes the measures to assure the technical reliability by tests, experiences and analyses.
NS-R-2, 5.18	What safety reviews are undertaken if there is a need to conduct a non-routine operation, test or experiment?	The licensee, pursuant to the operational safety program, must establish a committee which reviews items significant to safe operation of nuclear reactor such as an alteration of the operational safety program or operational procedures (please refer to 19.3 of this report).
NS-R-2, 7.4	What procedures are established by the operating organization to ensure proper design, review, control and implementation of all permanent and temporary modifications?	7.2 (4), "Regulation at Operational Stage" of this report describes the measures to be taken for modification or repair of electric facilities after commissioning. For the management of a temporary modification, refer to the description of the preceding column on non-routine operation.
NS-R-1, 5.69, 5.73	How has the safety analysis of the plant design made use of the results of deterministic and probabilistic methods? How have the computer programs, analytical methods and plant models used in the safety analysis been verified and validated, and adequate consideration been given to uncertainties?	While safety assessment using deterministic method is described in 14.1 of this report, the effective use of probabilistic method is described in 14.4 and 14.6 of this report. The computer programs, analytical methods, plant models used in safety analysis and the associated uncertainties are examined in the process of safety review of the design.
NS-R-2, 10.1 - 10.6	What are the objectives, scope and frequency of Periodic Safety Reviews and how are the results are used?	14.3 (2) of this report describes periodic safety review and the evaluation for aging.
GS-R-1, 5.11	To what extent and how does the regulatory body review and assess modifications to safety related aspects of a facility or activity taking into	7.2 (4), "Regulation in Operational Stage" of this report describes the measures to be taken for modifications and repairs after commissioning. The Nuclear and Industrial Safety Agency establishes a procedure for authorization of

	account the potential magnitude and nature of the associated hazard?	alteration of the operational safety program.
GS-R-1, 5.7	How does the regulatory body when performing reviews and assessments take into account the potential magnitude and nature of the hazard associated with the particular facility or activity?	14.1 (2) of this report describes the evaluation of safety design.

14(ii)

NS-R-2, 6.1 - 6.3	How have the operating organization prepared and implemented a program of maintenance, testing, surveillance and inspection of those structures, systems and components which are important to safety?	As described in 7.2 (4) of this report, the operator must specify items concerning operations management of reactor facilities in an operational safety program and receive approval from the regulatory body before commissioning of the reactor facility. The regulatory body confirms observance of operational safety program by the operator through the operational safety inspection.
NS-R-2, 5.1	How is it ensured that operational limits and conditions reflect the provisions made in the final design?	As described in 7.2 (4) of this report, the operator must describe the operational limits and conditions in the operational safety program and receive approval from the regulatory body, which evaluates their adequacy.
NS-R-2, 5.5	How has the operating organization established and implemented an appropriate surveillance program to ensure compliance with the operational limits and conditions, and how are its results evaluated and retained?	As described in 7.2 (4) of this report, the operator must specify the operational limits and conditions in the operational safety program and receive approval from the regulatory body. Through the operational safety inspection by the regulatory body, observance of the operational safety program including operational limits and conditions by the operator is confirmed.
GS-R-1, 5.14	How does the regulatory body take into account in its inspection program the potential magnitude and nature of the hazard associated with the facility or activity?	When deciding the depth and extent of an inspection and the frequency of an inspection, the potential magnitude and nature of the hazard associated with the facility or activity are taken into account. Concerning the extent of the inspections, electric structures for business use that are particularly important to ensuring public safety are designated as “specific electric structures for business use” and subjected to inspection before use. Boilers for power generation, turbines, and like items that are particularly important to ensuring public safety are designated as “specific important electrical structures for business use” and subjected to periodic inspections. Concerning the frequency of inspections, the inspection before use is conducted before a facility is placed in service, the periodic inspections are conducted every 13 months, and the safety preservation inspections are conducted four times a year. In addition, an on-site inspection is conducted as necessary.

Article 15: Radiation Protection

	Question	Reply
NS-R-2, 8.1, and 8.2	What program has the operating organizations have established and implemented to ensure that, in all operational states, doses due to exposure to ionizing in the plant or due to any planned releases of radioactive material from the plant are kept below prescribed limits and as low as reasonably achievable?	Refer to 15.3 (1) and (2) of this report. As described in 19.3 of this report, the operator must specify items concerning radiation control in the operational safety program and receive approval from the regulatory body.
NS-R-2, 8.3	How does the operating organization ensure that the radiation protection function in its organization has sufficient independence and resources to enforce and advice on radiation protection regulations, standards and procedures, and safe working practices?	The operator must establish a dedicated organization in charge of radiation control to confirm that the exposure of radiation workers in reactor facilities complies with the legislative requirements. As described in 19.3 of this report, the operator must specify items concerning radiation control in the operational safety program and receive approval from the regulatory body. The observance of the operational safety program is confirmed through the operational safety inspection.
NS-R-2, 8.6	What are the requirements to ensure that all site personnel working in a controlled area or regularly employed in a supervised area have their occupational exposures assessed and what are the dose limits required by the regulation?	Refer to 15.2 and 15.3 of this report.
NS-R-1, 6.90	What kind of systems is provided to treat radioactive liquid and gaseous effluents in order to keep the quantities and concentrations of radioactive discharges controlled and within prescribed limits? How is the ALARA principle applied?	Refer to 15.2 (2), 2) of this report.
NS-R-2, 8.10	By what means has the operating organization demonstrated that the assessed radiological impacts and doses to the general public are kept as low as reasonably achievable?	As described in 15.1 of this report, the operator, pursuant to ALARA guideline, must establish the control target values. As described in 15.2 (2) of this report, the operator must control release to assure that the control target values are not exceeded. The amount of release is disclosed to the public.
NS-R-2, 8.11	How are the discharges of radioactive effluents monitored and controlled?	Refer to 15.2 (2) of this report. Released radioactivity of radioactive gaseous waste and radioactive liquid waste is monitored according to “Radioactive Materials Measurement Guidelines” and an environmental monitoring is implemented in the vicinity of a nuclear power station according to “Environmental Radiation Monitoring Guidelines”.
NS-R-2, 8.12	What programs have been established and implemented, if required by the regulatory body, for monitoring the environment in the vicinity of the plant in order to assess the radiological impacts of radioactive releases on the environment?	Refer to 15.2 (3) of this report.

Article 16: Emergency Preparedness

16.1

	Question	Reply
NS-R-2, 2.31 - 2.38	On-site emergency preparedness: These requirements are covered by the following quotations.	As described in 16.2 (1), 3) of this report, the nuclear operator has an obligation to establish license holder's plan for nuclear emergency preparedness and to submit the plan to the Minister of METI before commencing operation of a reactor.
SS 115, V.4, V.12, V.13, V.17, V.19	How do the emergency plans include, as appropriate intervention levels for relevant protective actions and the scope of their application, with account taken of the possible severity of accidents or emergencies that could occur?	As described in 16.2 of this report, in a case that a certain specific event occurs in the nuclear power station, the operator has an obligation to promptly notify the Minister of METI and the head of local government(s) of it. Then, the national government issues Declaration of Nuclear Emergency at a specific level of the event and initiates prescribed actions. The matters such as the reporting criteria of operator and the conditions, by which the national government issues Declaration of Nuclear Emergency are described in Table 16-1 of this report.
GS-R-2, 3.8	How does the regulatory body: • Ensure that appropriate emergency preparedness and response arrangement are established when nuclear fuel is brought to the site, and complete emergency preparedness is ensured before operation? (NS-R-2, 2.36)	• The response to a nuclear emergency of reactor equipment is described in the operational safety program.
	• Ensure that such emergency arrangements provide a reasonable assurance of an effective response?	• The operational safety program must be approved by the regulatory body prior to fuel loading.
	• Require that the emergency arrangements are tested in an exercise before the commencement of operation of a new nuclear installation, and thereafter: -At what intervals are exercises of the emergency arrangements held? - Which ones does the regulatory body witness? (NS-R-2, 2.37)	• The licensee must establish an emergency preparedness action plan and submit it before the commencement of operation of a new nuclear installation. However, it is not required that the emergency arrangements should be tested in an exercise before the commencement. -Pursuant to the emergency preparedness action plan, the licensee normally implements the exercise once a year. For details, see 16.3 of this report. -As described in 16.3 (1) and (3) of this report, the regulatory body participates in the exercises implemented by the local governments and the national government.
	• Require that emergency plans are periodically reviewed and updated? (SS115; V.3)	• Article 7 of the Special Law on Nuclear Emergency Preparedness stipulates that the plan for nuclear emergency preparedness be reviewed annually and amended, as necessary.
GS-R-2, 3.12	How do all organizations that may be involved in the response to an emergency ensure that management arrangements are adopted to meet the timescales for response throughout the emergency and for an effective and coordinated response?	As described in 16.2 and 16.3 of this report, related organizations prepare each own plan for emergency preparedness, ensure effectiveness of the preparedness through periodically implemented exercise and amend the plan, as necessary.
GS-R-2, 4.7	How is transition from normal to emergency operations defined and made?	As described in 16.2(3) of this report, the notifying criteria for events recognized as emergency are provided in Table 16.1 of this report.
GS-R-2, 4.12	When circumstances necessitate an emergency response, how do	The Special Law on Nuclear Emergency Preparedness defines the notifying criteria such as radiation dose, upon which the

	operators determine the appropriate emergency class (see paragraph 4.19) or the level of emergency response and initiate the appropriate on-site actions? How does the operator notify and provide updated information, as appropriate, to the off-site notification point?	operator must notify the regulatory body of the event, and the criteria, upon which the government announces the Declaration of Nuclear Emergency. The operator or the government takes pre-determined actions based on their plan for nuclear emergency preparedness. Further subdivided classifications for emergency situation are not specified. But, the events that may lead to nuclear emergency are defined for individual systems and components, and the operator transmits those plant information to an off-site center on line, as described in 16.2 (1), 1) of this report.
GS-R-2, 4.14	How is it ensured that appropriate emergency response actions are initiated promptly upon the receipt of a notification from another State or information from the IAEA of an actual or potential transnational emergency that could affect the State?	For a nuclear emergency occurred in a neighboring country, our country will respond to it within the framework established by the Convention on Early Notification of a Nuclear Accident, the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency and the Basic Law on Emergency Preparedness.
GS-R-2, 4.20	What criteria for emergency classification are used to predefine emergency action levels (EALs) for abnormal situations (see paragraph 4.70)? How does the classification system aid the initiation of a response to allow effective management and implementation of emergency operations?	The Special Law on Nuclear Emergency Preparedness defines the notifying criteria such as radiation dose, upon which the operator must notify the regulatory body of the event, and the criteria, upon which the government announces the Declaration of Nuclear Emergency. The operator or the government takes pre-determined actions based on their plan for nuclear emergency preparedness. Further subdivided classifications for emergency situation are not specified. But, the events that may lead to nuclear emergency are defined for individual systems and components, and implementation of emergency operation starts in a timely manner.
GS-R-2, 4.27	What arrangements have been made for response organizations to have sufficient personnel available to perform their assigned initial response actions?	Response organizations are obliged to take measures prevent the propagation of nuclear accidents.
GS-R-2. 4.28	What arrangements have been made to provide a response to a nuclear or radiological emergency for which detailed plans could not be formulated in advance?	An exercise where its scenario has not been provided for participants in advance is conducted, as necessary, so that in case of unexpected events they can handle them appropriately.
GS-R-2, 4.39	What arrangements are in place for actions by the operator to prevent an escalation of the threat, to return the nuclear installation to a safe and stable state, to reduce the potential for releases of radioactive material or exposures and to mitigate the consequences of any actual releases or exposures?	As described in 12.2 of this report, the operator must develop the operational procedures for accidents and required actions are specified in the procedures. And, training of operators is implemented according to the operational procedures.
GS-R-2, 4.48	What arrangements are in place for making and implementing decisions on urgent protective actions to be taken off the site?	Please refer to 16.2 of this report.
GS-R-2, 4.56	What arrangements have been made to protect emergency workers?	As described in Table 15-1 (Article 15) of this report, effective dose limits for emergency workers are established.
GS-R-2, 4.67	How is radiation monitoring and environmental sampling and assessment carried out in order to identify new hazards promptly and to refine the strategy for response?	Please refer to 16.2 of this report. Information such as measurements of environmental radiation monitors installed in the periphery of a nuclear facility are transmitted online to and indicated in an off-site center near the reactor facility.
GS-R-2, 4.68	How is information about emergency conditions, emergency assessments and the protective actions recommended and taken made	Please refer to 16.2 (1), 1) of this report.

	available to all relevant response organizations throughout the period of the emergency?	
GS-R-2, 4.71	What arrangements are there for promptly assessing any radioactive contamination, releases of radioactive material and doses for the purpose of deciding on or adapting the urgent protective actions to be taken following a release of radioactive material?	Please refer to 16.2 (1), 1) of this report.
GS-R-2, 4.80	What arrangements are there at the national level to treat people who have been exposed or contaminated?	As described in 16.1 of this report, preparation of a medical treatment system for nuclear emergency is stipulated, and related medical institution or emergency hospital is assigned for each regional block.
GS-R-2, 4.86	What arrangements are there to manage radioactive waste and contamination resulting from an accident?	The management of radioactive waste and contamination resulting from an accident is defined as a series of actions after a nuclear disaster according to the Special Law on Nuclear Emergency Preparedness. The head of Designated national administrative Agency or the head of Designated locally-stationed administrative Agency must implement series of actions pursuant to their plan for nuclear emergency preparedness or license holder's plan for nuclear emergency preparedness, based on the measurements of concentration or density of radioactive materials or radiation in the areas where emergency response are to be taken.
GS-R-2, 5.10	What arrangements are there for the coordination of emergency response and protocols for operational interfaces between license holders and local, regional and national governments, as applicable?	As described in 16.2 (1), 2) and 3) of this report, local governments and the operator have an obligation to prepare their own plan for nuclear emergency preparedness and the interface between their activities is explicitly specified in their plans.
GS-R-2, 5.29	What national emergency facility or facilities are designated for the coordination of response actions and public information?	An off-site center is established to implement coordination between related organizations (refer to Figure 16-2, Article 16 of this report).
GS-R-2, 5.33	What exercise programs are conducted on functions required for emergency response and organizational interfaces? How do these programs include the participation in some exercises of as many as possible of the organizations concerned? What is done to systematically evaluate the exercises and for some exercises to be evaluated by the regulatory body? How is the program updating in the light of experience gained?	Please refer to 16.3 of this report.

16.2

GS-R-2, 4.82, 4.54	What steps have been taken by the appropriate responsible authorities to provide the public with information throughout a nuclear or radiological emergency?	Such steps are specified in the Basic Plan for Emergency Preparedness, Chapter1, Section 2, Item 7, "Appropriate Transmission of Information to Residents in the Vicinity" and the Emergency Preparedness Guidelines 2-4, "Provision of Information to Residents in the Vicinity".
[GS-R-2, 3.5]	[What actions are taken by the national coordinating authority to foster the implementation of	Such actions are implemented in the framework for bilateral or multilateral cooperation in a timely manner.

	emergency arrangements by other States?]	
GS-R-2, 5.12	What arrangements have been made to ensure that all States within defined emergency zones are provided with appropriate information for developing their own preparedness to respond to an emergency and what arrangements have been made for appropriate transboundary coordination?	Please refer to 16.4 of this report.

16.3

GS-R-2, 3.15	How is any risk (threat) associated with nuclear installations in other States considered? In the risk assessment how are populations at risk identified and, to the extent practicable, the likelihood, nature and magnitude of the various radiation related risk considered?	For a nuclear emergency occurred in a neighboring country, our country will respond to it within the framework established by the Convention on Early Notification of a Nuclear Accident, the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency and the Basic Law on Emergency Preparedness.

Article 17: Siting

17(i)

	Question	Reply
NS-R-1, 5.18	In determining the design basis of a nuclear installations how are the various interactions between the installation and the environment considered, e.g. factors like population, meteorology, hydrology, geology, seismology and off site services (e.g. electricity supply)?	As described in 17.2 and 17.3 of this report, such interactions between the installations and environment are taken into consideration.
NS-R-3, 2.4, 2.14, and 2.15	How are all site characteristics that may affect the safety of the nuclear installation investigated and assessed, including natural phenomena and human induced situations and activities in the region of the proposed site?	As described in 17.2 and 17.3 of this report, the conditions to be considered in a siting process have been established.
NS-R-3, 2.5	How are the proposed sites for nuclear installations examined with regard to the frequency and severity of external natural and human induced events and phenomena that could affect the safety of the installation? How are the following external events, as applicable, evaluated? - Earthquakes, paragraphs 3.1-3.4 - Surface faulting, paragraphs 3.5-3.7 - Meteorological events, including extreme values, paragraphs 3.8-3.10 - Lightning, paragraph 3.11 - Tornadoes, paragraphs 3.12-3.14 - Tropical cyclones, paragraphs 3.15-3.17 - Floods due to precipitation and other causes, paragraphs 3.18-3.23 - Water waves induced by earthquakes or other geological phenomena, paragraphs 3.24-3.28 - Floods and waves caused by failure of water control structures, paragraphs 3.29-3.32 - Slope instability, paragraph 3.33 - Collapse, subsidence or uplift of the site surface, paragraph 3.35-3.37 - Soil liquefaction, paragraphs 3.38-3.40 - Behavior of foundation materials, paragraphs 3.41-3.43 - Aircraft crashes, paragraphs 3.44-3.46 - Chemical explosions, paragraphs, 3.48-3.49	As described in Article 18 of this report, detailed design review guide for earthquake and earthquake induced events has been established by the Nuclear Safety Commission (3.1-3.7, 3.24-3.28). The guide was revised in 2006, as described in 18.5 of this report. External events other than earthquake are required to be examined pursuant to the safety design guides of the Nuclear Safety Commission and are subjected to appropriate review and assessment process.
NS-R-3, 2.21, 2.17, and 3.52	What kind of data are use to characterize the site? How are the necessary data collected (site specific, data from other regions that are sufficiently relevant to the region of interest, prehistorically and historical data, simulation techniques, instrumentally recorded information) and how are they analyzed for reliability, accuracy and completeness?	For an earthquake, as shown in the Article 18, they are specified in the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities which was revised in 2006. The reliability, accuracy and completeness of the safety design guides and the Regulatory Guide for Reviewing Seismic Design represent the current highest level through the discussion and deliberation by top level of experts in Japan.
NS-R-3, 3.51	How is the region (including all facilities within the site boundary) investigated for installations in which materials are stored, processed, transported and otherwise dealt with that, if released under normal or accident conditions, could jeopardize the safety of the facility?	Potential effects by the events such as chemical explosion in a neighboring facility are evaluated at the stage of license review and are confirmed to be negligible or insignificant for safety.
NS-R-3,	How has the ambient radioactivity in the region	The pre-operational survey of the ambient

4.15	assessed before commissioning of the nuclear installation so as to be able to be determine the effects of the installation and hence provide a baseline in future investigations?	radioactivity for comparison with the data after commissioning of a nuclear power reactor facility is not implemented.

17(ii)

NS-R-3, 2.12, and 2.22	How is for each proposed site the potential radiological impacts in operational states and in accident conditions on people in the region, including impacts that could lead to emergency measures, evaluated with due consideration of the relevant factors, including population distribution, dietary habits, use of land and water, and the radiological impacts of any other releases of radioactive material in the region?	Please refer to 17.2 and 14.1 of this report.
NS-R-3, 2.26	In order to prepare for emergency planning, how is the proposed site and region studied to evaluate the present and foreseeable future characteristics and the distribution of the population of the region? How are the present and future uses of land and water in the region evaluated that may affect the potential consequences of radioactive releases for individuals and the population as a whole?	Please refer to 17.2 and 14.1 of this report.
NS-R-3, 2.27	How is it ensured that: (a) For operational states of the facility the radiological exposure of the population remains as low as reasonably achievable, economic and social factors being taken into account? (b) The radiological risk to the population associated with accident conditions is acceptably low?	Please refer to 17.2 and 14.1 of this report.

17(iii)

NS-R-3, 5.1, and 2.4	How and to what extent are the characteristics of the natural and human induced hazards as well as the demographic, meteorological and hydrological conditions of relevance to the nuclear installation observed and monitored throughout the lifetime of the nuclear installation?	Although the natural and human induced hazards as well as the demographic, meteorological and hydrological conditions are reviewed when constructing a nuclear installation, they are not monitored and their impacts are not evaluated throughout the lifetime of the plant after its construction.
NS-R-2, 10.3	How are site characteristics and corresponding external events taken into account in a Periodic Safety Review to determine to what extent the existing safety analysis report remains valid?	In the current periodic safety review process, we do not re-evaluated whether the conditions at the time of siting assessment remain valid or not.

17(iv)

NS-R-2, 5.12	What arrangements are in place or planned to ensure that all States within defined emergency zones are provided with appropriate information for developing their own preparedness to respond to an emergency? What arrangements are in place for appropriate transboundary co-ordination (participation in the licensing procedure and in environmental impact assessment)?	As described in 16.4 of this report, arrangements for information exchange with neighboring States have been established.

NS-R-1, 4.11	The safety of facilities and activities is of international concern. Several international conventions relating to various aspects of safety are in force. What kind of arrangements have been established by your national authorities, with the assistance of the regulatory body, as appropriate, for the exchange of safety related information, bilaterally or regionally, with neighboring States and other interested States, and with relevant intergovernmental organizations, both to fulfill safety obligations and to promote co-operation.	Please refer to the Introduction of this report.

Article 18: Design and Construction

18(i)

	Question	Reply
NS-R-1, 2.10	How are the five levels of the defense in depth have been taken into account in the design and operations of the nuclear installations of a plant to: (1) Prevent deviations from normal operation, and to prevent system failures? (2) Detect and intercept deviations from normal operational states in order to prevent occurrences from escalating to accident conditions? (3) Control design basis accidents to reach safe shut down of the plant? (4) Address severe accidents where design basis may be acceded and insure that radioactive releases are kept as low as practicable? (5) Mitigate the radiological consequences of potential releases of radioactive materials that may result from accident conditions?	For the five levels of the defense in depth, please refer to the following descriptions; Levels One to Three: 18.3 of this report; Level Four: 18.4 of this report; and Level Five: Article 16 of this report.
NS-R-1, 5.1, and 5.3	How are structures, systems and components, including software for instrumentation and control, important to safety identified and classified on the basis of their function and significance to safety? How is it ensured they are designed, constructed and maintained so that their quality and reliability is commensurate with this classification? How is it ensured that any failure in a systems classified in a lower class will not propagate into a system classified in a higher class?	The policy for the Regulatory Guide for Reviewing Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities is described in 18.3 of this report. A requirement that “any failure in a systems classified in a lower class will not affect a system classified in a higher class” is clarified in the item IV. 4 of the regulatory guide mentioned above.
NS-R-1, 5.8 - 5.20	What internal and external events and combination of events are been considered in the design of the nuclear installations?	The events to be considered as subjects for a safety assessment are described in 14.1 (2), 2) of this report. Namely, by analyzing the failures of equipment or systems or any operational error of them, the events causing most severe consequence among a group of events with similar progression are selected. Then, these postulated events are classified into “abnormal transients during operation” and “accident” as shown in the Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities, depending on their possibility of occurrence and their potential severity of impact if they occur. Finally, these classified events are evaluated for their significance to safety according to the criteria established for each classification. a. “Abnormal transient during operation” is an event leading to abnormal situation, which may be caused by a single equipment failure or an erroneous action or a single operational error by operating personnel, or which may be caused by an external disturbance, anticipated to occur during the lifetime of a commercial power reactor. As examples of such events, 14 events and 12 events have been selected for a pressurized water reactor (PWR) and a boiling water reactor (BWR),

		respectively. In a safety analysis for the events, the integrity of a core and a reactor coolant pressure boundary is verified according to the criteria specified in the Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities, so that the adequacy of the safety design of important safety related equipment such as a safety protection system and a reactor shut-down system is verified. b. "Accident" is an abnormal condition exceeding "abnormal transient during operation". Although its frequency of occurrence may be extremely low, its occurrence is intentionally postulated from the viewpoint of evaluating the release of radioactive materials from a commercial power reactor. As examples of such accidents, 10 events and 9 events have been selected for a PWR and a BWR, respectively.
NS-R-1, 5.33, and 5.34	To what extent are single failures and common cause failures prevented in the designs of the nuclear installations?	To prevent a single failure and a common cause failure, the Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities, Guideline 9, "Design Consideration for Reliability" stipulates that a system performing especially important safety function must incorporate features such as redundancy, diversity and independence, taking its structure, operation principle and the nature of its safety function into account.
NS-R-1, 5.31	What severe accident vulnerability studies have been performed and what measures have been implemented as the result of the studies.	As described in 18.6 of this report, the measures for AM have been developed.

18(ii)

NS-R-1, 3.6	How is it ensured that, wherever possible, structures, systems and components important to safety: • Are designed according to the latest or currently applicable approved standards? • Are of a design proven in previous equivalent applications? • Are selected to be consistent with the plant reliability goals necessary for safety? • Where codes and standards are used as design rules, How are they • Identified and evaluated to determine their applicability, adequacy and sufficiency? • Supplemented or modified as necessary to ensure that the final quality is commensurate with the necessary safety function?	As described in 18.7 of this report, the measures to assure technical reliability through the feedback of operational experiences, tests and analyses have been taken.
NS-R-1, 3.7	Where an unproven design or feature is introduced or there is a departure from an established engineering practice, how is safety demonstrated to be adequate? How is the development: • Tested before being brought into service? • Monitored in service, to verify that the expected behavior is achieved?	As described in 18.7 of this report.

NS-R-1, 3.9	How does the design take account of relevant operational experience that has been gained in operating plants and the results of relevant research programs?	As described in 18.7 of this report.

18(iii)

NS-R-1, 4.8, 5.5, and 5.40	How is it ensured that the design allows for reliable, stable easily manageable?	Please refer to 18.7 of this report.
NS-R-1, 3.3	How does the design management ensure that the requirements of the operating organization are met and that due account is taken of the human capabilities and limitations of personnel? How does the design organization supply adequate safety design information to ensure safe operation and maintenance of the plant and to allow subsequent plant modifications to be made, and recommended practices for incorporation into the plant administrative and operational procedures (i.e. operational limits and conditions)?	The operator, according to the quality assurance program established to comply with the national requirements, must clarify and enssure the resources including human resources required for nuclear safety. Furthermore, the operator must implement management of the competence of personnel engaged in duties that may influence nuclear safety. Design parameters required for operation are handed over from the design organization to the operating organization, and the operating organization reflects the information in "Operational Limits", etc. in the operational safety program. Also the operational limits in the operational safety program may be changed following a plant modification. The modified operational safety program is required to be approved by the regulatory body.
NS-R-1, 5.50	How is it ensured that human behavior and the human-machine interface systematically is taken into account early in the design process?	As described in Article 12 of this report, when designing a main control room human factors are taken into consideration, and human factors and human error prevention are taken into consideration in the operation management.
NS-R-1, 5.51	How is it ensured the design provide operators with comprehensive but easy manageable information, compatible with the decision and action times?	Guideline 41 of the Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities requires that a main control room should be designed so that; the main control room allows operators 1) to monitor the operational conditions of a reactor and its associated main facilities as well as the principal parameters, and 2) to carry out rapid manual actions if they are required.
NS-R-1, 5.56	How is it ensured that the need for operators to intervene on a short-time scale is kept to a minimum?	The Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities, Main Text, Explanation II, "Safety Design Assessment", 4. "On the Items to be considered in Analysis" (3) stipulates followings; A system and equipment performing safety function should be generally designed so that it can perform its required function immediately after the occurrence of an abnormal condition without any operator's action. In a case that any operator's action will be expected, operators must be provided with sufficient time and appropriate information to accurately judge the situation and to carry out actions with high reliability. At least 10 minutes are allowed for operators to carry out any required action after they gain appropriate information to make accurate judgment.

Article 19: Operation

19(i)

	Question	Reply
NS.R2, 4.6, and NS-R-1, 3.10	How does the operating organization ensure that the commissioning program reflects the appropriate safety analysis (see Art 14) and includes all the tests necessary to demonstrate that the plant as installed: • Meets the design intent, and • Can be operated in accordance with the operational limits and conditions? How is it ensured that no tests are performed which could put the plant into conditions that have not been analyzed? How is it ensured that 'baseline' data on systems and components, which are important for the safety of the plant and for subsequent safety reviews, are collected and retained?	The commissioning test corresponds to a pre-service inspection in our terminology. In the pre-service inspection, testing operation are carried out by the operator. The test procedures, their items and contents, are confirmed to be consistent with the establishment license and the approved construction plan. Fuels can be loaded only after the operational safety program, with operational limits and conditions, has been approved by the regulatory body. The data obtained in commissioning process are preserved to be available in subsequent safety review as baseline data.
NS.R.2, 4.11	How is it ensured that reactor criticality and initial power rising is not authorized until all tests deemed necessary by the operating organization and the regulatory body have been performed and results acceptable to both parties have been obtained?	Pre-service inspection is divided into two stages, that is, a stage for performance tests of individual equipment and systems and a stage of power testing. The criticality testing and the power testing can be carried out only after all the testing data required for them have been acquired at the former stage.

19(ii)

NS.R.2, 5.1	How are operational limits and conditions developed to ensure that the plant is operated in accordance with the final design assumptions and intent? How do they cover actions to be taken and limitations to be observed by the operating personnel?	The operational limits and conditions are specified in an operational safety program, which is required to be approved by the regulatory body. Their approval can be granted if the operational safety program are consistent with the establishment license and the approved construction plan. Therefore, the plant can be operated in accordance with the final design assumptions and intent.
NS.R.2, 5.2	How are operating personnel directly responsible for the conduct of operation made familiar with the intent and content of the operational limits and conditions?	The operational limits and conditions are specified in the operational safety program, and the operating personnel can be made familiar with the operational safety program through a process such as education and training course in an operator training center.
NS.R.2, 5.5	How does the operating organization ensure that an appropriate surveillance program is implemented to ensure compliance with the operational limits and conditions, and that its results are evaluated and retained?	An operational safety program includes the surveillance program, which specifies the surveillance frequency to confirm the observance of the operational limits and conditions, and the time given to restore normal operation when a deviation was identified. The operating organization implements the surveillance process according to this program.

19(iii)

NS.R.2, 5.11	How are operating procedures developed and implemented for normal, abnormal	As describe in 12.2 (2), 2) of this report, operating procedures are developed for the conditions of

	and emergency conditions, in accordance with the policy of the operating organization and the requirements of the regulatory body?	normal operation, accidents or failures and emergency.
NS-R-2, 5.12	Describe which kind of procedures are in place for normal operation, abnormal conditions, design basis accidents and severe accidents?	As describe in 12.2 (2), 2) of this report, dedicated procedures and an accident management guideline are developed against severe accidents.
NS-R-2, 5.10	Describe the administrative procedure for the development, elaboration, validation, acceptance, modification and withdrawal of operating instructions and procedures.	Operating instructions and procedures are developed and controlled under the responsibility of the operator without any regulatory involvement. The regulatory body confirms some aspects of the operating procedures which are described in the operational safety program.
NS.R.2, 5.14	How is it ensured that operating personnel are knowledgeable of, and have control over, the status of plant systems and equipment for all operational states?	Operating personnel are educated and trained according to the provisions described in the operational safety program. Through the education and training, operating personnel are made familiar with systems and equipment of a plant.
	How is it ensured that only designated and suitably qualified members of the operating personnel control or supervise any changes in the operational states of the plant?	The operator establishes a qualification system for operating personnel, and personnel without any qualification can not engage in the operation of plant facilities.
NS.R.2, 5.18	How is it ensured that a non-routine operation, test or experiment, is the subject of a safety review and specific operational limits and conditions and a special procedure?	The operational limits and conditions are specified in an operational safety program. Any alteration of operational limits and conditions, with corrective actions from deviation from the limits, requires regulatory approval.
	If, during the non-routine operation, any of the specific operational limits or conditions is violated, how is it known what corrective action is to be taken?	Operational Safety Program states that special safety measures should be taken for non-routine operation.
NS.R.2, 6.1, and 6.6	What are the operating organization's program and types of procedures for maintenance, testing, surveillance and inspection of those structures, systems and components that are important to safety? How often is it re-evaluated in the light of experience?	Work instructions, management procedures, maintenance manuals and maintenance guides are developed for individual equipment and systems and they are periodically re-evaluated by the operator.

19(iv)

NS.R.2, 5.11	How are operating procedures developed and implemented for abnormal and emergency conditions, in accordance with the policy of the operating organization and the requirements of the regulatory body?	As describe in 12.2 (2), 2) of this report, operating procedures are developed for the conditions of normal operation, accidents or failures and emergency.
NS-R-2, 5.12	Describe which kind of procedures are in place for abnormal conditions, design basis accidents and severe accidents?	As describe in 12.2 (2), 2) of this report, dedicated procedures and an accident management guideline are developed against severe accidents.

NS.R.2, 5.8	How is it ensured that after an abnormal event, the plant is brought into a safe operational state and the appropriate remedial actions taken immediately? How is it ensured the operating organization undertakes a review and evaluation of the case and notifies the regulatory body?	Please refer to 19.4 and 19.6 of this report (Response in Accidents and Report of Events).

19(v)

NS.R.2, 2.10	How is it ensured that all activities that may affect safety are performed by suitably qualified and experienced persons, including activities performed by contractors?	The operator, according to the quality assurance program established to comply with the national requirements, must assess and select a supplier in a procurement process. The operator must implement actions such as inspections to ensure that requirements are satisfied.
NS-R-2, 2.4 (5),(6)	What provisions did the license holder take to establish liaison with organizations for design, construction, manufacturing and plant operation and with other organizations (national and international) as necessary to ensure the proper transfer of information, expertise and experience and the ability to respond to safety issues? What adequate resources, services and facilities are provided?	The operator, according to the quality assurance program established to comply with the national requirements, must ensure that appropriate in-house communication and information exchange on the effectiveness of a quality management system be maintained. Also, the operator must specify requirements for procurement and ensure requirements to a supplier through the procurement process.

19(vi)

NS-R-2, 2.17	What procedures are in place for reporting abnormal events to the regulatory body in accordance with established criteria?	As described in 19.6 and Table 19-2 of this report, reports on accidents and failures are stipulated by legislation and regulations in detail.

19(vii)

NS.R.2, 2.21	How are abnormal events with safety implications investigated? How is the outcome of such investigations converted into recommendations to the plant management and corrective action? How is information from such evaluations and investigations fed back to the plant personnel?	Please refer to 19.7 of this report, which describes processes for recurrence prevention and the feed back of lessons learned from the experiences of accidents and failures.
NS.R.2, 2.22	How does the operating organization use operating experience at other plants to derive lessons for its own operations?	Please refer to 19.7 of this report, which describes processes for recurrence prevention and the feed back of lessons learned from the experiences of accidents and failures.
NS.R.2, 2.23	How is operating experience examined for any precursors of conditions adverse to safety, so that any necessary corrective action can be taken before serious conditions arise?	The accident sequences leading to the postulated major accident are analyzed and their probabilities are calculated by PSA. A premise of consideration is that a corrective action should be taken at the early stage of a small event leading to a significant accident. In a case an event actually occurs, it will be examined whether postulated accident sequence was correct or not, and the sequence will be modified, as necessary.
NS.R.2, 2.25, and 2.4(5)	What mechanisms are used to share important experience with other national and international organizations?	Please refer to 19.7 of this report.

NS-R-2, 2.26	How is data on operating experience collected and retained for use as input for the management of plant ageing, for the evaluation of residual plant life, and for probabilistic safety assessment and periodic safety review?	The operator, according to the quality assurance program established to comply with the national requirements, must implement ageing control measures and periodic safety review.

19(viii)

NS-R-2, 8.8	How is the generation of radioactive waste kept to the minimum practicable by operating practices?	For radioactive waste management, please refer to the Japanese National Report for the Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management.
NS-R-2, 8.9	What programs are in place to manage radioactive waste at the site safely, also taking into consideration conditioning and final disposal?	For radioactive waste management, please refer to the Japanese National Report for the Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management.
NS-R-2, 5.21 - 5.23	How is spent fuel managed at the nuclear installation?	For radioactive waste management, please refer to the Japanese National Report for the Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management.

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