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

(As of the end of March, 2007)
### Table 11-2 Maintenance and Repair Training Centers of Licensees

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

*(As of the end of March, 2007)*
Fig. 11-1 Training on Nuclear-Safety Regulation

<table>
<thead>
<tr>
<th>Cross-cutting training</th>
</tr>
</thead>
<tbody>
<tr>
<td>Training on nuclear safety regulation</td>
</tr>
<tr>
<td>Nuclear emergency preparedness, Crisis management</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Meister</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nuclear power generation (BWR, PWR) for experts</td>
</tr>
<tr>
<td>- Nuclear power generation (FBR) for experts</td>
</tr>
<tr>
<td>- FBR sodium technical training</td>
</tr>
<tr>
<td>- Special training course on QA of nuclear installation</td>
</tr>
<tr>
<td>- Risk communication training for managers</td>
</tr>
<tr>
<td>- Public-relations training for Nuclear Safety Inspectors</td>
</tr>
<tr>
<td>- Quality Assurance training</td>
</tr>
<tr>
<td>- Nuclear emergency preparedness, Advanced</td>
</tr>
<tr>
<td>- Nuclear emergency preparedness, on-site training</td>
</tr>
<tr>
<td>- Off-site center desk-top drill</td>
</tr>
<tr>
<td>- Emergency preparedness and response</td>
</tr>
<tr>
<td>- Off-site center management</td>
</tr>
<tr>
<td>- Off-site center functional group</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Senior expert</th>
</tr>
</thead>
<tbody>
<tr>
<td>- Nuclear Safety Inspector basic training</td>
</tr>
<tr>
<td>- Electric Structure Inspector (nuclear power) training</td>
</tr>
<tr>
<td>- Nuclear Facility Inspector basic training</td>
</tr>
<tr>
<td>- Nuclear power station risk assessment technology</td>
</tr>
<tr>
<td>- Nuclear reactor safety design, basic</td>
</tr>
<tr>
<td>- Overseas training</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Expert</th>
</tr>
</thead>
<tbody>
<tr>
<td>- Basic Safety Regulation</td>
</tr>
<tr>
<td>- Participation to the various basic lectures by the Japan Atomic Energy Agency</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Entry</th>
</tr>
</thead>
<tbody>
<tr>
<td>- Radiation safety</td>
</tr>
</tbody>
</table>
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 misoperation.

3) The Nuclear and Industrial Safety Agency (NISA) and the Japan Nuclear Energy Safety Organization (JNES) clarified the requirements for prevention of misoperation, 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

<table>
<thead>
<tr>
<th>Item</th>
<th>Requirements</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Environmental conditions of a main control room</td>
<td>Main control room should be in comfortable environmental conditions taking into consideration temperature, lighting and noise so that operators can operate appropriately.</td>
</tr>
</tbody>
</table>
| 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.

![Operator Console (ABWR)](image1)

![Fig. 12-1 Main Control Panel of ABWR](image2)

![Fig. 12-2 Main Control Panel of Latest PWR (proto type)](image3)

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

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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 viewpoint 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 viewpoint 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
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 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 $1 \times 10^{-6}$ 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 $1 \times 10^{-5}$ 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, 2007 respectively. 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

<table>
<thead>
<tr>
<th>Time of Inspection</th>
<th>Contents of Inspection</th>
</tr>
</thead>
<tbody>
<tr>
<td>(1) At the time of installation of each structure and component</td>
<td>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.</td>
</tr>
<tr>
<td>(2) At the time of installation of steam turbine and auxiliary boilers</td>
<td>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.</td>
</tr>
<tr>
<td>(3) At the time of fuel loading</td>
<td>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.</td>
</tr>
<tr>
<td>(4) At the time of criticality</td>
<td>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.</td>
</tr>
<tr>
<td>(5) At the time of completion of construction</td>
<td>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.</td>
</tr>
</tbody>
</table>
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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 ICRP 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>
<thead>
<tr>
<th>Item</th>
<th>Limit</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Effective dose limits</td>
<td></td>
</tr>
<tr>
<td>a) Radiation workers</td>
<td>100 mSv / 5 years, but do not exceed 50 mSv for any year</td>
</tr>
<tr>
<td>b) Female</td>
<td>100 mSv / 5 years, but do not exceed 5 mSv for any 3 months</td>
</tr>
<tr>
<td>c) Pregnant female</td>
<td>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</td>
</tr>
<tr>
<td>2. Equivalent dose limits</td>
<td></td>
</tr>
<tr>
<td>a) Eye lens</td>
<td>150 mSv/ year</td>
</tr>
<tr>
<td>b) Skin</td>
<td>500 mSv/ year</td>
</tr>
<tr>
<td>c) Female abdominal region</td>
<td>2 mSv from notification of pregnancy to delivery</td>
</tr>
<tr>
<td>3. Dose limits in emergency</td>
<td></td>
</tr>
<tr>
<td>a) Effective dose</td>
<td>100 mSv/ incident</td>
</tr>
<tr>
<td>b) Equivalent dose for eye lens</td>
<td>300 mSv/ incident</td>
</tr>
<tr>
<td>c) Equivalent dose for skin</td>
<td>1 Sv/ incident</td>
</tr>
</tbody>
</table>
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 increase from 216 of the previous year.

<table>
<thead>
<tr>
<th>Fiscal year</th>
<th>2000</th>
<th>2001</th>
<th>2002</th>
<th>2003</th>
<th>2004</th>
<th>2005</th>
<th>2006</th>
</tr>
</thead>
<tbody>
<tr>
<td>Collective Dose (man-Sv)</td>
<td>78.83</td>
<td>78.05</td>
<td>84.03</td>
<td>96.41</td>
<td>77.86</td>
<td>66.91</td>
<td>67.43</td>
</tr>
<tr>
<td>Average annual individual dose (mSv)</td>
<td>1.2</td>
<td>1.2</td>
<td>1.3</td>
<td>1.4</td>
<td>1.2</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Total number of workers</td>
<td>65,900</td>
<td>67,800</td>
<td>63,800</td>
<td>66,600</td>
<td>66,700</td>
<td>66,300</td>
<td>66,900</td>
</tr>
</tbody>
</table>
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>
<thead>
<tr>
<th>Fiscal year</th>
<th>2000</th>
<th>2001</th>
<th>2002</th>
<th>2003</th>
<th>2004</th>
<th>2005</th>
<th>2006</th>
</tr>
</thead>
<tbody>
<tr>
<td>BWR (man-Sv)</td>
<td>1.96</td>
<td>1.68</td>
<td>2.10</td>
<td>2.40</td>
<td>1.58</td>
<td>1.39</td>
<td>1.33</td>
</tr>
<tr>
<td>PWR (man-Sv)</td>
<td>1.03</td>
<td>1.27</td>
<td>1.00</td>
<td>1.07</td>
<td>1.25</td>
<td>0.97</td>
<td>1.08</td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>Items</th>
<th>Limit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dose limits outside the peripheral monitoring area</td>
<td></td>
</tr>
<tr>
<td>Effective dose</td>
<td>1 mSv/year</td>
</tr>
<tr>
<td>Equivalent dose for eye lens</td>
<td>15 mSv/year</td>
</tr>
<tr>
<td>Equivalent dose for skin</td>
<td>50 mSv/year</td>
</tr>
</tbody>
</table>

(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

<table>
<thead>
<tr>
<th>Year</th>
<th>Station</th>
<th>1997</th>
<th>1998</th>
<th>1999</th>
<th>2000</th>
<th>2001</th>
<th>2002</th>
<th>2003</th>
<th>2004</th>
<th>2005</th>
<th>2006</th>
<th>Numerical Discharge Control Guides</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Station - A</td>
<td>N.D.*</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>6.7 x 10^{15}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Station - B</td>
<td>4.3 x 10^{15}</td>
<td>6.1 x 10^{15}</td>
<td>1.2 x 10^{13}</td>
<td>5.7 x 10^{13}</td>
<td>1.5 x 10^{15}</td>
<td>2.8 x 10^{15}</td>
<td>1.8 x 10^{15}</td>
<td>4.1 x 10^{15}</td>
<td>6.2 x 10^9</td>
<td>2.9 x 10^9</td>
<td>3.7 x 10^{13}</td>
</tr>
</tbody>
</table>

* N.D. indicates a value below the detection limit concentration of 2 x 10^{-2} Bq/cm³.

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

<table>
<thead>
<tr>
<th>Year</th>
<th>Station</th>
<th>1997</th>
<th>1998</th>
<th>1999</th>
<th>2000</th>
<th>2001</th>
<th>2002</th>
<th>2003</th>
<th>2004</th>
<th>2005</th>
<th>2006</th>
<th>Numerical Discharge Control Guides</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Station - A</td>
<td>N.D.*</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>2.3 x 10^{11}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Station - B</td>
<td>8.6 x 10^5</td>
<td>1.2 x 10^5</td>
<td>1.6 x 10^5</td>
<td>1.1 x 10^5</td>
<td>2.7 x 10^5</td>
<td>N.D</td>
<td>N.D</td>
<td>1.9 x 10^9</td>
<td>N.D</td>
<td>N.D</td>
<td>1.0 x 10^{11}</td>
</tr>
</tbody>
</table>

*: N.D. indicates a value below the detection limit concentration of 7 x 10^{-9} Bq/cm³.

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

<table>
<thead>
<tr>
<th>Year</th>
<th>Station</th>
<th>1997</th>
<th>1998</th>
<th>1999</th>
<th>2000</th>
<th>2001</th>
<th>2002</th>
<th>2003</th>
<th>2004</th>
<th>2005</th>
<th>2006</th>
<th>Numerical Discharge Control Guides</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Station - A</td>
<td>N.D.*</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>2.5 x 10^{11}</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Station - B</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>N.D.</td>
<td>1.4 x 10^{11}</td>
<td></td>
</tr>
</tbody>
</table>

*: N.D. indicates a value below the detection limit concentration of 2 x 10^{-2} Bq/cm³.

(Represented by ⁶⁰Co)

(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 serious 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

<table>
<thead>
<tr>
<th>Events that licensees should report.</th>
<th>Criteria for reporting by licensees and “Nuclear Emergency” declaration by national government</th>
</tr>
</thead>
<tbody>
<tr>
<td>a) Dose of radiation near the site boundary detected dose</td>
<td>5 micro Sv/h at one point for more than 10 min</td>
</tr>
<tr>
<td>b) Detection of the radioactive materials in usual release points, such as exhaust pipes</td>
<td>5 micro Sv/h at more than 2 points at the same time</td>
</tr>
<tr>
<td>c) Radiation by fire, explosion, etc. or detection of radioactive materials (outside the control zone)</td>
<td>Radioactive materials worth more than 5 micro Sv/h</td>
</tr>
<tr>
<td>d) Individual events of each nuclear installation (Example for reactor)</td>
<td>Radiation dose of more than 50 micro Sv/h</td>
</tr>
<tr>
<td>- Failure of scram</td>
<td>Release of radioactive materials worth more than 5 micro Sv/h</td>
</tr>
<tr>
<td>- Loss of reactor coolant</td>
<td>When the nuclear reactor shutdown cannot be performed by usual neutron absorbers</td>
</tr>
<tr>
<td>- Loss of all AC power supplies</td>
<td>Occurrence of leakage of nuclear reactor coolant which needs operation of the emergency core coolant system (ECCS)</td>
</tr>
<tr>
<td></td>
<td>When all of reactor shutdown functions are lost</td>
</tr>
<tr>
<td></td>
<td>When water cannot be poured to the nuclear reactor by all ECCSs</td>
</tr>
<tr>
<td></td>
<td>When all measures for the cooling reactor core are lost with loss of all AC power supplies.</td>
</tr>
</tbody>
</table>

**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

<table>
<thead>
<tr>
<th>Conductor</th>
<th>Date</th>
<th>Nuclear Power Station</th>
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<tbody>
<tr>
<td>(1) Drills conducted by the National Government (April 2004 - March 2007)</td>
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<tr>
<td></td>
<td>–11/10/2005 (Thu.)</td>
<td></td>
</tr>
<tr>
<td>METI</td>
<td>10/25/2006 (Wed.)</td>
<td>Ikata Power Station (Shikoku Electric Power Co., Inc.)</td>
</tr>
<tr>
<td></td>
<td>–10/26/2006 (Thu.)</td>
<td></td>
</tr>
<tr>
<td>(2) Drills conducted by Local Governments (April 2004 - March 2007)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Shizuoka Pref.</td>
<td>06/29/2004 (Tue.)</td>
<td>Hamaoka Nuclear Power Station (Chubu Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Shimane Pref.</td>
<td>10/08/2004 (Fri.)</td>
<td>Shimane Nuclear Power Station (The Chugoku Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Miyagi Pref.</td>
<td>10/19/2004 (Tue.)</td>
<td>Onagawa Nuclear Power Station (Tohoku Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Hokkaido</td>
<td>10/22/2004 (Fri.)</td>
<td>Tomari Power Station (Hokkaido Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Ehime Pref.</td>
<td>10/26/2004 (Tue.)</td>
<td>Ikata Power Station (Shikoku Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Aomori Pref.</td>
<td>11/16/2004 (Tue.)</td>
<td>Higashidori Nuclear Power Station (Tohoku Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Saga Pref.</td>
<td>11/22/2004 (Mon.)</td>
<td>Genkai Nuclear Power Station (Kyushu Electric Power Co., Inc.)</td>
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<td>Nagasaki Pref.</td>
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</tr>
<tr>
<td>Fukushima Pref.</td>
<td>11/24/2004 (Wed.)</td>
<td>Fukushima Daini Nuclear Power Station (Tokyo Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Kagoshima Pref.</td>
<td>01/30/2005 (Sun.)</td>
<td>Sendai Nuclear Power Station (Kyushu Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Fukui Pref.</td>
<td>03/21/2005 (Mon.)</td>
<td>Takahama Power Station (The Kansai Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Kyoto Pref.</td>
<td></td>
<td></td>
</tr>
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<td>Ishikawa Pref.</td>
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<td>Shika Nuclear Power Station (Hokuriku Electric Power Co.)</td>
</tr>
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<td>Fukui Pref.</td>
<td>08/02/2005 ( Tue.)</td>
<td>Mihama Power Station (The Kansai Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Aomori Pref.</td>
<td>08/10/2005 (Tue.)</td>
<td>Higashidori Nuclear Power Station (Tohoku Electric Power Co., Inc.)</td>
</tr>
<tr>
<td>Hokkaido</td>
<td>10/21/2005 (Fri.)</td>
<td>Tomari Power Station (Hokkaido Electric Power Co., Inc.)</td>
</tr>
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<td>Ehime Pref.</td>
<td>10/21/2005 (Fri.)</td>
<td>Ikata Power Station (Shikoku Electric Power Co., Inc.)</td>
</tr>
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<td>Miyagi Pref.</td>
<td>10/28/2004 (Fri.)</td>
<td>Onagawa Nuclear Power Station (Tohoku Electric Power Co., Inc.)</td>
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<td>Niigata Pref.</td>
<td>11/09/2005 (Wed.)</td>
<td>Kashiwazaki Kariwa Nuclear Power Station (Tokyo Electric Power Co., Inc.)</td>
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<td></td>
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D. Safety of Installations
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.

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.
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


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
General Text:

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.


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 $S_2$ 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)

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<td>Guideline 30. Isolation function of reactor containment</td>
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<td>Guideline 39. Separation of safety protection system from instrumentation and control systems</td>
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<td>Guideline</td>
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<td>Control room</td>
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<tr>
<td>Classification</td>
<td>Definition</td>
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</tr>
</tbody>
</table>
| Class 1        | PS-1       | (i) Reactor coolant pressure boundary function  
                (ii) Excessive reactivity insertion prevention function  
                (iii) Core shape maintenance function |
|                | MS-1       | (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 |
|                | Class 2    | (i) Generation function of an actuation signal for the engineered safety features and to the reactor shutdown system  
                (ii) Specially important safety related functions |
|                | PS-2       | (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 |
|                |            | (i) Safety valve and relief valve re-closing function |

<table>
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<tr>
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<th>Definition</th>
<th>Function</th>
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<td><strong>MS-2</strong></td>
<td>(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.</td>
<td>(i) Fuel pool water supply function (ii) Function to prevent the discharge of radioactive materials</td>
</tr>
<tr>
<td></td>
<td>(ii) Structures, systems and component with an especially important function in the response of abnormal situations.</td>
<td>(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</td>
</tr>
<tr>
<td><strong>Class 3</strong></td>
<td><strong>PS-3</strong></td>
<td>(i) Structures, systems and components where initiating events of abnormal situations take place, and which are other than PS-1 and PS-2 components.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(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</td>
</tr>
<tr>
<td><strong>MS-3</strong></td>
<td>(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</td>
<td>(i) Function for mitigation of reactor pressure increase (ii) Function to control the power increases (iii) Reactor coolant make-up function</td>
</tr>
<tr>
<td></td>
<td>(ii) Structures, systems and components required for the response of abnormal situations</td>
<td>(i) Important for emergency response and function for recognizing abnormal situations</td>
</tr>
<tr>
<td>Plant type</td>
<td>BWR 2 &amp; 3</td>
<td>BWR 4</td>
</tr>
<tr>
<td>------------</td>
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</tr>
<tr>
<td>Containment type</td>
<td>MARK-I</td>
<td>MARK-I</td>
</tr>
<tr>
<td>Name of power station</td>
<td>Unit 1 of Tsuruga PS (BWR 2), Unit 1 of Fukushima Daiichi, (BWR 3)</td>
<td>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</td>
</tr>
<tr>
<td>Reactor shutdown system</td>
<td>SCRAM system</td>
<td>SCRAM system</td>
</tr>
<tr>
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<td>Stand-by Liquid Control System</td>
<td>Stand-by Liquid Control System</td>
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### Table 18-3 Establishment situation of prevention and mitigation system (BWR nuclear installation) (2/2)

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<th>BWR4</th>
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<th>ABWR</th>
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<tbody>
<tr>
<td>Containment type</td>
<td>MARK-I type</td>
<td>MARK-I type</td>
<td>Improved MK-I, MK-II and Improved MK-I</td>
<td>RCCV type</td>
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<tr>
<td></td>
<td>SHC</td>
<td>ADS</td>
<td>ADS</td>
<td>ADS</td>
</tr>
<tr>
<td>Divisions of system configuration</td>
<td>2 partitions</td>
<td>2 partitions</td>
<td>3 partitions</td>
<td>3 partitions</td>
</tr>
<tr>
<td>Number of D/G</td>
<td>2</td>
<td>2</td>
<td>3</td>
<td>3</td>
</tr>
</tbody>
</table>

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.  
RCIC: Reactor Core Isolation Cooling Sys.  
ADS: Automatic Depressurization Sys.  
LPCI: Low Pressure Coolant Injection Sys.  
HPCS: High Pressure Core Spray Sys.  
HPCF: High Pressure Core Flooder  
LPFL: Low Pressure Core Flooder
### Table 18-4 Establishment situation of prevention and mitigation systems (PWR nuclear installation) (1/2)

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<th>4 Loop</th>
<th>2 Loop</th>
<th>3 Loop</th>
<th>4 Loop</th>
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</thead>
<tbody>
<tr>
<td>Containment type</td>
<td>PCCV type</td>
<td>Dry type</td>
<td>Dry type</td>
<td>Ice condenser type</td>
</tr>
<tr>
<td>Name of power station</td>
<td>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</td>
<td>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</td>
<td>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</td>
<td>Unit 1 of Ohi PS, Unit 2 of Ohi PS</td>
</tr>
<tr>
<td>Reactor shutdown system</td>
<td>Scram system, Boric acid injection system</td>
<td>Scram system, Boric acid injection system</td>
<td>Scram system, Boric acid injection system</td>
<td>Scram system, Boric acid injection system</td>
</tr>
<tr>
<td>Containment shape</td>
<td>PCCV type</td>
<td>Dry type</td>
<td>Ice condenser type</td>
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<tr>
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<td>Freestanding steel type (with top dome)</td>
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<td></td>
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<tr>
<td>Plant Type</td>
<td>4 Loop</td>
<td>2 Loop</td>
<td>3 Loop</td>
<td>4 Loop</td>
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</tr>
<tr>
<td>Containment type</td>
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<td>Dry type</td>
<td>Dry type</td>
<td>Ice condenser type</td>
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<td><strong>ACC 4 units</strong></td>
<td><strong>ACC 2 units</strong></td>
<td><strong>ACC 3 units</strong></td>
<td><strong>ACC 4 units</strong></td>
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<td><strong>HPIS (/RHR)</strong></td>
<td><strong>HPIS (/RHR)</strong></td>
<td><strong>HPIS (/RHR)</strong></td>
<td><strong>HPIS (/RHR)</strong></td>
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<tr>
<td></td>
<td><strong>AFWS (motor driven)</strong></td>
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<td><strong>AFWS (turbine driven)</strong></td>
<td><strong>AFWS (turbine driven)</strong></td>
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<td>Divisions of system configuration</td>
<td><strong>2 systems</strong></td>
<td><strong>2 systems</strong></td>
<td><strong>2 systems</strong></td>
<td><strong>2 systems</strong></td>
</tr>
<tr>
<td>HPIS boosting unnecessary</td>
<td>HPIS boosting necessary</td>
<td>HPIS boosting necessary</td>
<td>HPIS boosting necessary</td>
<td></td>
</tr>
<tr>
<td>Number of D/Gs</td>
<td>2</td>
<td>2</td>
<td>2</td>
<td>2</td>
</tr>
</tbody>
</table>

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**

Operation (operational period (less than 13 months) + shutdown period)

<table>
<thead>
<tr>
<th>Regulatory authority</th>
<th>Observance status of safety preservation rule is inspected in accordance with Paragraph 5 of Article 37 of the Reactor Regulation Law etc. (Once in 4 months).</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inspection</td>
<td>Inspection performed periodically in accordance with Paragraph 1 of Article 54 of the Electricity Utilities Industry Law etc.</td>
</tr>
<tr>
<td>Audit of Licensee's Periodic Check System (JNES)</td>
<td>Document check and witnessing of implementing system of Periodic Licensee's Check in accordance with Paragraph 4 of Article 55 of the Electricity Utilities Industry Law etc.</td>
</tr>
<tr>
<td>Licensee</td>
<td>Licensees Control the Conformity to the Technical Standards for nuclear power generation equipment voluntarily in accordance with Paragraphs 1 and 3 of Article 55 of the Electricity Utilities Industry Law etc.</td>
</tr>
<tr>
<td>Review</td>
<td>Situation of reflection of the newest technical knowledge is evaluated by the licensee every ten years and a technical evaluation for aging is performed after 30 years operation in accordance with Paragraph 3 of Article 15 of the Regulations for Establishment, Operation etc. of Commercial Power Reactors</td>
</tr>
<tr>
<td>Man-power system</td>
<td>Resident Nuclear Safety Inspectors, witnessing inspections (NISA), On-site Inspection</td>
</tr>
<tr>
<td></td>
<td>Chief engineer of reactors</td>
</tr>
<tr>
<td></td>
<td>“Investigation on regulatory activities” (the NSC)</td>
</tr>
</tbody>
</table>

(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.

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