THE REPUBLIC OF CHINA
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FOR THE
CONVENTION ON NUCLEAR SAFETY

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EXECUTIVE SUMMARY

The Republic of China (hereafter referred to as the ROC or Taiwan) has not yet signed the Convention on Nuclear Safety of the International Atomic Energy Agency (IAEA) as adopted on June 17, 1994. However, the safety of the civil applications of the nuclear energy in this nation is always considered as the top priority. Thus, although being a non-contracting party, the ROC is willing to act as a contracting party to meet all the requirements addressed in the applicable articles established by this Convention. This ROC's national report shows how the obligations under the terms of the Convention on Nuclear Safety were met by this country.

In Taiwan, there are currently three operating nuclear power plants (NPPs), namely, Chinshan, Kuosheng and Maanshan, which are also named the first, second and third NPPs, respectively. In addition, there is another one nuclear power plant under construction, the Lungmen NPP. Each of these four NPPs is equipped with two identical nuclear power units. All the nuclear power plants in this country are owned and operated by a state-owned utility, the Taiwan Power Company (hereafter referred to as the TPC or Taipower).

The regulatory body for all the nuclear affairs in Taiwan is the Atomic Energy Council (hereafter referred to as the AEC), which is at the ministry level in the current governmental organization and reports directly to the Executive Yuan (i.e. the Cabinet) or the Premier. In this report, the specific improvements made to the regulatory requirements of the AEC were described. A lot of the important nuclear regulatory laws and acts have been strengthened and legislated or re-legislated in recent years, if necessary. The enforcement rules and regulations related to these laws and acts were strengthened in the meantime.

A compact reactor oversight process (ROP), similar to the ROP adopted by the United States Nuclear Regulatory Commission (USNRC), has been established and implemented as a part of actions for the AEC's Information Transparency Policy. The purpose of this compact ROP system is to establish a system for inspecting and assessing the plant performance to ensure the plant safe operation and, in the meantime, for an easily understood indicators of the safety situations of an operating nuclear power unit for the public.

The surveillance of the NPPs' operation and/or construction by the regulatory body is strictly implemented by inspections and document or report reviews. Various kinds of inspections are being performed including, for example, the residence inspection, periodic inspection, refueling outage inspection, expert team inspection, unannounced midnight inspection and whatever inspections when needed.

In the Chinshan NPP, the technical specifications (TS or tech. spec.) have been successfully transferred from the customer's (or conventional) tech. spec. to the improved technical specifications (ITS) in February 2002. With this successful experience, the technical specifications of the Kuosheng and Maanshan NPPs were also converted into the ITS in January 2008 and September 2004, respectively.

In case a nuclear accident should occur, the ROC's nuclear emergency response organizations, among which the National Nuclear Emergency Response Center (NNERC) is the leading agency, will immediately react to take the responsibility to protect the public and to mitigate the effects on the public. An on-site nuclear emergency response drill is required for each NPP to be conducted every year. In the meanwhile, a national nuclear emergency response exercise is conducted annually with one of the NPPs as the reference plant. Through the experiences obtained from these exercises, the contingency plans for the response are in place and will be continually updated and enhanced.

Since the last National Report for the Convention on Nuclear Safety of the ROC was issued in September 2004, a number of significant safety-improvement or nuclear-related progresses had been evolved in this nation. To name several of them, the following are some of the examples:

- The reactor oversight process (ROP) adopted by the AEC is now more comprehensive, which includes the use of a probabilistic risk assessment code PRiSE in the significance determination process for the inspection findings.

- The measurement uncertainty recapture (MUR) power uprate program was first conducted for the Kuosheng Unit 2 and completed in July 2007. Now all six operating nuclear units have implemented the MUR technique with a total power uprate of about 55.64 MWe.

- The license renewal application for the Chinshan NPP, of which the Units 1 and 2 have been operated since December 1978 and July 1979, respectively, was submitted by the TPC to the AEC in July 2009.

- In May 2006, the 'Act for Establishment of Low Level Radioactive Waste Final Disposal Facility' was enacted and became effective to stipulate the disposal site selection process.

- A protection of the nuclear power plant (NPP) from great earthquake by the use of the automatic seismic trip system (ASTS) had been formally on-line for all three operating NPPs in Taiwan since the end of November 2007.

- The enforcement rules for various nuclear regulatory laws as well as their associated regulations had been either established or amended to more strictly and safely regulate the nuclear-related activities ever since then.

A more detailed description of these progresses can be found in the following sections.

INTRODUCTION

The Republic of China (ROC) is not a contracting party to the Convention on Nuclear Safety of the International Atomic Energy Agency (IAEA). However, the safety of the nuclear applications in this nation is always considered as a top priority. Thus, any international activities, as long as they are helpful to the promotion of this nation's nuclear safety, Taiwan is willing to participate, if possible, and fulfill the obligations. This report is the updated ROC's National Report for the Convention on Nuclear Safety for peer review in the Bilateral Technical Meeting between the USA and the ROC in April 2011. The fulfillment of the obligations of the Convention by Taiwan was evaluated in this report. It demonstrates how the ROC meets the safety requirements of all the articles established by the Convention in its Chapter 2.

The indigenous energy resources are severely scarce in this country. In 2009, the total supply of the primary energy was about 138.06 x 10^6 kilo-liter oil equivalent, a decrease of 2.25% as compared to that in the previous year. Of this total supply, 99.37% is imported and the indigenous energy supply contributed only 0.63%. When classified by the kinds of energy sources, the nuclear power constituted about 8.7% of the total. Since 2001, coal was no longer produced domestically and the supply of it depended totally on imports. While percentagewise the installed hydropower capacity is not so small, the electricity actually produced from the hydropower contributed only a small part of the total amount demanded and supplied. For example, although the hydropower occupied about 6.5% of the total installed power capacity in 2009, it generated only about 1.7% of the total electricity generated in that year. As for the resources like the crude oil and natural gas, they were very limited in this country. Therefore, the diversification of energy supply sources is quite essential for this country in order to maintain a secured energy supply.

In the aspect of nuclear power development, there are currently three nuclear power plants (NPPs) in operation and one NPP under construction. Each of these four NPPs has two identical nuclear units The first nuclear power unit installed in this country was the Chinshan Unit 1, which started its commercial operation in December 1978, and the operation of the Chinshan Unit 2 was commercialized in July 1979. The Units 1 and 2 of the Kuosheng NPP were in commercial operation in December 1981 and March 1983, respectively, while the Maanshan Units 1 and 2 started their commercial operation in July 1984 and May 1985, respectively. By the end of December 2009, the total installed capacity of nuclear power was 5,144 MWe (not including the MUR power uprates) from the three operating nuclear power plants, representing about 15.9 % of the total installed power capacity in the TPC which was about 32,310 MWe. In 2009, the electricity generated from nuclear power contributed 20.7% to the total domestic supply of electricity compared to 19.6% in the previous year. The average capacity factor for all six operating nuclear units was 92.25% in 2009.

At the end of December 2009, the construction work of the Lungmen NPP with two ABWR units was near completion. The initial fuel loading of the Lungmen Unit 1 is currently scheduled to be conducted by the end of 2010, while the commercial operations of Units 1 and 2 are planned to start by the ends of 2011 and 2012, respectively. After the commercial operations of both units of the Lungmen NPP, the total installed nuclear power capacity will then become 7,844 MWe, and the share to the total installed power capacity is expected to become roughly 18.0%.

The Atomic Energy Council (AEC) is the governing authority for all atomic energy-related affairs. It was founded in 1955 at the ministerial level as a Cabinet member under the Executive Yuan which is the top administrative authority in this country. Before the programs for installing nuclear power plants began and the diverse civilian nuclear applications emerged in this nation, the principal missions of the AEC at the beginning of its establishment as well as in the early years were limited to the management of international affairs concerning atomic energy and the promotion of the peaceful applications of atomic energy. Since late 1970s, three nuclear power plants have been stably operated and applications of the atomic energy in the areas of medicine, agriculture, industry, and research have been expanding in great pace. Therefore, the most important tasks of the AEC nowadays have been shifted to the areas of nuclear power reactor safety regulation, radiation protection, nuclear emergency response preparedness, radioactive waste administration, environmental radiation monitoring, and the research and development (R&D) of the nuclear technology as well as regulations for other civilian applications of the nuclear energy.

Under the AEC, there are three affiliated organizations, including the Institute of Nuclear Energy Research (INER), the Fuel Cycle and Materials Administration (FCMA), and the Radiation Monitoring Center (RMC). The INER is the sole nuclear R&D institute in this country. As an illustration, the major R&D areas conducted in the INER in 2009 included: nuclear safety, nuclear facility decommissioning and radioactive waste management, radiobiological medicine, new energy and renewable energy, and environmental plasma applications. The FCMA has two major responsibilities: first, the safety regulation of the treatment, transportation and final disposal of the radioactive wastes which include both the low level radioactive waste and the nuclear spent fuels; and secondly the safety regulation of the import, export, storage, and transfer of the nuclear materials as well as nuclear fuels. The major responsibility of the RMC is the monitoring of natural and man-made ionizing radiation in the environment, including the radioactivity content in the civilian consumed foods.

The Taiwan Power Company (TPC) is a state-owned utility. It used to be the sole utility and generated all the electrical power needed in this country. However, because of the national policy toward the privatization of electrical power generation, the private power companies began to emerge since June 1999. The percentage of the total nation-wide electricity as produced by the TPC's power plants decreased from 100% in 1998 to 78.3% in 2003 (among which 3.9% by hydropower and pumped storage power, 52.9% by fossil power, and 21.5% by nuclear power) and 74.1% in 2009 (among which 3.4% by hydropower, pumped storage power and renewable energy, 50% by fossil power, and 20.7% by nuclear power). In the year 2009, the total installed power capacity in the TPC was 32.31 GWe, and the nuclear installed capacity (5.144 GWe) stood for 15.9% of it. The total gross amount of electricity generated in 2009 in this nation was 193.61TWh, of which 20.7% was from the nuclear power.

All nuclear power plants here are owned and operated by the TPC. With the best efforts done by the TPC staff, the performance of the three operating nuclear power plants continued to be excellent in recent years. In terms of the WANO (World Association of Nuclear Operators) performance indicators (PI), the PI values of all six operating nuclear units in 2006 were better than the WANO median values in the indicators of the unit capability factor (UCF), unplanned capability loss factor (UCLF), forced loss rate

(FLR), and chemistry performance (CP). Meanwhile, most of the six units were better than the WANO median values in other performance indicators. In some of the WANO PIs, for example, the UCF, UCLF, UA7 (Unplanned Automatic Scrams per 7,000 Hours Critical), SSP (Safety System Performance) and ISAR (Industrial Safety Accident Rate), no less than half of the Taipower's operating nuclear units were among the upper quartile of the best performed commercial power reactors in year 2006.

Generally speaking, the operations of NPPs in the ROC are quite satisfactory with respect to safety and reliability. To foster a good safety culture and to ensure that a high level of nuclear safety will continue to be the primary goal for both the AEC and the TPC, the review process of the Convention on Nuclear Safety is a good practice for Taiwan to examine the performance of its domestic NPPs and to share experiences with other contracting parties. It is of great importance to the international community to ensure that the use of nuclear energy is safe, well regulated, and environmentally sound, as stated in the preamble of the Convention. In conclusion, Taiwan complies with all the obligations of the Convention on Nuclear Safety of the IAEA.

This National Report of the ROC (Taiwan) is a self-standing document and there is no need to get familiar with the earlier reports in advance.

ARTICLE 6. EXISTING NUCLEAR INSTALLATIONS

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

6.1 Existing Nuclear Power Plants in Taiwan

There are a total of four land-based civil nuclear power plants (NPP) in Taiwan. Among them, three NPPs are currently in operation and one under construction. These four nuclear power plants, according to the sequence of the project starting dates, are named Chinshan, Kuosheng, Maanshan, and Lungmen, respectively. Each of these four NPPs has two nuclear power reactor units. All nuclear units are owned and operated or will be operated by the Taiwan Power Company (TPC), a state-owned utility.

The Chinshan Units 1 and 2 were first commercially operated on December 6, 1978 and July 16, 1979, respectively, while the Kuosheng Units 1 and 2 were on December 28, 1981 and March 15, 1983, respectively. These four units are all boiling water reactors (BWR) with the nuclear steam supply systems (NSSS) supplied by the General Electric Company (GE). The reactor type of the Chinshan units is BWR-4 with Mark I containment and that of the Kuosheng units is BWR-6 with Mark III containment. The Maanshan Units 1 and 2 were first commercially connected to the grid on July 27, 1984 and May 18, 1985, respectively. Both Maanshan units are three-loop pressurized water reactors (PWR) with the NSSS supplied by the Westinghouse Electric Corporation (W). Chinshan, Kuosheng, and Maanshan are also named the First, Second, and Third NPP, respectively, in Taiwan.

With the best efforts done by the TPC staff, the performance of the three operating nuclear power plants was steadily maintaining excellent records in recent years. In terms of the WANO (World Association of Nuclear Operators) performance indicators (PI), the PI values of all six operating nuclear units in 2006 were better than the WANO median values in the areas of the unit capability factor (UCF), unplanned capability loss factor (UCLF), forced loss rate (FLR), and chemistry performance (CP). Meanwhile, most of the six units were better than the WANO median values in other performance indicators such as the unplanned automatic scrams per 7,000 hours critical (UA7), safety system performance (SSP) (including high pressure safety injection system, residual heat removal system, auxiliary feedwater system, and emergency power supply), fuel reliability (FR), collective radiation exposure (CRE), and industrial safety accident rate (ISA). In most of the PIs, for example, the UCF, UCLF, UA7, SSP and ISA, no less than half of the Taiwan operating nuclear units were among the upper quartile of the best performed commercial power reactors in year 2006. As for the year 2009, the performance of the TPC's nuclear power units was continuously outstanding as shown in Table 6.1. Currently these performance indicator items remain the major areas targeted by the three operating NPPs to improve their performance.

In the meanwhile, the annual average capacity factor of all six operating nuclear units in recent years has been maintained at around about 90% as shown in Figure 6.1. Figures 6.2 and 6.3 show the trends of the annual average numbers of the reportable events per unit and the automatic scrams per unit, respectively.

More than 90% of the low level radioactive waste (LLRW), by volume, generated in this nation came from the three operating nuclear power plants. With the use of the High Efficiency Solidification Technology (HEST) developed by the Institute of Nuclear Energy Research (INER) as well as the plant personnel's efforts, the annual output of the solidified LLRW from these nuclear power plants drastically dropped from a peak of nearly 12,000 drums (200 liters each) in 1983 to only about 251 drums in 2009, as shown in Figure 6.4.

The power uprate program was first completed in the Kuosheng NPP for Unit 2 in July 2007 and Unit 1 in November of the same year by using the measurement uncertainty recapture (MUR) technique by installing a ultrasonic feedwater flow rate measurement system. Similar power uprate programs were also completed for the Chinshan Units 2 & 1 in July 2008 and February 2009, respectively, and for Maanshan Units 2 & 1 in December 2008 and July 2009, respectively. The total power uprated due to these three MUR programs was about 55.64 MWe as described in more detail in Subsection 14.1.2(4) of this report.

The construction of the Lungmen Nuclear Power Plant, which consists of two advanced boiling water reactors (ABWR) with the NSSS supplied by the GE, is still under way. In order to enhance the safety, reliability, operability, and maintainability of these two new power reactors, the TPC has incorporated into their design requirements the operation experience feedback of its 6 operating power reactors and the design modifications with the successful experiences from international nuclear industries.

Although the two new power reactors of the Lungmen NPP are of a standard design certified by the Nuclear Regulatory Commission of the United States (USNRC), the licensing process of the USNRC 10 CFR 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," in terms of the early site permits and combined licenses, is not applicable in Taiwan. This nation's nuclear reactor licensing regulations still follow a two-step licensing process similar to that of the USNRC 10CFR50.

The application for constructing the Lungmen NPP with two nuclear units was submitted to the Atomic Energy Council (AEC) by the TPC on October 16, 1997. After an intensive and careful review of about 15 months, the Construction License (CL) was granted for this plant on March 17, 1999. By the end of July 2009, the overall construction of the Lungmen NPP was about 90.35% completed. The commercial operation date of the Lungmen NPP is scheduled to be in December 2011 for Unit 1 and one year later for Unit 2.

A brief summary of the basic data of all the existing nuclear power units in Taiwan is given in Table 6.2, while an overview of the main technical characteristics of these NPPs is presented in Annex 1. Figure 6.5 shows the locations of these nuclear installations in Taiwan.

6.2 Major Safety Assessments

The detailed requirements of the safety assessments throughout the life of a nuclear power plant in Taiwan are described in Article 14 of this report.

6.2.1 Licensee's Nuclear Safety Culture Program

Based on the IAEA (1991) Report: "Safety Culture Series No.75-INSAG-4", the TPC developed its own nuclear safety culture (SC) emphasizing the idea of "safety first, quality top priority" and asking all its employees as well as organizations to cultivate the right ideas and proper altitude toward nuclear safety.

The nuclear safety culture program in the TPC started in 1988. The implementation of this program was divided into three phases: the Learning Period (1988 ~ 1992), Cultivating Period (1993 ~ 1997) and Enhancing Period (since 1998).

During the Learning Period, the activities conducted included: (1) requiring managers in all levels to study reports or journals related to the SC ideas; (2) opening training courses for employees to learn the SC ideas; (3) communicating and discussing the SC concepts in all appropriate meetings; and (4) publishing a special column of safety cultures in the TPC publications to talk about the SC concepts.

During the Cultivating Period, a nuclear safety culture cultivating program was first implemented in early 1993. This cultivating program consisted of five major areas of principles, namely, the responsibility, training, discipline, control, and implementation. In order to track the performance of the implementation, eighteen nuclear SC indicators were selected to evaluate the effects of the implementation of this nuclear SC program. These 18 nuclear SC indicators are:

- Implementation percentage and effects of the issues raised by the employees,
- Number of abnormal events due to human errors,
- Number of abnormal events due to the faults of procedures,
- Number of repeated abnormal events,
- Number of repeated abnormal events due to human errors,
- Number of violation cases during refueling outage or operating periods,
- Number of weaknesses undiscovered by the Department of Nuclear Safety (DNS) of the TPC, but identified by the AEC,
- Number of uncorrected weaknesses, which were previously identified by the DNS, discovered by the AEC, and
- 10 WANO Performance Indicators.

With the feedback of the nuclear SC implementation experiences, the SC indicators were reevaluated during the Enhancing Period which started in early 1998, leading to the selection of 12 new indicators as given in the following:

(a) Consequence Indicators (6 items):

- Number of abnormal events due to human errors,
- Number of violation cases during refueling outage or operating periods,
- Unplanned automatic scrams per 7,000 hours critical (UA7),
- Safety system performance,
- Industrial safety accident rate, and
- Unplanned capability loss factor.

(b) Procedure Indicators (6 items):

- Nuclear safety culture activity,
- Supervisory tool box meeting,
- Supervisory self-checking,
- Safety status improvement,
- Accomplishment rate of equipment repair request, and
- Number of systems discussion meetings.

As an illustration of the effects of implementing the nuclear SC, Table 6.3 shows some typical examples of the TPC's nuclear safety-related performance indicators before and after the implementation of the nuclear safety culture program.

6.2.2 Reporting Requirements

The licensee of a NPP is required to submit the following reports to the AEC within the required periods:

- Operation report — quarterly and annually,
- Radiation safety and environment monitoring reports — quarterly and annually,
- Nuclear accident reports — reporting within 15 minutes after knowing an accident occurred and submitting a written report within one hour,
- Radioactive waste production reports — monthly,
- Reports on in-service inspections and tests as well as containment leakage rate test — within 90 days after each refueling outage, and
- Reports of the dose evaluation for the residents who live in the vicinity of the NPP — every 5 years.

6.2.3 Regulatory Reviews/Inspections and Assessments

6.2.3.1 Application and Approval for the Construction or Operating License

To construct a nuclear reactor facility, the applicant has to submit a preliminary safety analysis report (PSAR) to the AEC for review and demonstrate its ability to fulfill the following requirements in order to obtain a CL:

- Installation must be for peaceful use.
- Equipment and facilities of the installation must be able to protect the health and safety of the public.
- Impact of the installation on the environment must meet the requirements of the relevant regulations of the Environmental Protection Administration (EPA).
- Technical, managerial and financial capabilities of the applicant must be adequate to conduct the construction, operation and back-end activities of the installation.

In order to fulfill the above-mentioned environment-related requirements, the applicant has to submit an environmental impact assessment (EIA) report to the EPA for review and approval.

After completing the construction work, in order to initially load the nuclear fuels into the reactor core, the applicant has to submit the following documents to the AEC to obtain approval for initial fuel loading:

- Final safety analysis report (FSAR),
- Reports on modifications implemented during the construction stage,
- List of the operating procedures, fuel loading plan and startup test program, and
- Systems' functional test reports.

Finally, after the completion of all necessary power tests, the applicant is required to submit the following documents to the AEC to obtain an operating license (OL):

- Approval of the EIA report,
- Updated FSAR,
- Summary of the test results during various power test stages, and
- Financial assurance of the applicant.

6.2.3.2 Integrted Safety Assessment

After the OL is granted and the plant starts its commercial operation, the licensee is required by the Regulations to conduct a comprehensive safety assessment of the operating NPP every ten years and submit an integrated safety assessment report (ISAR) to the AEC for review, six months before the corresponding ten-year operation date is due. This requirement is similar to that of the international practice of the periodic safety review (PSR). The contents of the ISAR are required to include at least the following areas:

- Review and assessment of the management of the nuclear facility over the past ten years, including the review of the operation safety, radiation safety and radioactive waste management,

- Review of the modifications or reinforcements to be implemented for the nuclear facility and explanation of the items of modifications or reinforcements committed,

- A summary which summarizes the items that should be paid attention to and the modifications committed as well as the related schedules during the next 10 years operating period, based on the above two reviews, and

- Other items requested by the Regulatory Body as needed.

Based on the above requirements, a typical ISAR submitted to the AEC by the TPC will consist of chapters with the following contents, as a common practice:

(1) Review and assessment of the plant operation safety over the past 10 year's operation history,

(2) Review and assessment of the plant radiation safety over the past 10 year's operation history,

(3) Review and assessment of the radioactive waste management over the past 10 year's operation history,

(4) Review and assessment of the committed betterment or reinforcement items,

(5) Integrated assessment of the plant aging management,

(6) Seismic safety evaluation,

(7) Evaluation of the Maanshan Station Blackout Incident (or sometimes the so-called "Maanshan 3A accident" or "318 incident") (please refer to Subsection 6.3.8(3-3) for detail) for the relevant systems of this nuclear unit, and

(8) Summary.

6.2.3.3 Regulatory Inspections

The inspection of the three operating NPPs as well as the one under construction is one of the important tasks of the AEC in its nuclear safety enhancement regulatory program.

For daily operations, the resident inspectors from the AEC for each NPP perform their daily monitoring and regulation tasks on site. Occasionally, an unannounced inspection, normally at the boring midnight time, is performed to enhance the alertness of the plant operators. The midnight inspection is regularly performed every quarter for each nuclear plant. At the end of each operation cycle, when a nuclear power unit is scheduled to be shutdown for refueling, inspection, maintenance, and modification of the structures, systems ,and components (SSCs) to assure a stable operation in the next cycle, it is essential for the AEC to examine the quality of all these activities conducted and performed by the licensee and its contractors. The AEC has established stringent requirements to audit the implementation quality of these outage activities on site in order to assure the operational safety and stability of the nuclear power unit. In addition to these regularly performed inspections, some expert team inspections based on some pre-selected topics are conducted as well. Besides, there will also be a special team inspection whenever needed.

For the NPPs under construction, the resident inspectors are dispatched to the site for monitoring and inspecting the construction activities undergoing, especially those related to the nuclear safety. The expert team and special inspections as described above are also implemented for the plant under construction.

6.2.3.4 Reactor Oversight Process

Under the AEC's policy of transparency of nuclear safety information, a compact reactor oversight process (ROP) system for a quickly evaluated and easily understood indicator of the safety status of an operating nuclear power unit was first implemented at the end of 2004. This compact ROP system used for inspecting and assessing the plant performance to ensure the plant safe operation is similar to that of the USNRC but, at the beginning, was limited to the strategic performance area of "reactor safety" with three cornerstones only (instead of three strategic performance areas with seven cornerstones) and was thus termed the compact reactor oversight process. The three cornerstones used in the early stage were the initiating events, mitigating systems, and barrier integrity (i.e., without the emergency preparedness) and consisted of only one indicator, the performance indicator (PI). Thus, initially, the AEC evaluated the plant performance by analyzing only one input, the performance indicators reported by the licensee. In the year 2005, another indicator, the inspection indicator was added to this compact ROP system and thus the AEC would analyze another input, the inspection findings resulting from the AEC's inspections, for the ROP evaluations. The assessment results of this reactor oversight process will be posted and updated quarterly on the AEC's public Web site: www.aec.gov.tw.

Beginning from the first quarter of 2009, the compact ROP included additionally the other two key strategic performance areas: the radiation safety and the safeguards. Therefore, in this ROP system, there are now seven cornerstones, including the emergency preparedness, public radiation safety, occupational radiation safety, and physical protection in addition to the three cornerstones mentioned above, within the three strategic performance areas (i.e. reactor safety, radiation safety, and safeguards). However, because of the sensitivity of information about the physical protection, the assessment result of the safeguards was not published on the AEC's public Web site.

This ROP system now consists of two categories of assessment indicators: the performance indicators and the inspection indicators. The performance indicators based on those reported by the licensee are used to evaluate the performances of the safety systems in a nuclear unit, while the evaluation for the inspection indicators is based on the findings of the AEC's inspectors. The performance indicator data were assessed and integrated with the findings of the AEC inspections. A computer code, PRiSE, developed by the INER was used in the significance determination process as a risk evaluation tool to assist the AEC inspectors to determine the safety significance (or safety concerns) of the inspection findings, if the inspector can not determine the finding as 'no safety concern' (i.e., a green one) at the very beginning.

The assessment result for each indicator will be given a color designation by a color coding system according to the degree of safety significance (or safety concerns) of it. A green color means 'no safety concern', while white, yellow and red colors stand for 'minor', 'median' and 'significant' safety concerns, respectively. The plant assessment is based on evaluations of all these indicators. Other information about this evaluation can be found in Subsection 10.5(11) of Article 10 of this report as well as at the AEC's public Web site in more detail.

Currently the AEC's ROP system consists of 15 performance indicators and 4 inspection indicators. Since the implementation of this system, the evaluation results showed the performances of all six operating nuclear units were quite good. All performance and inspection indicators were green in color except the indicator for the unavailability rate of the reactor core isolation cooling (RCIC) system of the Chinshan Unit 1, which was assigned white in color during the period from the 4th quarter of 2005 to the 4th quarter of 2007.

6.2.3.5 International Peer Reviews

The operation of the NPPs in Taiwan has been reviewed by a number of international expert groups. For example, the Institute of Nuclear Power Operations (INPO), based in Atlanta, GA, USA, has organized several intensive and in-depth reviews of the operation of nuclear power plants in Taiwan. The WANO-Tokyo Center also sent an expert team to conduct a peer review to one of the nuclear power plants every six years. The reports of these visits are very valuable. Unfortunately, these INPO and WANO reports were unavailable to the general public. Nevertheless, none of the previous groups uncovered any problems that were deemed serious enough to warrant shut down of any of the reactors, even temporarily, in Taiwan.

As an example, on November 7, 2005, the WANO-Tokyo Center (WANO-TC) organized a team of 23 experts for an 18-days visit to Taiwan in order to conduct a peer review of the Maanshan NPP. Two years later, on December 6, 2007, the WANO-TC sent another team of 14 experts to spend a total of 13 days to visit the Kuosheng NPP for carrying out a peer review on six areas, including the organization and administration, operation, maintenance, engineering support, radiation protection, and operating experience of this plant. The evaluation results and recommendations from these reviews were quite beneficial to the TPC.

6.3 Programs and Measures for Safety Upgrading

6.3.1 Regulatory Requirements for Changes and Modifications

Similar to that in 10CFR50.59, a design change or equipment modification in a nuclear power plant during the operating period must be approved by the Regulatory Body in advance before its implementation if it involves any one of the following important safety concerns:

- Change of the technical specifications,
- Resulting in more than a minimal increase in the frequency of occurrence or the consequence of an accident previously evaluated in the FSAR,
- Resulting in more than a mimimal increase in either occurrance of a malfunction or the malfunction consequence of the structure, system, and component (SSC important to safety) previously evaluated in the FSAR,
- Creating a possibility for either an accident of a different type or a malfunction of an SSC important to safety with a different result than previously evaluated in the FSAR,
- Change of the design basis limit for a fission product barrier as described in the FSAR,
- Change of the evaluation method used in establishing the design bases and safety analyses as described in the FSAR, and
- Others as required by the regulatory body.

6.3.2 Automatic Seismic Trip System

On January 17, 1995, a devastating earthquake with a moment magnitude of 6.9 struck the Osaka-Kobe area of Japan and made a great casualties and destruction. Because of the consequence of this great Hanshin earthquake (also known as the Kobe earthquake, Osaka-Kobe earthquake or Hyogoken Nanbu earthquake) in the neighboring nation, the AEC sent a message on January 28, 1995 to the TPC, asking it to study the feasibility of adopting a regulation requiring an automatic reactor scram on a strong earthquake for the operating nuclear units.

A little more than four and half years later, at 01:47 a.m., September 21, 1999, another major earthquake, measured 7.3 on the Richter scale, badly damaged the central areas of Taiwan. Almost immediately after this great 921 earthquake (also called the Gi-Gi earthquake, Chi-Chi earthquake or 921 Gi-Gi earthquake), in order to protect the reactor from seismic damage, the AEC formally sent a request to the TPC on November 4, 1999 requiring the installation of equipment that will automatically trip the reactor on a signal of strong earthquake for all three operating NPPs. The set-point of the signal to trigger the automatic reactor trip on strong earthquake is set at the design value of the operating basis earthquake (OBE). A protection of the nuclear power reactor from great earthquake by the automatic seismic trip system (ASTS) had been formally on-line for all three operating NPPs since the end of November 2007.

As an illustration, the ASTS in the Maanshan NPP is an independent reactor scram system. There is no relationship between the ASTS and the Strong Motion Accelerometer System. The ASTS includes six triaxial seismic sensors (0~1g accelerometers) and three signal conditioning panels. Three seismic sensors are installed in the auxiliary building on the elevation (El.) 74 ft floor, and the other three seismic sensors in the same building on the El.126 ft floor. These six seismic sensors' signals are connected to the signal conditioning panels respectively, and each seismic signal will compare with the previous setting in a bistable circuitry. The bistable circuitry contacts of the same separation group are parallelly wired (“logic-or”) in the signal conditioning panel. Then the “logic-or” signals are connected to both trains of the reactor protection system (RPS). If the seismic scale exceeds the setpoint of the bistable circuitry and the logic of “2 out of 3” is reached, it will actuate the RPS to trip the reactor.

6.3.3 Update of FSAR

For an operating nuclear power plant, the first update of the FSAR shall be completed within two years after the operating license is granted. The follow-up FSAR updates shall be completed within six months after each fuel reload. If one FSAR is shared by multiple units, the reference date will be set by the second unit.

However, a change or modification of the FSAR which involves the contents related to the important safety concerns as listed in Subsection 6.3.1 must submit a written application and get a formal approval before it can be done.

Examples of some important FSAR updating are as follows:

(1) Chinshan NPP:

- Modifying the description about the post-LOCA hydrogen recombiner system (2004),
- Adding the description of the load on the 125 VDC bus during the station blackout (SBO) period (2007),
- Adding a sub-section to accommodate the implementation of the ASTS (2007), and
- Adding a layout of the power sources for the remote alternate shutdown panel (ASP) (2008).

(2) Kuosheng NPP:

- Changing the test period for the integrated leak rate test (ILRT) of the containment (2005),
- Adding the quality classification of the 5th emergency diesel generator (EDG)(2006),
- Adding the high efficiency solidification volume-reduction system for the wet radioactive wastes (2006),
- Modifying the specifications of the nuclear fuel to that of ATRIUM-10 (2006), and
- Adding a sub-section to accommodate the implementation of the ASTS (2007).

(3) Maanshan NPP:

- Removing the standard technical specifications (STS) out of Chapter 16 and making the necessary modifications to accommodate the adoption of the improved technical specifications (ITS) (2004),

- Modifying the description about the nuclear fuel to accommodate the change of fuel rod length (2004),

- Modifying the description about the emergency preparedness to accommodate the implementation of the newly promulgated Nuclear Emergency Response Act (of 2003) on July 1, 2005 (2005),

- Adding a sub-section to accommodate the implementation of the ASTS (2007), and

- Adding a description about the incinerator system for the low-level radioactive wastes and its flowchart (2008).

6.3.4 Update of Technical Specifications

According to the Nuclear Reactor Facilities Regulation Act and its Enforcement Rules, during the operating period of a nuclear reactor facility, any design modification or equipment change which involves the modifications or revision of the technical specifications (TS) must be approved by the regulatory authority in advance in order to conduct the modification or change.

Ever since its installation, each of the three currently operating NPPs in Taiwan adopted the customer TS (CTS) (Chinshan NPP) or standard TS (STS) (Kuosheng and Maanshan NPPs). In 1988, the AEC asked the Chinshan NPP to replace the CTS with the STS (NUREG-0123, Rev. 4). However, in the early 1990's, the TPC noticed the development of the improved STS (or simply the improved TS) in the USA and initiated a project to convert the Chinshan's TS to ITS.

During the process of the above-mentioned TS conversion, the AEC agreed to the TPC's proposal to convert the CTS of Chinshan NPP directly to the improved TS based on "NUREG-1433 Revision 0" in 1992, which was later changed to be "NUREG-1433 Revision 1" in 1995. Finally, on February 26, 2002, after the scheduled outage of the 18th end of cycle (EOC-18) of Unit-2, the ITS was implemented in the Chinshan NPP.

Not only the TPC but also the AEC spent a lot of efforts on these TS conversion project-related affairs. In converting the TS, dozens of programs and hundreds of procedures were reviewed and revised. The entire operating crew of the Chinshan plant was trained several times.

Being a twin-unit station, Chinshan is the first NPP installed in Taiwan and has been operated with the CTS for about 24 years. Then, on Feb. 26, 2002, the Chinshan NPP converted its customer TS into the improved technical specifications. At the time when the Chinshan plant adopted the ITS, its unit 1 was in normal operation, while the unit 2 was shut down for refueling outage. Although the converting process was very energy consuming, the outcome of the implementation of ITS is very fruitful. Therefore, both standard TSs for the Maanshan and Kuosheng NPPs were also converted into the ITS in September 2004 and January 2008, respectively. More information about the implementation of the ITS in the domestic NPPs can be found in Subsection 10.5(9) of Article 10 of this report.

The content of the original CTS or STS was too complicated. It contains too many things and the explanations of the bases for the settings of the TS items were not clear enough. In addition, the requirements of some TS items were not quite clear and not addressed in a format consistent with the human engineering. Thus, it often gave the operators some headache in checking the CTS or STS. After adopting the ITS, the fire-fighting equipment, core operating limits report (COLR), radiation protection and environment monitoring, snubbers, equipment checklists, water chemistry, etc., were removed out of the TS and controlled by the Technical Requirement Manual (TRM) or programs instead. This reduced the administrative load and improved the performance of the plant. For example, the limiting conditions for operation (LCO) become more safety oriented, the allowed outage times and the surveillance requirements are optimized, and all the requirements are supported with strong bases.

6.3.5 International Cooperation

6.3.5.1 General

The TPC maintains good relationships with many international organizations. For example, the TPC is a member of the WANO, the Institute of Nuclear Power Operations (INPO), the Electric Power Research Institute (EPRI) and the Nuclear Procurement Issues Committee (NUPIC).

The TPC also participated in several mutual cooperation programs. Both the AEC and TPC jointly participated in the USNRC’s Thermal-Hydraulic Code Applications and Maintenance Program (CAMP) in 2004. In the meantime, the Chinshan NPP maintains a sister relationship with the Fukushima Daiichi Station of Japan, while the Kuosheng NPP is a sister plant of the Shimane Station.

On the other hand, the AEC joined the USNRC’s Cooperative Severe Accident Research Program (CSARP), the International Cooperative PRA Research Program (COOPRA), the OECD/NEA Cooperative Program on Decommissioning (CPD), and the OECD/NEA Computer-Based Systems Important to Safety (COMPSIS) Project.

For twenty years (up to 2009), the nuclear communities in both Taiwan and Japan take turns in hosting the annual Sino-Japanese Seminar on Nuclear Safety, which was usually held in the November. In these seminars, members from both parties presented papers and discussed topics which were mutually interested. This seminar has successfully served as a forum for discussing nuclear safety issues and exchanging operating experiences from both sides, and has proved mutually beneficial to both communities.

Besides, the TECRO-AIT Joint Standing Committee meeting on civil nuclear cooperation, which was held in turn in Taiwan and USA annually, gave a good opportunity for Taiwan to exchange her experience on nuclear regulations and operations with those of the USA. (TECRO represents the Taipei Economic and Cultural Representative Office in US and AIT stands for the American Institute in Taiwan.)

6.3.5.2 Seismic Study

Earthquake is one of the most important safety concerns in the design of a nuclear power plant, especially in Taiwan which is located in a seismic hazard zone. In the past years, although there were a lot of studies regarding the soil-structure interaction (SSI) phenomenon during the earthquake events, the SSI seismic design approach for a nuclear power plant was not satisfactory because of the lack of solid and realistic database associated with the SSI.

From 1985 to 1990, the TPC, in cooperation with the EPRI, performed the Lotung Project in Taiwan to study the soil-structure interaction for a site with the soft structure. As a result, several SSI analytical computer programs were successfully developed. Then, from 1990 to 2001, the TPC cooperated with the EPRI and other members such as the USNRC, Tokyo Electric Power Company, Central Research Institute of Electric Power Industry, Korean Group, and French Group to construct a 1/4 scaled containment test model on a Hualien site in eastern Taiwan with the hard structure of sand-gravel deposits. This cooperation project is called the Hualien Project with the mission to collect relevant data of the soil-structure interactions during the earthquake events.

By integrating the results achieved from both the Lotung and the Hualien Projects, a comprehensive knowledge for the soil-structure interaction was obtained. This knowledge is used for the verification and modification of related computer programs. The major achievements of these international cooperation programs on seismic study are as follows:

(a) Confirmation of the adequacy and validity of various SSI analysis methodologies, procedures, and related computer programs, and

(b) Construction of a full scope seismic database which could be widely used in the seismic engineering and research.

6.3.6 Probabilistic Risk Assessment and Its Risk-Informed Application

The development and application of the probabilistic risk assessment (PRA) technology in Taiwan can be divided into three phases. First, beginning in 1982, the AEC initiated a PRA program for the domestic nuclear power plants. Comprehensive PRA models were completed for the Kuosheng, Maanshan and Chinshan plants in 1985, 1987 and 1991, respectively. The possible core melt scenarios and their associated frequencies induced by the internal events as well as the external events, including earthquakes, typhoons, fires and internal floods, were investigated.

However, the PRA models established in the first phase had some drawbacks in their applications later on. These PRA models could not take into account the updated plant status with successive design changes. Besides, these models were installed on the mainframe computer with implicit complexity and thus were quite difficult to use. To improve this situation, an intensive project entitled “Application of PRA to the Daily Operation of Nuclear Power Station” was initiated by the TPC in the second phase starting from 1994 to 1997. The major tasks of this project included the PRA model update, the conversion of the model from the mainframe computer into the personal computer based tools, and the risk analysis for plant outage. At the end of this phase, the so-called “Living PRA” models were completed for all operating plants. These models are not only user-friendly but also easy to be modified with any changes of the plant systems.

Since 1980's, the increasingly competitive power generation market demands broader initiatives for reducing operation and maintenance (O&M) costs while maintaining plant safety. It is believed that the so-called "risk informed" approach is appropriate to be used to drive down the O&M costs without impairing the safety. This approach was the major application of PRA in the third phase (from 1998 up to now). In this phase, for example, a project entitled "Establishment and Application of TPC Risk Integrated Monitor (TRIM)" conducted mainly by the INER was sponsored by the TPC to develop an integral risk management system based on the plant specific living PRA models. This risk management system combined the PRA model and the plant supporting software into a user-friendly analytic tool. With this tool, the plant operators and managers are able to easily obtain the precise plant configuration for decision making.

Currently, the maintenance rules of the three operating NPPs all adopted the risk-informed concept. Prior to the conduct of the maintenance actions including (but not limited to) the surveillance tests, after-maintenance tests, corrective actions and preventive maintenance, the risk imposed by this maintenance practice will be assessed and controlled for the SSCs which were judged as safety-important by the risk-informed evaluation.

6.3.7 Re-qualification of Licensed Reactor Operators

According to Article 10 of the Regulations on Nuclear Reactor Operators' Licenses as amended in December 2009, the licensee of a nuclear power plant must submit a reactor operators' retraining program to the AEC for approval and all its reactor operators must be retrained in accordance with this approved retraining program and pass the examination at the end of the retraining. The retraining program should follow the retraining guidelines described in the Appendix 4 of the aforementioned regulation and should be reviewed every two years and implemented continuously. Contents of this retraining program should include the following:

(1) Schedule planning,

(2) Classroom training,

(3) Operation training,

(4) Methods and timing for the evaluation, and

(5) Documentation of the evaluation and training records.

The retraining guidelines require that a reactor operator must be retrained at least 90 hours on courses in the classroom and at least 30 hours on simulator every year and must pass the licensee's annual retraining examination.

Thirty days before the expiration of the license, a reactor operator needs to prepare such documents as the physical examination report for the latest one year, the licensee's recommendation letter, and a retraining certificate to apply for a license renewal.

In the meantime, the AEC may perform an operator re-qualification test which consists of both written examination and operational test on simulator as well as an oral examination on site if the AEC deems it necessary. According to Article 13 of the Regulations on Nuclear Reactor Operators' Licenses (as amended in 2009), possible candidates to be asked for this AEC's re-qualification test are those operators whose performance involved any one of the following concerns during the nuclear reactor operating period:

- Poor operational performance,
- Poor result and poor quality of his operator retraining, and
- Violation to his responsibility or having operational fault(s).

For those operators having been asked for re-qualification but unable to pass the test, the licensee must terminate the assignment of them to operate the nuclear power reactor immediately upon receiving the formal notice from the AEC.

6.3.8 Significant Corrective Actions

In each of the TPC's nuclear power plant there is a corrective action program (CAP) of the plant itself. This CAP program integrated all the mechanisms for resolving various problems of the plant into a system to carry out the tasks of problem discovery, classification, correction, following up, analysis, and resources integration. The purpose of the integrated CAP system is to enhance or improve the root cause analysis, common-cause analysis, trend analysis, evaluation of the effectiveness of the corrective actions, and the plant health indicator. In the Maanshan NPP, for example, the CAP is named the Excellent Management System.

In the following paragraphs, some representative or significant corrective actions implemented in the past years after the occurring of events or simply for the plant improvement in the nuclear power plants in Taiwan will be presented.

(1) Chinshan NPP :

(1-1) The Torus Crack Event

On November 25, 1994, the Chinshan Unit 1 was connected to the electrical power grid after its restart on the previous day. In accordance with the operating procedures, nitrogen gas was charged into the containment from 15:23 of November 25th. However, on the next day, a weld on a 6-in. pipe up-stream of the nitrogen charging valve SB-210 was found broken. The oxygen concentration in the containment was about 10% at this time and the operators gradually reduced the reactor power and shut the nitrogen charging system down according to the requirement of the Technical Specifications. Furthermore, a crack was found by the maintenance crew at the upper part of the outer shell of the torus (just beneath the broken weld). The reactor was brought to cold shutdown by operators. The root cause of the broken pipe and the cracked shell was later judged to be due to the effect of low temperature of the nitrogen gas. The major corrective and preventive actions to this event included:

- Repair of the broken pipe and the cracked torus shell,
- Regular check of repaired parts,

- Revision of nitrogen charging procedures to prevent adverse effects caused by low temperature,

- Improvement of nitrogen heating system to enhance its temperature control function during nitrogen charging operation, and

- Installation of temperature detectors on the important equipment near the nitrogen charging pipes.

(1-2) Core Shroud Repair

In March 1994, during the outage maintenance of the Chinshan Unit 1, cracks were found on the H3 weld of the core shroud. These cracks grew slightly when they were checked on the next outage maintenance (March 1995), and several more cracks were identified on the welds H5 and H6 as well. After intensive safety assessment made by the TPC, it was judged that the growth rates of these cracks were within the safety allowance for one more operational cycle. Therefore, the AEC agreed that the Unit could continue to operate until the end of that cycle and then the core shroud repair had to be performed during the following outage (April 1996). The major task of this repair plan was to install 4 sets of stabilizers between the core shroud and the reactor pressure vessel (RPV) to replace the structural function of horizontal welds H1 through H7 of the core shroud. The following requirements were strictly followed by the TPC for the repair work:

- Design requirements of the core shroud as described in the FSAR must be maintained.

- Designed lifetime of the repaired components must be no less than the remaining life of the Unit 1 reactor (including the consideration of plant life extension).

- Materials for repair must conform to the ASME code and the recommendations made by the BWR Vessel Internal Program (BWRVIP).

- Prevention of the Inter-Granular Stress Corrosion Cracking (IGSCC) and Irradiation-Assisted Stress Corrosion Cracking (IASCC) must be considered in the selection of repair materials.

- No welding is allowed in the repair processes.

- Loading and stress analyses must be performed for the repaired components, core shroud, and the RPV with all accidental conditions postulated in the FSAR.

- After the repair, only 0.46% of core flow is allowed to leak from the core bypass to the downcomer region under the worst accident condition.

Before the repair work was conducted, the TPC had made an integrated safety assessment and the major conclusions of it were as follows:

- Both the probability of the occurrence and the severity of the consequences of all the accidents postulated in FSAR were not significantly changed due to the repair of the core shroud.

- The repair would not give rise to the occurrence of possible accidents that are not included in the FSAR.

- Safety margins reserved in the Technical Specifications were not affected due to the repair.

The repair work was contracted to the GE and completed in 1995. A PRA conducted with the plant conditions after the repair showed that the increase of core damage frequency was 1.6E-7. Therefore, the safety impact of the core shroud repair to the plant safety was considered to be negligible.

(1-3) Low Pressure Turbine Blade Crack Event

On June 30, 1992, an alarm of high bearing vibration occurred at the #2 low pressure (LP) turbine of the Chinshan Unit 2. The examination after the reactor was shutdown found that the L-2 blade of the LP-1 rotor was broken. The reactor resumed operation after the damaged rotor had been replaced by a spare one. However, similar failures were found in the other LP turbines during the following outage maintenance of the Chinshan plant. Since there was not enough time to analyze the root causes and find the remedies, the TPC, with the help of the original turbine supplier, Westinghouse, decided to replace all L-2 blades of the LP turbines by new ones.

Nevertheless, after about 6,024 hours of operation, the cracks appeared again on several new L-2 blades and the root cause was still unknown. Under this circumstance, the Chinshan plant, to ensure the operational safety, could only shut the reactor down and change the turbine rotor every 6 to 8 months, having the blade cracks thought to be critical. This kind of practice had been performed until all the LP turbines were installed with the new rotors supplied by a different vendor in 1999. These new rotors were forged section by section and then welded together. By this way of manufacturing, the problems of stress corrosion cracking associated with the shrunk-on type rotor were effectively solved. In addition, the ability to prevent the problems caused by torsional vibration and stall flutter was greatly improved in the design of the new rotors. The Chinshan plant has been finally relieved from this operational burden associated with the turbine blade failure since 1999.

(1-4) Main Generator Stator Partial Rewind

During the Chinshan Unit 1 EOC-23 refueling outage in October 2008, a main generator stator winding transposition test was performed. The generator stator failed the test. The test showed there were general insulation breakdown existed between strands. Under the instruction of engineers from SIEMENS at site, the original equipment manufacturer (OEM), the suspected broken strands located in the top coil of Slot 29 were isolated (i.e. cut out) and measured. The results showed that some broken strands did exist.

After pulling out and thoroughly inspecting, it was surprisingly found that one of the coil tubes of the 30-years old replaced stator bar was deformedly bent. This was an obvious manufacturing defect. The deformed tube limited most of the flows of hydrogen, resulting in the over-heated insulation degradation between strands. The broken spots of the strands were found in the midsection. Thus, the stator bar (top coil of slot 29) had to be removed and replaced with a spare stator bar. In addition, the fact that low resistance between tube and copper revealed the existence of some shorts between them. To avoid developing potential phase-to-phase or ground shorts, the plant eventually had the damaged stator bar rewound.

(2) Kuosheng NPP :

(2-1) Workers Over-Exposure Accident

At the night of March 20, 1993, several workers were transferring spent control rods from the reactor to the spent fuel pool of Unit 2 of the Kuosheng plant. Because of improper operation during the transfer process, six workers received abnormal radiation dosage. The highest dose taken by one of these workers was 299.9mSv. Immediately after the TPC's notice, the AEC sent two section chiefs, one from the Department of Nuclear Regulation and the other from the Department of Radiation Protection, to the Kuosheng site on the next day (March 21, 1993). Several review meetings were held in the following days to find out the root causes and appropriate corrective actions for the accident. After these meetings and the relevant discussions, the following major actions of the TPC were required by the AEC:

- The hanging equipment of the control rods on the side of the spent fuel pool was not allowed to be used any more.

- The review process of the radiation safety and working safety associated with special operations needed to be improved.

- For operations with high risk of radiation exposure, there must be health physics personnel at the site of the operation to monitor the radiation level and oversee the working procedures.

- Workers had to carry an alarm type dosimeter while working in high radiation areas.

(2-2) RPV Over-Pressure at Low-Temperature Event

In the early morning on November 7, 1993, while the outage maintenance of the Kuosheng Unit 1 had been performed for 63 days, an RPV leak test was completed at 05:00 and the pressure inside the RPV was about 72.1 kg/cm^2 (≈ 1025.6 psi) at that time. Thirty minutes later, a group of maintenance workers started to perform a scram timing test for the control rods, with the RPV pressure kept at about 70.7 kg/cm^2 (≈ 1005.6 psi). At 07:41, after the recirculation pump B was switched from low speed to high speed for the vibration test, the RPV pressure started to increase. Consequently, a reactor high-pressure trip occurred at 07:45 with the RPV pressure at about 81.5 kg/cm^2 (≈ 1159.3 psi). The recirculation pumps A and B were also tripped a few seconds later due to the anticipated transient without scram (ATWS) protection signal. In order to mitigate this situation, a reactor operator manually stopped the control rod drive (CRD) pump and raised the reactor water clean-up system (RWCU) dump flow at 07:46. Sixteen seconds later, the RPV pressure dropped to 72 kg/cm^2.

The main root cause of this event was considered to be the switching of recirculation pump from low speed to high speed when the RPV water was nearly in a solid state.

Fortunately, the maximum RPV pressure in the event, about 81.5 kg/cm^2, was well below the safety limit (about 93.2 kg/cm^2). The temperature rising rate did not exceed the safety limit of 55ºC/hr during the event either. However, it was found in the relevant operational manual that only when reaching both the minimum RPV pressure (21.1 kg/cm^2) and the maximum RPV temperature (100ºC) was it required to switch the re-circulation pump from low speed to high speed. Obviously an upper limit of the RPV pressure allowed should also be added to the requirement. Therefore, correction of the operational manual and relevant training of operators were the most important corrective actions for this event.

(2-3) MCR Control Panel Loss of Power Event

On March 16, 2000, the outage maintenance of the Kuosheng Unit 1 had been performed for 13 days. At time 13:55, an “NSSS/BOP ANNUNCIATOR POWER SUPPLY FAILURE” alarm appeared and then all the alarms in the control room were out of work. The reactor operator informed the working crew to stop all the maintenance activities and declared it a second category nuclear accident (ref. to: Subsection 16.1.1(3) of this report for the accident categorization). All members of the Technical Support Center (TSC) of this plant reported at 14:10 to the Plant General Manager (who was previously called the Superintendent) according to the emergency procedure.

It was decided at 14:20 that this event should be categorized in the first category, after the investigation of the TSC members. The reason was that the Unit was under outage maintenance and the reactor remained shutdown during the event. The root cause was then found to be the inadvertent trip of the DC Breaker 1DE01B, and the control panels were back to normal condition after the breaker was reset. The major corrective actions resulted from this event included:

- To install protective cover on the DC Breaker 1DE01B to prevent it from inadvertent touch,

- To install redundant power supply for the NSSS and the balance of plant (BOP) annunciators in the main control room (MCR), and

- To include the reactor operational condition in the emergency response procedure so that the accident category can be judged more accurately than before.

(2-4) Generator trip on Loss of Excitation Caused by the AVR Failure

(a) Event Description:

On September 3, 2007, while operating at 98% thermal power with an electric power output of 975 MWe, the Kuosheng Unit 1 had experienced a generator loss of excitation which initiated a turbine-generator trip and the subsequent reactor scram.

During the incident, the rectifier wheel of the generator exciter was found being damaged slightly while all the plant safety systems have functioned normally as designed and there was no radioactivity release to the environment.

The follow-up event investigation revealed that the generator’s automatic voltage regulator (AVR) failed to function properly at the very first stage. According to the Time History Data Trend, it showed that right before the incident, the generator exciter field voltage and field current had dropped significantly with a consequence of the generator reactive power being reduced from 115 MVAR to -1,036 MVAR. On detecting a ‘loss of excitation”, after a default two-seconds time delay, the on-duty channel II AVR had switched over to the backup channel I for control. Although the AVR channel I functioned properly to raise the field voltage to the ceiling value, it was still too late to preclude the generator from being tripped on loss of excitation with the protection relay being triggered one second before.

During the incident, the chronological sequence on AVR control was listed as follows:

16：32：43.6	Channel II AVR was in normal control, with exciter field voltage 62.933V and generator reactive power 115.12 MVAR.
16：32：43.8	Channel II AVR failed and caused the exciter field voltage dropped to zero.
16：32：45.2	Generator reactive power was reduced to -262.5 MVAR, which initiated the AVR channels switch logic with 2-seconds time delay.
16：32：46.4	Generator reactive power continuously dropped to -637.65 MVAR, which initiate the #340 “loss of excitation” protection relay actuation with one-second time delay.
16：32：47.2	After a 2-seconds time out, the AVR was automatically switched from channel II to channel I for control, and the exciter field voltage was then boosted from the minimum of -1,036 MVAR to the ceiling value.
16：32：47.4	When the one-second time delay was counting out, the #340 “loss of excitation” protection relay was actuated to trip the main generator.
16：32：47.6	The turbine-generator trip resulted in the subsequent reactor scram.

Obviously, the loss of excitation trip at the Kuosheng Unit 1 was attributed to the channel II AVR failure with unidentified cause which was under investigation by the vendor, the ABB in Switzerland.

(b) Analysis / Comments:

In order to provide the redundant channels for the AVR control, both Kuosheng units had adopted the ABB’s new design of digital AVRs to replace the original analog type. The modifications on both units were fulfilled during the previous refueling outages.

For this incident, the AVR manufacturer (the ABB) had dispatched a service team as the task force to the Kuosheng site to take further measures and investigate the root cause. The duty channel (channel II) had been disassembled and sent to the ABB in Switzerland for further tests and investigations.

From the Failure Report issued by the ABB, it indicated that the disassembled channel had been tested thoroughly with the same environment as in the Kuosheng unit. The channel performed without any problems during a period of one week. Neither hardware nor software failures had been found.

The ABB believed that, if the AVR was the initiator of the event, this incident could be regarded as an occasional and unexpected event (random failure). A systematically fault of the product itself could therefore be excluded.

Before this incident, the Unit 1 of the Kuosheng NPP operated very well. There was no instability problem in the power system network or other disturbances, such as transient voltages, etc. The failure sequence occurred in the Kuosheng NPP was quite atypical. Obviously, the AVR was loosing its performance.

So far, the ABB did not find out any kind of failures in neither hardware nor software area in the channel II. The ABB will be trying to perform more tests for further investigations.

(c) Corrective Actions:

After the incident, a field inspection disclosed that some damages found inside the exciter. Two diode modules and one fuse were found blown by a flashover during the incident. In order to make a thorough investigation for the root cause and to replace the affected exciter, the Kuosheng NPP decided to commence the Unit 1 refueling outage thereafter, which was a few days earlier than scheduled. During the outage, the channel II AVR and the damaged exciter had been replaced accordingly. Besides the reliability problem, one of the root causes could be the time delay setting of the AVR switch-over function. It was decided then to change the setting of the channel switch-over in case of loss of protection to 0.5 second, instead of the original default value of 2 seconds.

Considering the time delay setting, a too short time delay could cause more opportunities to activate the inadvertent channel switch-over, whereas a too long time delay could cause a higher damage of related equipment. The setting was modified for the Kuosheng Unit 1 during this outage. A similar modification on the Kuosheng Unit 2 AVR was conducted as well during the next earliest scheduled shutdown.

In addition to that, the Surge Suppressors would be equipped in both units of the Kuosheng NPP for reducing probable or un-expected voltage spikes in the exciter field. This would protect the equipment from some kind of transient voltage.

(3) Maanshan NPP :

(3-1) Turbine Building Fire Event

At the time about twenty minutes passed 17:00 in the afternoon of July 7, 1985, while the Maanshan Unit 1 was operating at 97% power with an electrical power output of about 885 MWe and the reactor was under automatic control, an accident happened in the turbine building of this unit. The major scenarios of this accident are described as follows:

- At 17:21:00, the operators in the MCR felt high-frequency and low-amplitude quakes. Then it was found that both the reactor and the turbine were automatically tripped.

- At 17:21:40, 5 out of the 10 turbine vibration monitors reached their full-scale indications, while the other 5 dropped to zero. This phenomenon implies that the turbine had experienced very strong vibrations.

- At 17:21:42, the record of the process computer showed that the turbine was tripped earlier than the reactor.

- At 17:21:45, it was found that the hydrogen pressure inside the generator quickly dropped, implying that a large amount of hydrogen leaked out of the generator.

- During the initial 40 to 60 seconds of the event, the operators in the MCR heard a huge "ban" noise coming out from the turbine building.

- At 17:22:00, 5 to 6 local operators entered the turbine building and found heavy smoke coming out of the generator. They started to put out the fire immediately.

- At 17:27:30, the automatic fire extinguishers around the turbine and generator started to spray water, but the fire kept on.

- At 17:40:00, the plant fire brigade joined the fire extinguishing action.

- At 17:50:00, the fire department from the local county joined the fire fighting action.

- At 18:50:00, the fire was put out and the event was finally terminated at the same time.

After the event, it was found that the exciter, the generator, the low-pressure turbines, and the high-pressure turbine were all damaged. The event was later investigated and found to be caused by 8 broken blades in the low-pressure turbine near the generator. These 8 broken blades squeezed, pressed, and collided with the other blades in the turbine, caused a huge and unbalanced force on the shaft of the turbine and severely damaged the equipment nearby. From the root cause analyses, it was found that the natural frequency of the torsional vibration for the turbine/generator set of the Unit 1 was about twice that of the electrical system and this caused the large resonant vibration on the turbine and breaking 8 blades at the last stage. The major corrective actions resulted from this event included:

- As an immediate but temporary action, removing out the blades at the last stage of low-pressure turbines of the Unit 2 which was without event, it hence changed the natural frequency of torsional vibration to 119.05 Hz. This condition would be effective only for one operational cycle and the load of the unit was limited to 500 MWe (about 52.5% power) to ensure the safety of the operation of this unit.

- In the long run, the natural frequency of torsional vibration for the turbines in both units needed to be modified into the range smaller than 118 Hz or larger than 122 Hz. The modification of the Unit 1 needed to be completed before its restart, and that for the Unit 2 needed to be completed in the next first outage maintenance.

- The fire lasted for about one and half hour during the event because of the following reasons:

  (i) Turning off the lubrication oil pump for the low-pressure turbine was not early enough, and

  (ii) The firemen tried to save the generator so that water was not used to extinguish fire at the beginning.

  Accordingly, some hardware and software modifications to the fire fighting systems had been made to prevent the above-mentioned mistakes from happening again.

(3-2) Control Rods Crack Event

On September 24, 1988, after a reactor trip at the Maanshan Unit-1, the digital rod position indicator showed that the rod cluster control assembly (RCCA) R41 was stopped at step 12, which represented a distance of 7.5" away from the fully inserted position. In addition, two subsequent rod drop tests performed on R41 showed it stopped at steps 12 and 18, respectively. (For the readers' information, in the control rod system of the Maanshan NPP, the control rod positions are identified in terms of steps with step 0 for fully insertion and step 228 for fully withdrawn. The neutron absorber material used for the control rods then was hafnium.)

Because it was only a few days before the scheduled refueling outage, the TPC decided to proceed with the refueling maintenance immediately in parallel with the root cause investigation of the stuck rod event. During the investigation, underwater TV inspections were performed for all RCCAs in the core. Some cracks were observed on the rodlets of RCCA R41. No cracks were found on the other RCCAs. Among the twenty-four rodlets of R41, one corner rod had its end-plug detached. Another interior rod had its end-plug partially separated by a circumferential crack. One peripheral rodlet was found to have axial cracks in its tip portion. In addition to these three rodlets, five corner rods, three interior rods, and one peripheral rod were also removed and transported to a hot-cell laboratory of the INER for further examination.

The work scope in the hot-cell laboratory included: (1) visual inspections, (2) profilometry measurements, (3) metallographic examinations, and (4) fractographic examinations. In addition to the three cracked rodlets identified by the underwater TV, the hot-cell results revealed that seven more rodlets, out of the twelve examined, contained cracks. All the unusual phenomena, such as broken end plugs, multiple axial cracks, bulges and circumferential cracks were caused by a combination of the volumetric growth (volume expansion of the hydrided hafnium and irradiation embrittlement of the cladding) and the differential thermal expansion between the hafnium rod and the SS 304 cladding. Stress analyses were performed to determine the conditions needed for creating internal/external incipient cracks and circumferential cracks. In general, external incipient cracks were considered to be the result of a local interaction between the hafnium hydride and the stainless steel cladding, while internal incipient cracks were attributed to the ring-like formations of the hafnium hydride. After the axial movement of the hafnium rod within the cladding became restricted (because of blockages caused by hafnium hydride induced swelling), further hydriding at the end face was suspected to cause the circumferential cracks. To prove that the RCCA cracking being a generic issue in the Maanshan Unit-1, a second poolside inspection was conducted with periscope. This inspection revealed that twenty-one out of forty-six examined RCCAs contained cracked rodlets. The distribution of the defective RCCAs in the core did not show a clear dependence on the in-core position or the bank type.

After the root causes of the hafnium control rod failures were identified, then, under the demand of the regulatory body, the TPC decided to replace all the RCCAs containing hafnium with the control rods containing a mixture of silver, indium, and cadmium as the absorber materials.

(3-3) Station Blackout Incident

On March 17, 2001, Units 1 and 2 of the Maanshan Nuclear Power Plant were tripped at 03:21 and 03:23, respectively. This event was caused by the instability of the offsite extra voltage (345 kV) power transmission line, which in turn was caused by the seasonal sea smog containing salt deposit. While the reactor was maintained in hot standby condition after the reactor scram, the 345 kV offsite power supply system was still unstable.

At 00:41 of the next day (March 18), the power supply of the essential bus A of Unit 1 was automatically transferred from the 345 kV to the 161 kV offsite power supply because of the loss of the 345 kV offsite power. A few minutes later, at 00:46, the 161 kV offsite power was lost too. Right after that, the emergency diesel generator (EDG) A was successfully started, but cannot supply power to the essential bus A due to a bus grounding fault. In the meantime, the EDG B could not generate power because of losing excitation either. This situation of losing all AC power (i.e., station blackout) in Unit 1 lasted for almost two hours until the swing EDG (i.e. the fifth EDG in the Maanshan Nuclear Power Plant) successfully started at 02:50 and supplied power to the essential bus B at 02:54 of that day. This incident, a site area emergency incident without any release of radioactive material, was considered to be a station blackout incident. After the incident, the two units of the Maanshan were required to remain shutdown for corrective actions. For Unit 1, the operators were asked to find out the root causes and corresponding counter measures of the incident, and to make sure that no major equipment was damaged. For Unit 2, it was also required to make sure that no potential risk may cause an incident similar to that happened in Unit 1. The two units were allowed to re-start again, after the AEC confirmed that the above requirements were fulfilled.

Since this incident occurred on March 18, 2001, it was also called the Maanshan 318 incident (or simply the 318 incident). However, according to the regulation: "Emergency Response Plan" which was effective until July 2005, the nuclear accidents were classified into four categories, namely, (1)notification of unusual event, (2)alert, (3)site area emergency, and (4)general emergency. Depending on the degree of radiation release, Categories 2 and 3 were further divided into A, B and even C classes. The Class A stands for an event without release of radioactive materials to the environment. Since the above-mentioned Maanshan SBO incident belonged to Category 3 but without radiation release, it was thus also named as a 3A incident domestically. This incident was also called the 318 incident. (The Emergency Response Plan regulation was amended to become "Emergency Response Basic Plan" which was effective on July 1, 2005 and simplified the nuclear accident classification to become only three categories by discarding the category of abnormal event notification.)

(3-4) Modifying the Spare 345 kV Startup Transformer to Become a Standby One

According to the operating procedures, if the startup transformer MC-X01 is de-energized for a period more than 72 hours, the nuclear power unit must be shut-down. In order to avoid this problem when the transformer MC-X01 needs to be de-energized for a period longer than 72 hours for inspection and maintenance, the spare startup transformer MC-X04 was upgraded in May 2005 to become a standby startup transformer. Therefore, whenever the transformer MC-X01 is unavailable, MC-X04 can be immediately on-line to take over the required functions of the startup transformer and the operational stability is improved.

(3-5) 345 kV Startup Transformer Fire Event

There are two startup transformers, MC-X01 and MC-X04, in the Maanshan NPP, one in use and the other for backup. The startup transformer is used to supply the plant power during the reactor startup and refueling outage. When in normal power operation, the startup transformers become the backup power supply source for the safety equipment.

On June 12, 2009, while both units of the Maanshan NPP were at full power operation, the 345 kV startup transformer (MC-X04) was in its normal energized condition as a backup power supply to the safety equipment. At 15:13 in the afternoon, a lot of alarms appeared in the MCR. The gas cooled breakers GCB-3510 and GCB-3520 were automatically opened and the fire water spray system was actuated. The plant personnel at the field reported that the 345 kV startup transformer, MC-X04, was on fire. Operators in the MCR immediately notified the fire brigade of the station to take action and, because the fire was reported to be big, asked the local community fire department for help simultaneously. Then, the shift manager conservatively decided to reduce the reactor power of both units: from 100% to 91% for Unit 1 and to 94% for Unit 2. The fire was completely extinguished at 15:48.

During the root cause investigation, it was found that in the upper portion of the expansion housing of the phase B high voltage insulation sleeve (or the bushing body), there was a small rusty perforation. This caused the rain water and moisture to intrude into the high voltage insulation sleeve. The intruded water in turn failed the electricity insulation and caused a flashing and an instant high oil temperature and high oil pressure which resulted in the opening of the hand-hole cover in the tube-side of the phase B high voltage insulation sleeve. The high temperature and high pressure flashing oil moisture coming out of the sleeve immediately ignited upon in contact with the oxygen in the atmosphere.

The follow-up corrective actions included the following areas:

- Check all high voltage insulation sleeves with identical design to see if there were perforation phenomena,
- Improve the design of the outer casing of the high voltage insulation sleeve to prevent the deposit of water,
- Perform the necessary non-destructive test (NDT) while doing maintenance on the transformer, and

- Include the relevant maintenance or check-up steps into the procedures.

6.3.9 Research and Development Programs in Nuclear Safety

The INER is the sole R&D institute in the field of civil applications of the nuclear energy in Taiwan. One of the INER's major missions is to promote the domestic technology for nuclear safety. It regularly carries out R&D programs in the areas of nuclear safety. For example, one of the major R&D scopes in the fiscal year 2009 was in the establishment of domestic nuclear safety and regulatory technologies, which includes the development of the independent verification technology for nuclear safety analysis, the development of the regulatory tools and guidelines for regulations on the nuclear and radiation safety, the establishment of the accreditation platform for the nuclear grade industrial technologies, etc. In addition to the regular R&D programs, the INER can also organize a special technical team or establish a project with the purpose of solving a particular safety issue when requested.

Table 6.1 Comparison of the Performance Indicators of the TPC's Nuclear Power Units with the WANO Median in 2009

PI [(a)]	CS-1	CS-2	KS-1	KS-2	MS-1	MS-2	WANO-2009 (median)
1. UCF (%)	87.32	89.33	89.92	92.06	93.75	93.69	86.28 (3 yr av)*
2. UCLF (%)	2.05	4.65	0.87	0.32	0.07	0.02	1.48
3. FLR	2.05	0.06	0.97	0.33	0.07	0.02	0.87
4. UA7	0.00	0.00	0.88	0.00	0.00	0.00	0 (world av)*
5. SSP (%)							
BWR(HPSI)	0.56	0.04	0.00	0.02			0.30 (3 yr av)
(RHR)	0.16	0.34	0.05	0.02			0.10 (3 yr av)
PWR(HPSI)					0.00	0.00	0.10 (3 yr av)
(AFS)					0.01	0.00	0.10 (3 yr av)
EPS	0.00	0.00	0.02	0.02	0.01	0.69	0.40 (3 yr av)
6. FR							
BWR(μCi-sec)	1.00	1.00	1.00	1.00			2.00
PWR(μCi-sec)					1.0E-6	1.5E-4	6.2E-6
7. CP							
BWR	1.00	1.00	1.00	1.00			1.00
PWR					1.00	1.00	1.00
8. CRE (man-Ci/unit)							
BWR	1.07	1.07	1.68	1.68			1.20 (3 yr av)
PWR					0.73	0.73	0.61 (3 yr av)
9. ISAR ($2x10^5$man-hrs)	0.00	0.00	0.00	0.00	0.00	0.00	0.09
10.CISAR ($2x10^5$man-hrs)	0.00	0.00	0.00	0.00	0.00	0.00	0.18
11.GRLF (%)	0.00	0.00	0.00	0.00	0.00	0.00	0.00

* '3 yr av' and 'world av' stand for '3 years-average' and 'world average', respectively.

(a)The abbreviations in the PIs stand for the following:

1. UCF: Unit Capacity Factor
2. UCLF: Unplanned Capacity Loss Factor
3. FLR: Forced Loss Rate
4. UA 7: Unplanned Automatic Scrams per 7,000 Hours Critical
5. SSP: Safety System Performance
   HPSI: High Pressure Safety Injection System, AFS: Auxiliary Feedwater System
   RHR: Residual Heat Removal System, EPS: Emergency Power Supply
6. FR: Fuel Reliability
7. CP: Chemistry Performance
8. CRE: Collective Radiation Exposure
9. ISAR: Industrial Safety Accident Rate
10. CISAR: Contractor Industrial Safety Accident Rate
11. GRLF: Grid-Related Loss Factor

Table 6.2 Basic Data of the Nuclear Power Units in Taiwan

	Chinshan		Kuosheng		Maanshan		Lungmen	
Unit	Unit 1	Unit 2	Unit 1	Unit 2	Unit 1	Unit 2	Unit 1	Unit 2
Construction Permit	Dec. 15, 1971	Dec. 4, 1972	Aug. 19, 1975	Aug. 19, 1975	Apr. 1, 1978	Apr. 1, 1978	Mar. 17, 1999	Mar. 17, 1999
Commercial Operation	Dec. 6, 1978	July 16, 1979	Dec. 28, 1981	Mar. 15, 1983	July 27, 1984	May 18, 1985	–	–
Reactor Type (Vendor)	BWR-4 (GE)		BWR-6 (GE)		3-loop PWR (W)		ABWR (GE)	
Rated Power: Thermal* Electrical*	1,775 MWt 635 MWe		2894 MWt 985 MWe		2785 MWt 951 MWe		3926 MWt 1350 MWe	
T-G Vendor	W		W		GE		Mitshubishi	
A-E	Ebasco		Bechtel		Bechtel		S & W	
Containment	Mark I		Mark III		Large, Dry Post-Tensioned Reinforced Concrete		Reinforced Concrete Containment Vessel	

* Before power up-rated by MUR.

Table 6.3 TPC's Safety-Related Performance Indicators during 1991 – 2009 *

Year	No. of scrams per unit	No. of abnormal events per unit	No. of violations more serious than 4^{th} degree per unit **	Unit capacity factor, (%) (weighted average)
1991	2.33	34.8	7.8	78.32
1992	1.16	23.3	2.2	74.90
Nuclear SC cultivating program began in early 1993				
1993	2.16	22.3	5.0	76.24
1994	1.5	16.5	2.3	77.38
1995	1.83	13.1	3.2	78.37
1996	0.83	13.0	2.8	83.63
1997	0.83	7.6	2.3	80.49
Nuclear SC enhancing period began in early 1998				
1998	2	11.3	1.7	81.81
1999	2.67	10.3	2.2	85.25
2000	0.33	4.67	0.7	85.21
2001	0.67	7.67	0.5	78.75
2002	0.67	4.17	0.3	87.78
2003	0.33	4.0	0.5	86.3
2004	0.17	1.17	0	87.4
2005	0.5	1.5	0.67	88.71
2006	0.33	2.0	0.33	88.48
2007	0.33	2.33	0.17	89.96
2008	0.33	2.17	0	90.36
2009	0.17	1.33	0.33	92.25

* All data shown are the average of the 6 operating nuclear units' values.

** 1^{st} degree of violation is the most serious.

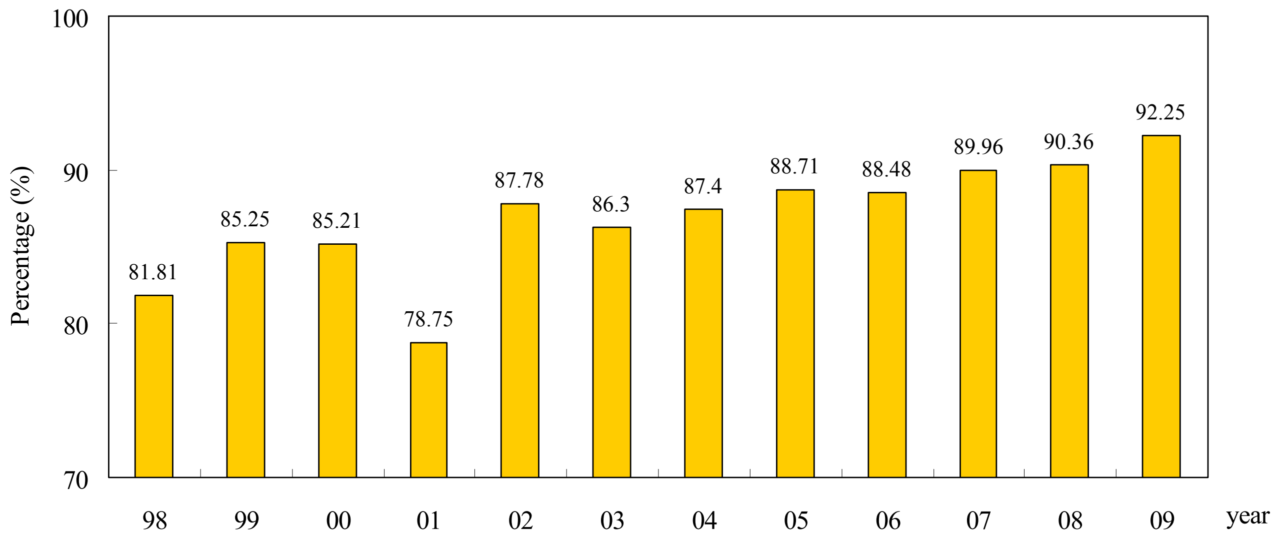


Figure 6.1 Annual Capacity Factor of the Operating NPPs in Taiwan

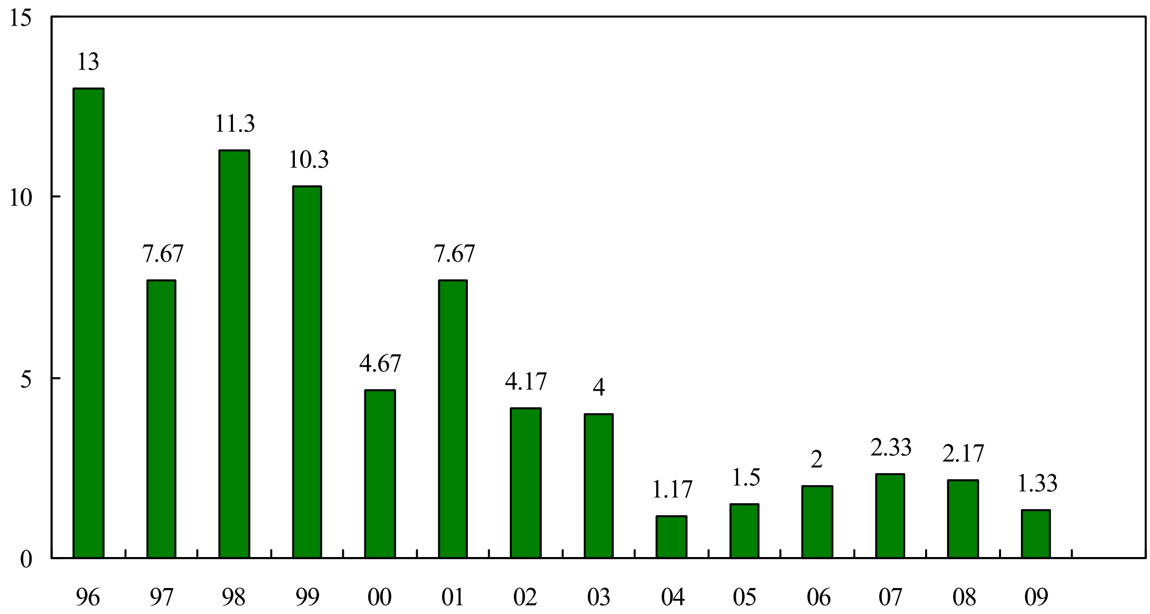


Figure 6.2 Average Number of RERs for the Operating NPPs in Taiwan

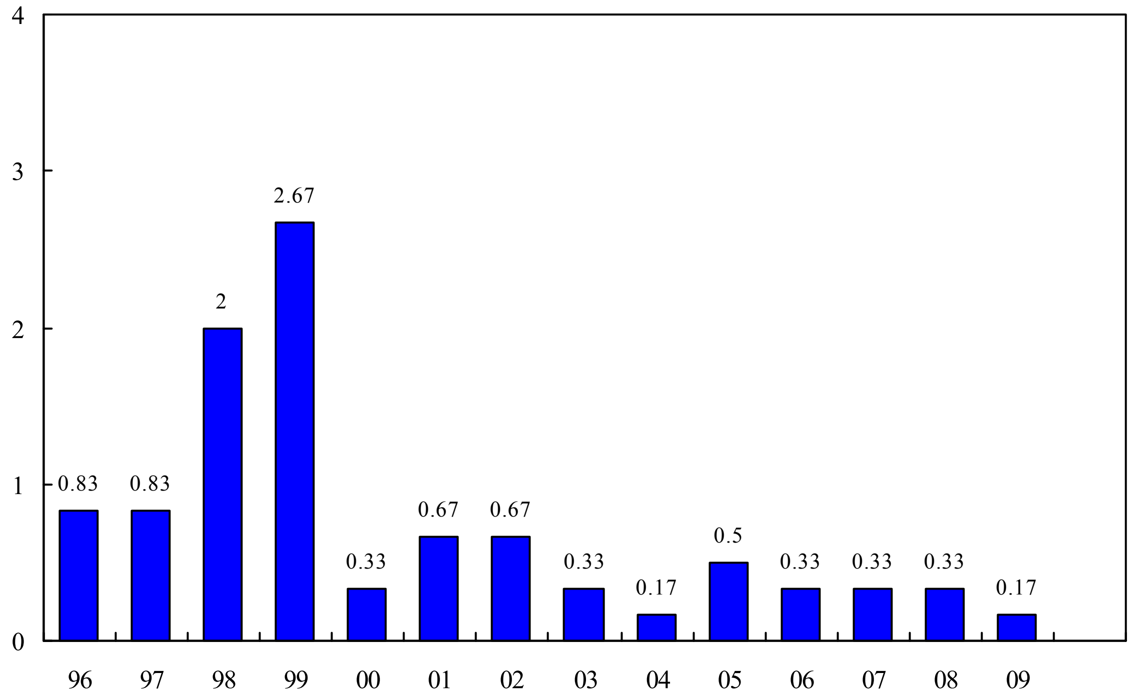


Figure 6.3 Average Number of Scrams for the Operating NPPs in Taiwan

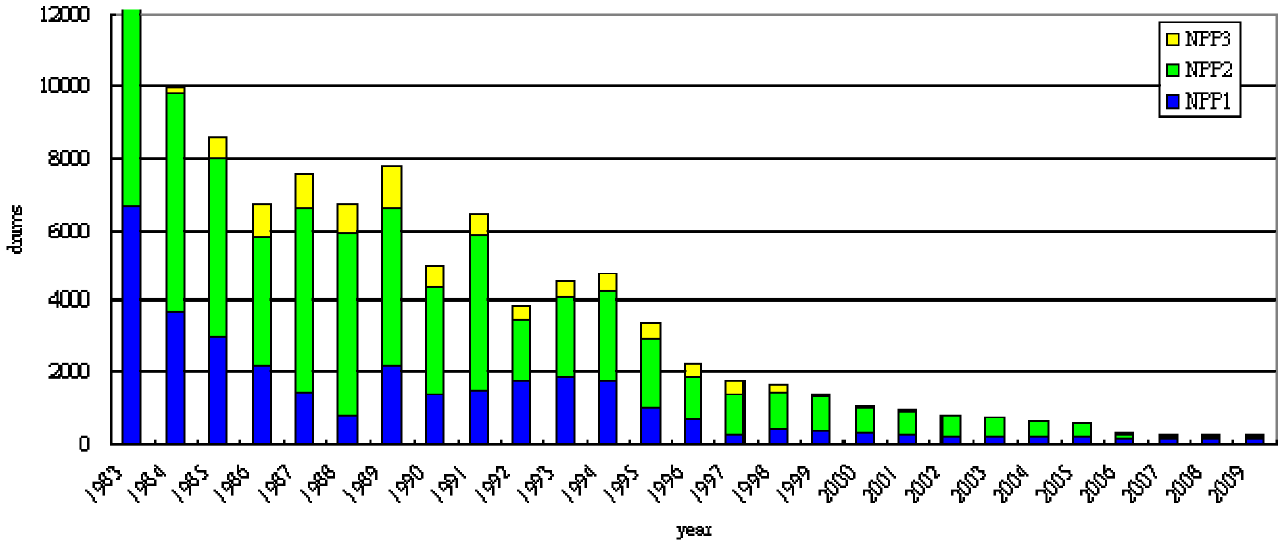


Figure 6.4 Number of Drums of Solidified LLRW from the Operating NPPs in Taiwan

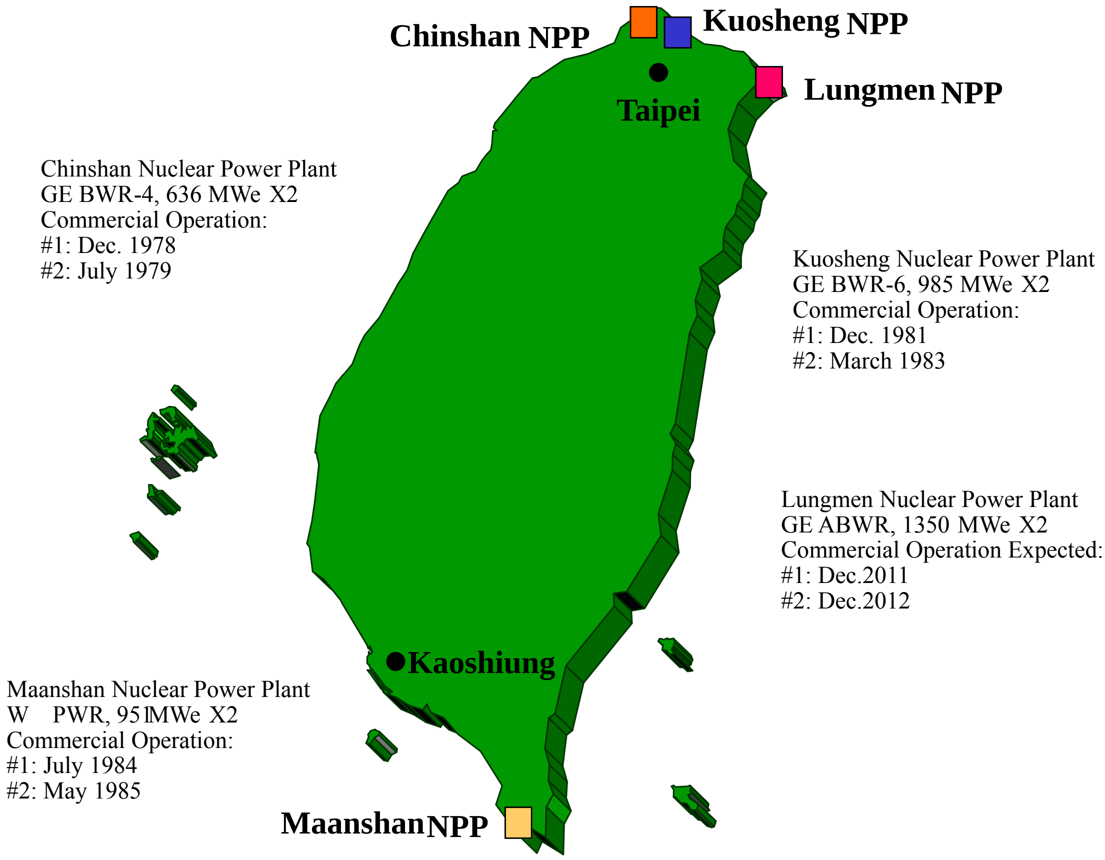


Figure 6.5 Locations of the Nuclear Power Plants in Taiwan

ARTICLE 7. LEGISLATIVE AND REGULATORY FRAMEWORK

1. Each Contracting Party shall establish and maintain a legislative and regulatory framework to govern the safety of nuclear installations.

2. The legislative and regulatory framework shall provide for:

(i) the establishment of applicable national safety requirements and regulations;

(ii) a system of licensing with regard to nuclear installations and the prohibition of the operation of a nuclear installation without a license;

(iii) a system of regulatory inspection and assessment of nuclear installations to ascertain compliance with applicable regulations and the terms of licenses;

(iv) the enforcement of applicable regulations and of the terms of licenses, including suspension, modification or revocation.

7.1 Legislative and Regulatory Framework in the ROC

The Atomic Energy Act is the basic law that provides the legislative and regulatory framework for the utilization of nuclear energy in the Republic of China (ROC). This Act was passed by the Legislative Yuan, which is the parliament equivalent of this nation, and signed by the President in 1968, with a later modification in 1971. The objectives of the Atomic Energy Act are to promote the research and development (R&D) of the atomic energy science and technology, and also the resource development and peaceful utilization of the atomic energy. Article 3 of the Atomic Energy Act stipulates that the "Responsible Agency" for the Act shall be the Atomic Energy Council (AEC). The AEC of the Republic of China in Taiwan was founded in 1955 at the ministerial level of the Executive Yuan, which is the Cabinet of this country. The principal mission of the AEC is described in Article 8 of this report.

To assure the principle of “administration by law”, the Act of the Administrative Procedure was put into effect in 1999 in order to strengthen the protection of human rights in the course of litigation and to increase the administrative efficiency. Accordingly, previous governmental regulations without approval or authorization by the Legislative Yuan will lose their legality after a buffer period being set at 2 years. In response to the promulgation of the Act of the Administrative Procedure, many major modifications of the Atomic Energy Act as well as other regulations and guidelines have been proposed, approved by the Legislative Yuan if necessary, and then put into actions.

The related draft Acts and Laws prepared by AEC will be submitted to Executive Yuan for review first, and then sent to Legislative Yuan for deliberation. The Education and Culture Committee of the Legislative Yuan is responsible for the bills of AEC, and thus AEC will report and make detailed description of the proposed draft Acts and Laws to the committee members. After necessary modifications are made and the draft Acts and Laws are approved by the committee, the formal approval of the Acts and Laws will be made in the Legislative Yuan's Conference. The new Acts and Laws will be promulgated by the President then and become effective. As the Regulations are authorized by the related Acts and Laws, AEC may prepare or modify the Regulations based on actual regulatory requirement. The Regulations will be reviewed by the Advisory Committee on Nuclear legislation and go through AEC's internal approval procedure for the promulgation.

In the following sections, current laws, regulations and requirements will be described. Selected contents of these new laws will be provided as supplemental information.

7.2 Nuclear Regulatory Laws, Regulations and Requirements

This section describes the seven basic laws for regulation of activities related to nuclear energy and radiations, the Enforcement Rules associated with these basic laws, and the regulations.

7.2.1 Basic Laws

The seven basic laws for nuclear regulation in this country are the Atomic Energy Act, the Nuclear Reactor Facilities Regulation Act, the Ionizing Radiation Protection Act, the Nuclear Emergency Response Act, the Nuclear Materials and Radioactive Waste Management Act, the Nuclear Damage Compensation Law, and the Act on Sites for Establishment of Low Level Radioactive Waste Final Disposal Facility. The Atomic Energy Act was passed by the Legislative Yuan and later first promulgated by the President in 1968 and amended in 1971, while the Nuclear Damage Compensation Law was first promulgated in 1971 and amended twice with the latest amendment promulgated in 1997. The remaining Laws except the Act on Sites for Establishment of Low Level Radioactive Waste Final Disposal Facility, which was promulgated in 2006, were all promulgated during the period of 2002 to 2003.

(1) Atomic Energy Act

The regulations on nuclear installations are governed by the Atomic Energy Act. This Act is composed of 34 articles, which are grouped into 9 chapters as follows:

- General Principles,
- Responsible Agency for the Atomic Energy,
- Research and Development of the Atomic Energy Science and Technology,
- Development and Utilization of the Atomic Energy Resources,
- Regulatory Control of Nuclear Materials, Fuels, and Reactors,
- Radiation Protection,
- Encouragement, Patent and Compensation,
- Penal Provisions, and
- Supplementary Provisions.

(2) Nuclear Reactor Facilities Regulation Act

The Nuclear Reactor Facilities Regulation Act, promulgated in January 2003, is to regulate nuclear facilities in order to protect the public heath and safety. It is composed of 44 articles grouped into 5 chapters as follows:

- General Principles,
- Regulations of Construction and Operation,
- Regulations of Off-Commissioning and Decommissioning,
- Penal Provisions, and
- Supplementary Provisions.

(3) Ionizing Radiation Protection Act

The regulations on radiation protection are governed by the Ionizing Radiation Protection Act promulgated in January 2002. This Act is composed of 57 articles that are grouped into 5 chapters as follows:

- General Principles,
- Radiation Safety and Protection,
- Management of Material, Equipment or Practice,
- Penal Provisions, and
- Supplementary Provisions.

A description of the evolution of the Ionizing Radiation Protection Act is given in Article 15 of this report.

(4) Nuclear Emergency Response Act

The Nuclear Emergency Response Act was promulgated in December 2003 to strengthen the emergency response system for nuclear accident, and to make an effort to consolidate the emergency response function so as to ensure the safety and health of the public and to protect their properties. This Act is composed of 45 articles that are grouped into 7 chapters as follows:

- General Principles,
- Organizations and Responsibilities,
- Preparedness Measures,
- Response Measures,
- Recovery Measures,

- Penal Provisions, and
- Supplementary Provisions.

(5) Nuclear Materials and Radioactive Waste Management Act

The Nuclear Materials and Radioactive Waste Management Act, promulgated in December 2002, is enacted to administrate the radioactive material, to prevent radioactive hazard and to protect the public health and safety. This Act is composed of 51 articles that are grouped into 5 chapters as follows:

- General Principles,
- Administration of Nuclear Materials and Nuclear Fuel,
- Administration of Radioactive Wastes,
- Penal Provisions, and
- Supplementary Provisions.

(6) Nuclear Damage Compensation Law

The compensation for nuclear damages resulting from the peaceful uses of atomic energy is governed by the Nuclear Damage Compensation Law. This Law was promulgated in 1971 and amended twice in 1977 and 1997. It is composed of 37 articles that are grouped into 5 chapters as follows:

- General Provisions,
- Liabilities for Damage Compensation,
- Maximum Amount and Guarantee for Liabilities,
- Right to Claim for Damage Compensation, and
- Supplementary Provisions.

A more detailed description of the Nuclear Damage Compensation Law is given in Article 11 and Article 16 of this report.

(7) Act on Sites for Establishment of Low Level Radioactive Waste Final Disposal Facility

The Act on Sites for Establishment of Low Level Radioactive Waste Final Disposal Facility was promulgated in May 2006. This Act is formulated for selecting the sites of final disposal facility of low level radioactive waste ("disposal facility" for short hereinafter) and conforming to the requirements on safety and environmental protection. This Act is composed of 21 articles that are grouped into eight chapters as follows:

- General Principles,
- The Competent and the Implementing Authority,

- Forbidden Area for the Final Disposal Facility,
- Procedure and Schedule for the Site Selection,
- Requirement of the Local Referendum,
- Requirement of the Environmental Impact Assessment,
- Amount and Distribution of the Feedback Subsidies, and
- Land Expropriation of the Final Disposal Facility.

7.2.2 Enforcement Rules

The seven basic laws mentioned above are laws with general and fundamental principles and concepts. Necessary enforcement rules for implementing these Laws have been provided for six of them to address the details. The status of these enforcement rules is shown below:

(1) Enforcement Rules for the Atomic Energy Act

Under Article 33 of the Atomic Energy Act, the Enforcement Rules for the Atomic Energy Act was promulgated by the AEC on December 7, 1976. This Enforcement Rules has been amended several times and the latest version of the amendment was promulgated on November 22, 2002.

(2) Enforcement Rules for the Implementation of Nuclear Reactor Facilities Regulation Act

Under Article 43 of the Nuclear Reactor Facilities Regulation Act, the Enforcement Rules for the Implementation of Nuclear Reactor Facilities Regulation Act was promulgated by the AEC on August 27, 2003.

(3) Enforcement Rules for the Ionizing Radiation Protection Act

Under Article 56 of the Ionizing Radiation Protection Act, the Enforcement Rules for the Ionizing Radiation Protection Act was promulgated on December 25, 2002 and amended on February 22, 2008.

(4) Enforcement Rules for the Implementation of the Nuclear Emergency Response Act

Under Article 44 of the Nuclear Emergency Response Act, the Enforcement Rules for the Nuclear Emergency Response Act was promulgated by the AEC on March 3, 2005.

(5) Enforcement Rules for the Nuclear Materials and Radioactive Waste Management Act

Under Article 50 of the Nuclear Materials and Radioactive Waste Management Act, the Enforcement Rules for the Nuclear Materials and Radioactive Waste Management Act was promulgated by the AEC on July 30, 2003 and amended twice on January 24, 2008 and on April 22, 2009.

(6) Enforcement Rules for Nuclear Damage Compensation Law

Under Article 36 of the Nuclear Damage Compensation Law, the Enforcement Rules for the Nuclear Damage Compensation Law was promulgated by the AEC on March 25, 1998.

7.2.3 Regulations

In addition to the basic laws described above, various regulations have been issued by the AEC. The Administrative Regulations, technical standards, and working notices are necessary for the effective implementation of these Acts or Law. A total of 14 Regulations for the Nuclear Reactor Facilities Regulation Act, 1 Regulation for the Nuclear Damage Compensation Law, 22 Regulations for the Ionizing Radiation Protection Act, 19 Regulations for the Nuclear Materials and Radioactive Waste Management Act, and 8 Regulations for the Nuclear Emergency Response Act are promulgated by the AEC as authorized by the corresponding Act. The titles of these regulations are listed in Tables 7.1 through 7.5.

7.3 Enforcement

The Nuclear Reactor Facilities Regulation Act and its enforcement rules mandate the AEC to conduct inspections, to revoke licenses, to issue orders, and to impose penalties, etc., whenever it is deemed necessary.

Articles 4 to 20 of this Act authorize the AEC for the licensing of the operation of nuclear reactor facility. Articles 21 to 28 authorize AEC to audit the off-commissioning and decommissioning of a nuclear reactor facility. Articles 29 to 40 authorize the AEC to impose civil or criminal penalties to the entity for the violation of the Act.

During the construction or operating period of a nuclear reactor facility, the AEC is authorized to ask the licensee for a timely improvement or some necessary measures under the following conditions: violation of regulation, concern for the public health and safety, or the relationships between different organizations were endangered. In the case of possible severe consequences, or the improvement remained incomplete, or necessary measures were not taken in the given time period, the AEC is authorized to suspend the ongoing activities or reactor operations, to revoke the operating license, or to ask the licensee to operate the reactor at reduced power. To impose the above mentioned penalty on the licensee, a written statement describing the decision should be delivered to the licensee. In case of emergency, a license suspension or revoking can be imposed with oral statements. However, the written statement should be delivered to the licensee within 7 days.

The way penalties are imposed on and the extent of penalties were also included in the Nuclear Reactor Facilities Regulation Act. The classification of penal provisions was defined there and the fines for civil penalties were raised significantly as compared to the previous ones defined in the Atomic Energy Act. In most situations, civil penalties and demands for timely improvements will be imposed upon first. Depending on licensee’s willingness for improvement, additional penalties will be further imposed upon if the licensee didn’t meet the requirements.

Table 7.1 Regulations related to the Nuclear Reactor Facilities Regulation Act

No.	Names of Related Regulations
1.	General Design Criteria for Nuclear Reactor Facilities
2.	Enforcement Rules for the Implementation of Nuclear Reactor Facilities Regulation Act
3.	Regulations on the Scope of Inspection and on the Certification of Authorized Inspection Agencies for Nuclear Reactor Facilities
4.	Regulations on the Dedication of Commercial Grade Items and Certification of Dedication Agency
5.	Regulations on the Restart of Nuclear Reactor Facilities after Operating Outage
6.	Regulations on Immediate Notification Requirements and Reportable Event Report for Nuclear Reactor Facilities
7.	Regulations on Nuclear Reactor Operators' Licenses
8.	Regulations on Quality Assurance Criteria for Nuclear Reactor Facilities
9.	Fees for Regulatory Services under the Nuclear Reactor Facilities Regulation Act
10	Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities
11	Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities
12.	Regulations on Medical Examination of Nuclear Reactor Operators
13	Regulations on the Review and Approval of Applications for Off-commissioning of Nuclear Reactor Facilities
14.	Regulations on Consignment Inspection for Nuclear Reactor Facilities

Table 7.2 Regulations related to the Ionizing Radiation Protection Act

No.	Names of Related Regulations
1.	Enforcement Rules for the Ionizing Radiation Protection Act
2.	Safety Standards for Protection against Ionizing Radiation
3.	Regulations for the Safe Transport of Radioactive Material
4.	Standards for Establishment of Radiation Protection Management Organizations and Radiation Protection Personnel
5.	Accreditation and Administrative Regulations for Personal Radiation Dose Evaluation Agencies
6.	Administrative Regulations for Radiation Protection Personnel
7.	Standards for Radiation-Caused Serious Environmental Contamination
8.	Standards for Limiting Radioactivity in Commodities
9.	Administrative Regulations for Radioactive Material and Equipment Capable of Producing Ionizing Radiation and Associated Practice
10.	Regulations for Administration of Radiation Protection Service Related Business
11.	Criteria for Management of Radiation Workplaces and Environmental Radiation Monitoring outside Them
12	Administrative Regulations for Operators of Radioactive Material or Equipment Capable of Producing Ionizing Radiation
13.	Administrative Regulations for the Operators of Production Facilities of Radioactive Material
14.	Classification of High Level Radiation Facilities and Administrative Regulations for Their Operators
15.	Radiation Protection and Control Regulations for Military Institutions
16.	Standards for Exemption from Regulation for Radiation Sources
17.	Standards for Collection of Regulation Fees for Ionizing Radiation Protection
18	Regulations on the Prevention and Management of Incidents of Radioactive Contaminated Buildings
19	Administrative Regulations on Establishment of Medical Exposure Quality Assurance Teams and Assignment of Specialists and Commissioning of Jobs to Relevant Organizations
20	Standards for Medical Exposure Quality Assurance
21	Regulations on the Management of Naturally Occurring Radioactive Materials
22	The Special Medical Surveillance Examination Items of Radiation Worker

Table 7.3 Regulations related to the Nuclear Emergency Response Act

No.	Names of Related Regulations
1.	Enforcement Rules for the Implementation of the Nuclear Emergency Response Act
2.	Regulations for Nuclear Emergency Classification, Response and Notification
3.	Regulations for Emergency Response of the Research Nuclear Reactor Facility
4.	Regulations for the Income and Expenditure, the Safekeeping and the utilization of the Nuclear Emergency Response Fund
5.	Emergency Response Basic Plan
6.	Nuclear Emergency Public Protective Action Guides
7.	Directions on the Operations of the National Nuclear Emergency Response Center
8.	Directions on the Operations of the Nuclear Emergency Radiation Monitoring and Dose Assessment Centers

Table 7.4 Regulations related to the Nuclear Damage Compensation Law

No.	Names of Related Regulations
1.	Enforcement Rules of Nuclear Damage Compensation Law

Table 7.5 Regulations related to the Nuclear Materials and Radioactive Waste Management Act

No.	Names of Related Regulations
1.	Enforcement Rules for the Nuclear Materials and Radioactive Waste Management Act
2.	Regulations for the Review and Approval of Applications for Construction License of Radioactive Wastes Treatment, Storage and Final Disposal Facilities
3.	Regulations on Final Disposal of Low Level Waste and Safety Management of the Facilities
4.	Regulations on Treatment and Storage of Radioactive Waste and Safety Management of the Facilities
5.	Regulations for the Review and Approval of Applications for Construction License of Nuclear Source Material and Nuclear Fuel Production and Storage Facilities
6.	Fees on Regulatory Services under the Nuclear Materials and Radioactive Waste Act
7.	Regulations for Awards for Radioactive Materials Research and Development
8.	Regulations on Clearance Level for Radioactive Waste Management
9.	Regulations for the Nuclear Fuels Operational Safety Management
10.	Regulations for the Nuclear Source Materials Operational Safety Management
11	Operational Regulations Governing Nuclear Safeguards
12	Regulations on the Nuclear Source Material Mine and Minerals
13	Regulations for the Permit of Import, Export, Transit, Transship, Transport, Discard, and Assignment of Low Level Radioactive Waste
14	Regulations for the Review and Approval of Applications for Decommissioning Permit of Nuclear Reactor Facilities
15	Regulations on the Final Disposal of High Level Radioactive Waste and Safety Management of the Facilities
16	Regulation for Entrusting Inspection on Radioactive Waste Facilities
17	Regulations on the Range and Criteria of the Forbidden Areas of Low Level Radioactive Waste Final Disposal Facility
18	Regulations on Naturally Occurring Radioactive Material Waste Management
19	Regulations on Qualification of the Operating Personnel of Radioactive Waste Treatment Facilities

ARTICLE 8. REGULATORY BODY

1. **Each Contracting Party shall establish or designate a regulatory body entrusted with the implementation of the legislative and regulatory framework referred to in Article 7, and provided with adequate authority, competence, and financial and human resources to fulfill its assigned responsibilities.**

2. **Each contracting Party shall take the appropriate steps to ensure an effective separation between the functions of the regulatory body and those of any other body or organization concerned with the promotion or utilization of nuclear energy.**

8.1 Nuclear Regulatory Body

The AEC was founded in 1955 at the ministerial level under the Executive Yuan as a Cabinet member. The Atomic Energy Act was passed in 1968 to give AEC the exclusive regulatory authority to ensure that the civilian use of nuclear energy and materials are conducted with proper regard for the public health and safety.

8.1.1 Mandate

The AEC's principal mission in the initial founding period was limited to the management of international affairs concerning the atomic energy and the promotion of the peaceful applications of the atomic energy in this country.

In more than 30 years since 1978, the first, second and third nuclear power plants were successively connected to the power grid. At the same time, radioisotope applications in the medical, agricultural, industrial and research fields were expanding in great pace. Therefore, the most important tasks for the AEC have been shifted to the nuclear reactor safety regulation, radiation protection, radioactive waste administration, and regulatory researches.

The AEC, in the implementation of the aforementioned regulatory tasks and R&D works, adheres to the following principles: safety first, reasonable control, and convenience to the people. Safety will remain to be the highest priority in the process of technical developments for nuclear applications.

8.1.2 Authority and Responsibilities

The AEC's mission is to ensure that the civilian use of the nuclear energy and materials, including the radioactive materials, is conducted with proper regards for the public health and safety, and to protect the environment from the radiation released out of nuclear reactors, radioactive materials, and nuclear waste facilities. The basic charter for these regulatory responsibilities is the Atomic Energy Act of 1968 (as amended in 1971), through which the Legislative Yuan (i.e. the Parliament) created a national policy of developing the peaceful uses of atomic energy. That statute has been amended or proposed to be amended over the years to cope with technology developments and worldwide changing perceptions of regulatory needs, such as the more specialized statutes prescribing the AEC's duties with regard to low-level and high-level radioactive wastes, decommissioning, safety reviews, and import/export control.

The AEC has been given the authority to regulate either naturally occurring or man-made radioactive materials, in addition to the nuclear materials such as the uranium and thorium. The AEC also has been given the authority to regulate the machine-produced radiation, such as the emissions from the X-ray units or linear accelerators.

The AEC's licensing authority also extends to the military sector which uses radioisotopes or machine-produced radiation in the respective hospitals, the academies and research laboratories, and the radiopharmaceutics in the hospitals.

The AEC's responsibilities include both nuclear safety and safeguards through which the agency ensures the security of machines and materials against radiological sabotage, lost, thefts and misuse.

8.1.3 Structure of the Regulatory Body

This section explains the structure of the AEC. It covers the Council itself, various offices, affiliated agencies and advisory committees. The organization chart of AEC is shown in Figure 8.1.

8.1.3.1 Atomic Energy Council

The Atomic Energy Council consists of more than 10 commissioners, mostly representatives of relevant ministries or agencies within the Executive Yuan and experts from the academia. There are four technical departments and four administrative units within the AEC headquarters in addition to eight advisory committees on nuclear policy and safety. Besides, under the AEC's supervision, there are three affiliated agencies, namely, the Institute of Nuclear Energy Research (INER), the Fuel Cycle and Materials Administration (FCMA) and the Radiation Monitoring Center (RMC).

The Minister of the AEC presides over the Council with the assistance of two Deputy Ministers and the Chief Executive Secretary to oversee the Council affairs and supervises the affiliated agencies.

The technical departments and administrative units, working directly under the Council's administration, include four technical units such as Department of Planning, Department of Nuclear Regulation, Department of Radiation Protection and Department of Nuclear Technology; and four administrative units such as Department of General Administration, Office of Personnel, Office of Accounting and Office of Civil Service Ethics. There is also a mission-oriented unit, the Office of Congressional Liaison, which is separated from these departments.

The eight advisory committees are: (1) the Advisory Committee on Nuclear Facility Safety, (2) the Advisory Committee on Ionizing Radiation Safety, (3) the Supervising Committee on Nuclear Safety of the Lungmen Station, (4) the Advisory Committee on Nuclear Accident Investigation and Evaluation, (5) the Advisory Committee on Nuclear Legislation, (6) the Advisory Committee on Radioactive Materials Safety, (7) the Evaluation Committee on Research and Development Achievement, and (8) the Advisory Committee on Handling of State Compensation Cases.

The AEC headquarters employs approximately 163 personnel with the FY2009 budget of NT$337 millions (not including the budget of the three affiliated agencies) as shown in Tables 8.1.

8.1.3.2 Offices of the Atomic Energy Council

The responsibilities of the various organizations within the AEC or under its supervision are described below.

(1) Department of Planning

The primary responsibilities of the Department of Planning include control and oversight of the major policy implementation, planning, integration and assessments of the R&D projects, development and review of regulations on atomic energy related subjects, nuclear safeguards and international cooperation. The major tasks are:

- Planning and development of policies on nuclear science and technology, as well as the planning, integration, control and assessments of related programs, implementation activities and R&D projects,
- Communication and cooperation with domestic and overseas organizations on nuclear science and technology,
- Coordination, implementation, supervision and assessment of nuclear safeguards activities,
- Planning of human resources on nuclear science and technology, including overseas training programs,
- Planning and coordination of educational programs on nuclear science,
- Transactions of patents on nuclear science and technology,
- Assessments, compensations, and insurance policies on nuclear accidents, and
- Other planning assignments.

(2) Department of Nuclear Regulation

The primary responsibilities for the Department of Nuclear Regulation (DNR) are to implement safety regulations for the design, construction and operation of nuclear reactors and other nuclear facilities. The major tasks are:

- Review and inspection of the design, construction, transport, operation, and maintenance of nuclear reactors,
- Review of the safety analysis for the reactor design, construction and operation,
- Issuance of nuclear reactor licenses,
- Review of the reactor system design modifications, equipment changes, and revision of technical specifications,

- Issuance of licenses to nuclear reactor operators,
- Review of the nuclear fuel reload safety analysis,
- Investigation and evaluation of the abnormal reactor events,
- Analysis and evaluation of the nuclear power plant operation,
- Regulation of the nuclear fuel usage, and
- Other regulatory tasks related to the nuclear energy as required by the AEC.

(3) Department of Radiation Protection

The primary responsibility for the Department of Radiation Protection (DRP) is to ensure the radiation safety of nuclear facilities, environment, and the medical and non-medical applications of radioactive materials and equipment capable of producing ionizing radiation. The major tasks are:

- Radiation protection and environmental radiation control of nuclear reactors,
- Radiation protection and environmental radiation control of radioactive waste storage and disposal sites,
- License issuance for radioactive materials and equipment capable of producing ionizing radiation and the related operating personnel,
- Regulation of radiation safety for radioactive materials and equipment capable of producing ionizing radiation,
- Inspection of ionizing radiation site and its environmental radiation,
- Regulation of the safe transport of radioactive material,
- Development of regulations for radiation safety,
- Issuance of radiation detection and measurement documents,
- Regulation and assessment of nationwide radiation dose and background radiation,
- Review of radiation safety assessment reports,
- Evaluation of the proficiency in radiation protection of radiation workers,
- Handling and investigation of radiation incidents, and
- Other assigned responsibilities.

(4) Department of Nuclear Technology

The primary responsibilities for the Department of Nuclear Technology (DNT) are the evaluation and analysis of the nuclear reactor performance, regulation and inspection on the implementations of the Nuclear Emergency Response Act, secretariat for the National Nuclear Emergency Response Center, and nuclear information management. The major tasks are:

- Development of regulations for nuclear emergency response,
- Planning and evaluation of nuclear emergency preparedness and exercises,
- Operation and maintenance of National Nuclear Emergency Response Center,
- Coordination and integration of nuclear and radiological emergency,
- Operation and maintenance of Nuclear Safety Duty Center,
- Management and security of nuclear information, and
- Other assigned responsibilities.

(5) Office of Congressional Liaison

The Office of Congressional Liaison is responsible for the liaison with the Legislative Yuan（Parliament）and the Control Yuan, the latter was enacted with the power of impeachment, censure and audit authority by the Constitution.

The primary responsibilities for the Office of Congressional Liaison are to strengthen the liaison between the AEC and the congressional organizations, and at the same time to enhance the understanding of the AEC activities by the congressional members so as to smooth the AEC's administrative works. This office monitors legislative proposals, bills, and hearings, and informs the AEC of the views of the Parliament on the AEC policies, plans, and activities.

The major tasks of the Office of Congressional Liaison are as follows:

- To conduct the liaison between the congressional organizations（including the Legislative Yuan and the Control Yuan）and the AEC,
- To conduct the liaison and communication with the administrative units of the congressional organizations,
- To communicate with the congressional members, including their assistants and staff, about the AEC's activities,
- To communicate with the congressional liaison offices of other ministries under the Executive Yuan,
- To collect the information about the interpellation of the congressional members and the related concerns, and
- To respond the related matters requested by the congressional members.

(6) Administrative Units

There are four administrative units within the AEC, which are: Department of General Administration, Office of Personnel, Office of Accounting and Office of Civil Service Ethics. The Department of General Administration is responsible for documentation and property management as well as the administrative support to all other departments and offices. The Office of Personnel and Office of Accounting are responsible for the general administrative matters related. The organization of Civil Service Ethics is responsible for supervising the ethics of the government employees across the nation, executing the anti-corruption work, protecting official secrets, and preventing the impairment and sabotage of the public agencies. Thus, the Office of Civil Service Ethics of the AEC is responsible for executing the similar jobs within the AEC.

8.1.3.3 Affiliated Agencies

(1) Institute of Nuclear Energy Research

The Institute of Nuclear Energy Research (INER) was established in 1968 under the Atomic Energy Council. INER is now under transformation in order to fulfill to the long-term development goal as well as to comply with the government re-structure policy. This transformation allows INER utilizing broadly her nuclear energy technology to the environmental protection and civilian applications. Hence, INER has established three research centers which are: Nuclear Safety Technology Center, Environmental and Energy Technology Center, and Radiation Application Technology Center. These centers operate with 11 existing functional divisions in a matrix manner.

As a national laboratory, the INER's missions are:

- To establish the advanced nuclear R&D capabilities,
- To utilize her technologies to the domestic industry, and
- To benefit the human living.

INER employs approximately 936 personnel including researchers, technicians, and supporting staff in 2009. The researchers are all professionals with graduate degrees including 117 with doctoral and about 174 with Master degree. The FY2009 budget of INER was NT$ 2,939 millions.

The core technology and the major research areas of the three research centers are as follows:

(a) Nuclear Safety Technology Center

- Nuclear Safety Regulation and Evaluation,
- Operational Safety of Nuclear Facilities,
- Radiation Protection and Emergency Preparedness,
- Commercial Grade Item Dedication and Inspector Qualification,
- Decommissioning and Reutilization, and

- Waste Treatment and Disposal.

(b) Environmental and Energy Technology Center

- Plasma Engineering,
- Clean Process,
- New Energy,
- Biomass-energy, and
- Renewable Energy.

(c) Radiation Application Technology Center

- Technology Development of Radioisotope on Medicine,
- Research and Development of Radiopharmaceutics, and
- Research and Application of Radiation Biology.

(2) Fuel Cycle and Materials Administration

The Fuel Cycle and Materials Administration (FCMA), an subsidiary agency under the Atomic Energy Council, was originally established under the name of the Radioactive Waste Administration (RWA) in 1980 to regulate the radioactive waste from the nuclear power stations as well as other small producers (i.e., from research, medical, industrial, and other facilities). The RWA was also entrusted to operate the first low level radioactive waste storage facility located in Lan-Yu, a small offshore island of Taiwan. That facility was designed to receive all the solidified low-level radioactive wastes generated in the country, especially that from the operation of nuclear power plants of the Taiwan Power Company (TPC). It was transferred to the TPC in July 1990.

The RWA changed its name to FCMA in January 1996. Its roles as a radioactive waste regulator are clearly separated from the producer (TPC) and the Administration's authority is enhanced. In addition to the licensing of various waste treatment and storage facilities as well as the disposal sites, FCMA also makes lots of efforts for the regulation of wastes from small producers, technologically enhanced naturally-occurring radioactive material and nuclear source materials.

In summary, FCMA is responsible for the safety regulation of the treatment, storage, transport and final disposal of radwaste, and the import, export, storage, and transfer of nuclear source materials and nuclear fuels. Its major tasks include:

- Licensing and certification of facilities associated with the design, construction, operation and decommissioning/closure of installations for radwaste treatment, storage, and disposal,
- Regulation and inspection for the treatment, storage, transport, disposal, import and export of radwaste,

- Regulation and inspection for the import, export, storage, utilization, discard, and transfer of nuclear source materials,

- Regulation and inspection for the import, export, storage, discard, and transfer of nuclear fuels,

- Development of regulations and technical standards for the radioactive material,

- International cooperation with respect to radioactive material regulation,

- Education and public communication with respect to radioactive material regulation,

- Policy and strategy development for the management of radioactive material,

- Promotion of the research and development on radwaste management technologies,

- Review, regulation and inspection of the nuclear reactor decommissioning,

- Issuance of nuclear fuel licenses,

- Review, regulation and inspection of the design, construction, transfer, dismantling and disposal of the nuclear fuel production facilities, and

- Other matters related to radioactive material management.

(3) Radiation Monitoring Center

The Radiation Monitoring Center (RMC) was previously named the Taiwan Radiation Monitoring Station (TRMS), which was established in 1974 as an affiliated agency under the Atomic Energy Council to carry out the monitoring of natural ionizing radiation in the environment and man-made ionizing radiation in the vicinity of nuclear power stations, nuclear research reactors, and radioactive waste facilities. The TRMC has been renamed as the Radiation Monitoring Center since July 1996.

The major tasks of this Center are:

- Formulation and implementation of the environmental radiation measurement plans,

- Measurement of the natural ionizing radiation in the environment,

- Measurement of the radioactive fallout,

- Measurement of the ionizing radiation in the vicinity of nuclear and other facilities with radioactive material,

- Measurement of the environmental radiation arising from treatment, storage, transport and final disposal of radioactive wastes,

- Radioactive contamination evaluation and measurement of radiation arising from accidents at nuclear facilities,
- Evaluation of the population radiation doses,
- Research and development of the radiation measurement technology,
- Providing information and advice to the public on environmental radioactivity, and
- Other matters related to environmental radiation monitoring.

8.1.3.4 Advisory Committees

There are eight technical or nuclear–related advisory committees within the AEC. Among them, seven are regularly operated, while the rest one is to be assembled only when needed. This section explains the structures and functions of these committees within the AEC.

(1) Advisory Committee on Nuclear Facility Safety

The Advisory Committee on Nuclear Facility Safety consists of 13 to 19 members with expertise in science and engineering. It gives advices to the AEC on the potential hazards of proposed or existing nuclear reactor facilities, the adequacy of proposed safety standards, and other matters on the Council's request. The statute requires that the Committee reviews certain types of applications, such as the construction licenses and the operating licenses for nuclear power reactors or research reactors. Before issuing a license like these, the AEC will listen to the advices from this Committee.

(2) Advisory Committee on Ionizing Radiation Safety

The Advisory Committee on Ionizing Radiation Safety consists of 13 to 19 members with expertise in science and engineering, including physicians, scientists and other representatives from the medical community. This Committee advises on radiation safety issues and gives expert opinions on the medical uses of radiation and radioisotopes. It also advises the AEC management, as required, on matters of radiation policy.

(3) Supervising Committee on Nuclear Safety of the Lungmen Station

The Supervising Committee on Nuclear Safety of the Lungmen Station consists of 12 to 14 members with expertise in science and engineering with emphasis on the public acceptance. The members also include representatives from the Taipei county and the two local townships. This Committee meets every 3 months. The responsibility of this committee includes supervising and checking of the engineering related safety and quality during construction and operation of Lungmen Station, together with the openness and transparency of information and other safety related issues of Lungmen Station.

(4) Advisory Committee on Nuclear Legislation

The Advisory Committee on Nuclear Legislation consists of 11 to 15 members from relevant agencies within the Executive Yuan and the private firms with expertise in law or nuclear disciplines. This Committee advises the AEC on proposed nuclear legislation before submitting to Executive Yuan or Legislative Yuan for approval, or important lawsuit involving the AEC, or petition from the citizen.

(5) Advisory Committee on Radioactive Materials Safety

The Advisory Committee on Radioactive Materials Safety consists of 11 to 15 members with expertise in science and engineering. This Committee advises on radioactive material safety issues, final disposal on radioactive waste, and other matters related to radwaste management. The committee will also advise on the review and safety regulation of major radioactive material facilities.

(6) Evaluation Committee on Research and Development Achievement

The Evaluation Committee on Research and Development Achievement consists of 11 to 17 members with expertise in related science and engineering research field. This Committee advises on the management, distribution and application of the research achievements for the projects sponsored by the AEC. The committee will also advise on the review and approval for the application of sole authorization on research results and products.

(7) Advisory Committee on Handling of State Compensation Cases

The Advisory Committee on Handling of State Compensation Cases consists of 6 to 8 members from scholars and the AEC senior staff, with the scholars as the majority. The responsibility of this committee includes negotiation as well as deliberating the state compensation cases and confirming the compensation authority of the state compensation and litigation on the state compensation cases.

Besides these 7 regularly operated advisory committees mentioned above, there is one more advisory committee chaired by AEC Minister, namely, the Advisory Committee on Nuclear Accident Investigation and Evaluation. In case there is a major nuclear accident this committee will be assembled.

(8) Advisory Committee on Nuclear Accident Investigation and Evaluation

The Advisory Committee on Nuclear Accident Investigation and Evaluation consists of 13 to 17 members, and will be setup after a major nuclear accident and damage claims from the public. The authority of this Committee includes: determination of the extent of a nuclear accident and investigation of the cause thereof, investigation and evaluation of the nuclear damage, recommendation on compensation, relief and rehabilitation measures for the nuclear accident, and recommendation on improvements of safety protections of nuclear installation. Reports of the aforementioned investigation, evaluation, and recommendation shall be prepared for public announcement. When the victims of a nuclear accident seek compensation by way of a judicial proceeding, the court may take into account these reports.

In case there is a major nuclear accident and recovery actions are needed after the accident, the Nuclear Emergency Recovery Committee will be assembled to perform the recovery measures. This committee is not an advisory type committee also chaired by

AEC Minister. The Nuclear Emergency Recovery Committee consists of 19 to 23 members from the AEC, Ministry of Interior, Ministry of National Defense, Ministry of Finance, Ministry of Economic Affairs, Ministry of Transportation and Communication, Directorate-General of Budget Accounting and Statistics, Government Information Office, Department of Health, Environmental Protection Administration, Financial Supervisory Commission, Council of Agriculture, National Communications Commission, the Local Government, the TPC and the relevant public neighboring the said NPP. The responsibility of this committee includes the following areas: to determine the recovery measures and supervise the implementation of these measures, to notify the relevant government agencies of various levels and the nuclear reactor facility licensee to implement relevant recovery measures, to coordinate the dispatch of manpower and resources for recovery, to announce orders for public protective actions during the recovery period, to issue press release for recovery, and to carry out any other recovery measure.

8.1.4 Financial and Human Resources of the Nuclear Regulatory Body

This section discusses the budget and funding of the AEC, its human resources, and financial management.

8.1.4.1 Financial Resources

Since the AEC and its three affiliated agencies, INER, FMCA, and RMC are government organizations, their major operational budgets all come from the government. The annual budget of the AEC together with its affiliated agencies will be applied through the Executive Yuan channel and has to be approved by the Legislative Yuan (LY) in advance before the fiscal year starts. The annual budgets in 2009 are 337, 2,939, 63 and 60 million NT Dollars for the AEC headquarters, INER, FCMA and RMC, respectively. The total annual budget for the AEC all together reaches 3,399 million NT Dollars in 2009. The Office of Accounting is responsible for the control of the annual budget thereafter of the AEC headquarters, INER, FCMA and RMC.

8.1.4.2 Fees Collected from the Licensees

Two types of fees are collected by the AEC from the licensees based on Fees for Regulatory Services under the Nuclear Reactor Facilities Regulation Act and Fees on Regulatory Services under the Nuclear Materials and Radioactive Waste Act. These fees will be reimbursed as part of the government income to fulfill the annual budget balance requirement. First, the license and safety review fees are established to recover the AEC's costs of providing individually identifiable services to the applicants or licensees. The services provided by the AEC are the review of the applications for the issuing of new licenses or approvals, amending or renewing licenses or approvals, and review of reload and topical reports. Secondly, the annual fees are collected to recover the generic (e.g., inspection, testing and research) and other regulatory costs that are not recovered through the license and safety review fees. The amounts of these two kinds of fees are based on the manpower requirement and their salaries approved by the Parliament.

8.1.4.3 Nuclear Emergency Response Fund

In order to implement the preparedness measures for the nuclear emergency response and to support the response operations during the occurrence or possible occurrence of an accident, based on the Nuclear Emergency Response Act article 43, a Nuclear Emergency Response Fund (NERF) has been raised. 24 million NT Dollars will be collected from each nuclear power plant every year by the AEC for the fund. The NERF management committee is responsible for the annual budget review and approval together with the performance review and audit of the NERF. This budget is required for the annual expense of the nuclear emergency exercise, routine operation and training of the National Nuclear Emergency Response Center, Radiation Monitoring and Dose Assessment Center, Nuclear Emergency Support Center and the Regional Nuclear Emergency Response Centers. It is estimated that 15 million NT Dollars of the budget will be saved each year, and it is expected to raise a total of 500 million NT Dollars in the NERF for the earlier on response expenses of a nuclear accident.

8.1.4.4 Human Resources

(1) Recruitment and Hiring Process

The number of staff in the AEC headquarters, INER, FCMA and RMC are 163, 936, 36 and 31 respectively as in the fiscal year 2009. As all the staff are public officials and specific knowledge or technology are required in this field, the condition of the staff retained in the AEC is relatively stable. The recruitment of new employees is mostly dependent on the availability of position provided by the government each year that takes into account of retirement or departure of current staff. The Civil Service Level 2 and 3 Senior Examination will be held usually once a year by the Ministry of Examination of the Examination Yuan based on the request of all the government organizations. The qualified personnel passed the above-mentioned examination in Nuclear Engineering, Radiation Safety or other Engineering Fields as required will be trained for one month and then dispatched to the AEC or its affiliated agencies.

Another channel of recruitment is through the cooperation with the Ministry of National Defense and the contract with the qualified graduate students with master or doctor degree. They will serve as the contracted staff in the AEC headquarters, INER, FCMA or RMC for 4 years instead of the mandatory military service of one year. The recruitment of these young generation staff who are well educated in related technical fields has proved to be a very effective channel for hiring qualified new employees.

(2) Training and Inspector Qualification

The new employees of the AEC headquarters will receive on-the-job training in related sections of the department he served. They will also be requested to get familiar with the regulations and guides for implementation of the inspection works. The senior staff are required to help for the training of new employees about the regulatory requirements and the inspection skills. Seminars, technical discussions, AEC web information and walk through of the NPPs also provide effective ways to train the new employees to catch all the up-to-date information and thus help improve their inspection capability.

The nuclear power plant inspector qualification system established in the AEC headquarters has proved to be an effective way to continuously enhance the knowledge and skills for staff in the AEC. To become a NPP inspector, a new employee has to perform the self studies, on-the-job training, and finish several basic courses and NPP's system courses. Then, after passing all the required tests and examinations, he or she may obtain an inspector certificate in order to formally serve as a qualified NPP inspector. An inspector may apply for a senior NPP inspector if he or she has served as an inspector for at least 6 years. In doing this, the applicant has to perform the self studies and on-the-job training with advanced courses again, and may then obtain a senior inspector certificate in order to formally serve as a qualified senior NPP inspector after passing the required examinations and evaluations. The effective periods for both inspectors and senior inspectors are 6 years. Both inspectors and senior inspectors are required to take at least 30 hours of training courses every two years to keep their certificates valid.

8.1.5 Position of the AEC in the Government

This section explains the relationship of the AEC to the Executive Yuan (i.e. the Cabinet), the local counties, and the Legislative Yuan (i.e. the Parliament).

8.1.5.1 Executive Yuan

This section explains the relationship between the AEC and the various related ministries of the Executive Yuan. These ministries (or their branches) are: the Ministry of Economic Affairs (MOEA), the Environmental Protection Administration (EPA), the National Fire Agency (NFA) of the Ministry of the Interior (MOI), the Department of Health (DOH), the Council of Labor Affairs (CLA), the Ministry of Foreign Affairs (MOFA), and the Directorate General of Budget, Accounting and Statistics (DGBAS).

(1) Ministry of Economic Affairs

The Ministry of Economic Affairs (MOEA) is in charge of the matters regarding national economic administration and construction. Its major functions encompass the industrial development, international trade, intellectual property, standard, metrology and inspection, investment and technology transfer, guidance and assistance for small and medium enterprises, technology development, national corporation and natural resources (energy, water and geology), etc. For details, please refer to Subsection 8.2.1.

The TPC, established on May 1, 1946, is one of the State-Owned Corporations supervised by the MOEA. The number of TPC's employees is approximately 26,921 and its assets worth 1,590 billions NT Dollars in August 2010. As of December 2009, the total installed electric power capacity of the TPC reached 32,310 MWe (Nuclear: 5,144 MWe, 15.9%; Fossil: 22,487 MWe, 69.6%; Hydro: 4,499 MWe, 13.9%; Wind: 180 MWe, 0.6%). Its main mission is to maintain the stable supply of electric power with good quality and low price.

(2) Environmental Protection Administration

The Environmental Protection Administration (EPA), a ministry-level agency within the Executive Yuan, was founded in 1987 with the mission of protecting and improving the environment nationwide. Its major functions encompass air quality protection and noise control, water quality protection, waste management, environmental sanitation and toxic substance management and dispute resolution for environmental pollution. The affiliated organizations of the EPA are Bureau of Environmental Inspection, Environmental Analysis Laboratory and Environmental Professionals Training Center.

After passage of the Environmental Impact Assessment Act in December 1994, the review of the environmental impact assessment reports of a new nuclear power station or other nuclear facilities, e.g., the spent fuel interim storage facility (or the independent spent fuel storage installation) and the low-level radioactive waste repository, has been transferred from the AEC to the EPA.

(3) National Fire Agency, Ministry of the Interior

The Disaster Rescue Command Center was formally established in July 2000 after the Chi-Chi Earthquake happened on September 21, 1999, under the National Fire Agency (NFA) of the Ministry of Interior. It has the leading responsibility for the emergency planning and response of all the major incidents including typhoon, flood, major fire, large explosion, airplane crash, etc. However, the AEC remains responsible for developing the emergency response plan on nuclear power stations. The NFA will assist the AEC in its licensing process especially on the offsite emergency planning and response documents review as well as the observing and evaluating emergency drills at the nuclear power stations.

(4) Department of Health

The Department of Health (DOH) has the major responsibility for health of the general public. It has the authority to regulate hospitals and medical related equipment and facilities. The AEC cooperates with the DOH to issue licenses for hospital workers operating the X-ray units or accelerators, or handling the radioisotopes or radiopharmaceuticals that release radiation.

(5) Council of Labor Affairs

The AEC closely monitors the legislations proposed by the Council of Labor Affairs (CLA), especially the Acts and regulations on occupational health and safety which may have impacts on radiation workers in the nuclear power stations and hospitals. For example, the Occupational Health and Safety Act specifies the physical examination requirements for radiation workers.

(6) Ministry of Foreign Affairs

The AEC works with the Ministry of Foreign Affairs (MOFA) on the following matters: the cooperation with international organizations such as the IAEA and the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (NEA/OECD), the policy development for nuclear issues that are under the AEC's purview, and the program planning and coordination of nuclear safety assistance to other countries.

(7) The Directorate-General of Budget, Accounting and Statistics

The Directorate-General of Budget, Accounting and Statistics (DGBAS) is the agency responsible for budget, accounting and statistics affairs within the central government as well as local county governments. The AEC submits the annual budget requests, including proposed personnel requirements, to this agency for approval.

For the relationship of the AEC with other ministries under a possible nuclear accident, such as Ministry of National Defense and Ministry of Interior, please refer to Subsection 16.2.3.

8.1.5.2 Local Counties

The Atomic Energy Act of 1968 chartered the AEC with preemptive authority over the health and safety regulation of the nuclear energy. As a result, the general rule is that the nuclear safety, like aviation safety, is the exclusive province of the Central Government and cannot be regulated by the local governments or counties.

However, some local counties have shown their desires to participate in matters relating to safety matters on the nuclear power stations. In response, the AEC declared its intent to cooperate with the local counties in the area of nuclear safety by keeping the counties informed of matters interest them, and considering participation of the county officials in the AEC inspection activities or the advisory committees. However, the counties can only observe and assist AEC's inspections, but can not conduct their own independent health and safety inspections.

The TPC, the largest producer of radioactive wastes in this nation, also plays a major role to communicate the local counties and townships on selecting the site of low-level or high-level waste repositories.

8.1.5.3 Legislative Yuan

The Constitution provides that the Legislative Yuan (LY), constituted of the public-elected representatives, shall be the supreme legislative organization of the country and shall exercise the legislative power on behalf of the people. In terms of its competence, power, and function, the Legislative Yuan is equivalent to a parliament in other democracies.

According to Article 5(1) of the organizational regulation of the Legislative Yuan's Procedure Committee, the Education and Culture Committee is responsible for the related bills of Ministry of Education, Council for Cultural Affairs, National Palace Museum, Government Information Office, National Youth Commission, Sports Affairs Commission, Academia Sinica and AEC. According to Article 2 of the organizational regulation of each LY's Committee, each Committee shall deliberate bills consigned by the LY's Conference and petitions of the public. At the beginning of a session, legislators may invite government representatives to provide reports or make presentations at the committee meetings and provide their comments on these issues.

8.2 Separation of Functions of the Regulatory Body from Those of Bodies Promoting Nuclear Energy

8.2.1 Separation of Functions of the AEC from the MOEA

Based on the Atomic Energy Act, the AEC has the regulatory authority for the nuclear power and radioactive materials. The MOEA under the Executive Yuan has the responsibility to maintain the stable supply of electricity. The Bureau of Energy (BOE) is a subordinate organization of the MOEA for developing the national energy policy. The BOE also has the responsibility on forecasting, and promoting of supply and demand of electricity. The MOEA also plays the role for promoting various types of energy including nuclear power while the AEC focuses on the regulatory part. The functions of the AEC are well separated from the MOEA. The Taiwan Power Company, as a State-Owned Corporation, is being supervised by the MOEA and is responsible of keeping nuclear power operation safe through research and demonstration projects and the accumulated experiences on construction and operation of nuclear power plants.

According to the Atomic Energy Act, the AEC may also promote and at the same time regulate for radiation applications. For example, the gamma irradiation plants require AEC's license to design, construct and operate and are under AEC's regular inspections. On the other hands, the AEC also supports researches to use the gamma irradiation plant for the purpose of improving public health.

8.2.2 National Policy and Role of the AEC on the Sustainable Energy

Sustainable energy development should balance the objectives of energy security, economic development and environment protection, and consider the need of future generations.

Taiwan is insufficient in natural resources and constrained by limited environment carrying capacity. In order to create a win-win-win solution in energy, environment, and economy, the sustainable energy policies should support the efficient use of limited energy resources, the development of clean energy, and the security of energy supply. Our targets are as follows:

(1) Improving energy efficiency

The goal is to improve the energy efficiency by more than 2 % per annum, so that when compared with the level in 2005, the energy intensity will decrease 20% by 2015. Supplemented by further technological breakthroughs and proper administrative measures, the energy intensity will decrease 50% by 2025. (The energy intensity is measured by the quantity of energy required per unit output or activity, so that using less energy to produce a product reduces the intensity.)

(2) Developing clean energy

A goal is set at reducing the nationwide CO_2 emission, so that the total emission could return to its 2008 level between 2016 ~ 2020, and further reduce to the 2000 level in 2025.

The share of the low carbon energy (or clean energy) in electricity generation systems is set to be increased from the current 40% to 55% (including natural gas) in 2025.

(3) Securing stable energy supply

A secure energy supply system was built to meet the economic development goals, such as 6% annual economic growth rate from 2008 to 2012, and 30,000 USD per capita income by 2015.

The basic principles of a sustainable energy policy is to establish a high efficiency, high value-added, low emission, and low dependency energy consumption and supply system.

The 2009 National Energy Conference of Taiwan was held on April 14 through 16. One of the topics discussed is “Sustainable Development and Energy Security”, and the nuclear power was discussed in one of its sub-topic "Adjustment of Low Carbon Energy Structure". The main conclusion is to enhance the nuclear power safety as below:

(i) To secure the safety of the construction, operation and radioactive waste management of the nuclear power plants, to improve the emergency response capability of a nuclear accident, and to establish the evaluation mechanism for the reasonable utilization of the nuclear power.

(ii) To establish a system to make information transparent and open to the public, to carry out a policy for the involvement in nuclear activities and jointly overseeing the nuclear power plants by the public and local residents, and to evaluate the performance of this by non-government organizations according to internationally recognized standards.

(iii) To strengthen the nuclear manpower cultivation and the evaluation of the nuclear technology research, and to carry out the education and guidance for the public to enhance their recognition of the nuclear safety management.

Table 8.1 Budget and Staffing by Appropriation

Appropriation	Budget (Million NT Dollars)			FY Staffing (Man-year)		
	FY2007	FY2008	FY2009	FY2007	FY2008	FY2009
AEC Headquarters	350	359	337	165	159	163
INER	2,547	2,662	2939	930	925	936
FCMA	63	63	63	35	35	36
RMC	60	58	60	30	30	31
Total	3,020	3,142	3,399	1,160	1,149	1,166

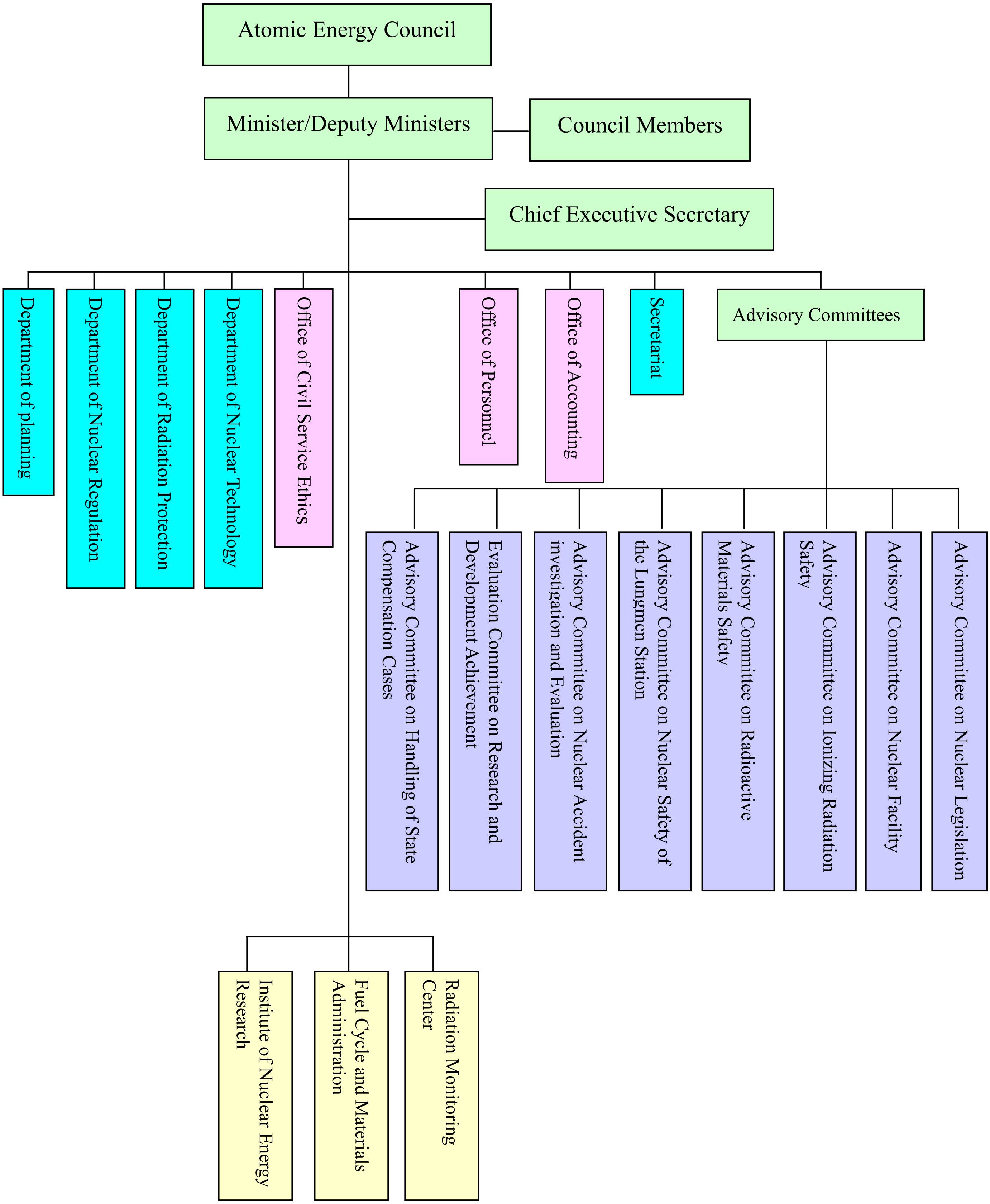


Figure 8.1 Organization Chart of the AEC

ARTICLE 9. RESPONSIBILITY OF THE LICENSE HOLDER

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

9.1 Prime Responsibility for the Safety of Nuclear Installations

According to the Articles 5 and 7 of the Nuclear Reactor Facilities Regulation Act, the holder of the construction license of a nuclear reactor facility assumes the responsibility to safely construct a nuclear power plant as approved and with the conditions imposed by the regulatory body at the time when the construction license was issued. Then as addressed in the Articles 6 and 7 of this Act, after the construction was completed and the application for operating a nuclear power plant was granted, the holder of the operating license shall assume the responsibility to safely operate the nuclear power plant as approved to ensure that the health and safety of the public are protected.

As specified in the approved final safety analysis report (FSAR) of each operating nuclear power plant, the license holder of the plant shall be responsible for the design, construction and safe operation of the plant. Also described in the FSAR of each plant are the responsibilities of the operational staff to carry out their duties for safely operating the plant. The Plant General Manager (who was previously named as the 'Superintendent' of the plant) shall take the principal responsibility for all phases of operation and maintenance. He is responsible for the safe, orderly, and efficient operation of the station and for the compliance of operation with the requirements of the operating license and technical specifications. He is also responsible for the training and retraining of the reactor operators (ROs) and senior reactor operators (SROs) as well as maintaining a qualified staff of technical and operational personnel for his plant. A Plant General Manager reports to the Vice-President of Nuclear of the TPC via the Director of the Department of Nuclear Generation (DONG) and carries out the policies as set forth by the TPC management as well as those prescribed by the Nuclear Safety Committee (NSC) at the TPC headquarters.

Figure 9.1 shows the general organization structure of a TPC nuclear power plant (NPP). In each of the three operating NPPs, there are three Deputy Plant General Managers with one of them in charge of the operation. These Deputy Plant General Managers will act as assistants to the Plant General Manager and, in case the general manage is absent or not available, one of them will take the authorities to act as the plant general manager. Both the Plant General Manager and his deputy of operation must have valid SRO certificates issued by the TPC. However, most of them actually had valid SRO licenses issued by the AEC.

During the periods when the Plant General Manager and all three Deputy Plant General Managers are not available, the Plant General Manager will delegate his responsibility to the Operation Section Manager (who was called 'Head of the Operation Division' previously). In the time period after office hours or when the Plant General Manager, three Deputy Plant General Managers and the Operation Section Manager are all not onsite, the on-duty Shift Manager (who was called 'Shift Engineer' previously) will be responsible for the operation of the plant and the compliance of operation with the requirements of the operating license and technical specifications.

The supervision of the plant operation and performance is under the direction of the Operation Section Manager who reports to the Deputy Plant General Manager in charge of operation and works closely with the managers of other technical sections. The Operation Section Manager must have a valid SRO license issued by the AEC.

The Shift Managers receive technical direction from the Operation Section Manager, but report directly to the Deputy Plant General Manager in charge of operation. The Shift Managers who must have valid SRO licenses are in charge of the plant operation during their shifts and have the authority to shutdown the reactor if necessary under their judgment.

The Shift Leaders (who were called 'Main Control Room Supervisors' previously) on duty are mainly responsible for controlling the nuclear power reactor units and meeting the plant load demand. They must have valid SRO licenses issued by the AEC, too. In case of an emergency event, if the Shift Manager is not available, the on-duty Shift Leader of a unit has the authority and responsibility to act in place of the Shift Manager.

At the TPC headquarters, both the Department of Nuclear Generation (DONG) and Department of Nuclear Safety (DNS) are responsible for assisting the plant staff to safely operate the plant. Of course, the final responsibility lies fully on the license holder.

9.2 Mechanism for the Regulatory Body to Ensure that the License Holder Will Meet Its Prime Responsibility for Safety

The AEC, in accordance with the Atomic Energy Act of 1968 (as amended in 1971) and the Nuclear Reactor Facilities Regulation Act of 2003, assumes the responsibility to verify that the license holder of a nuclear installation complies with the license conditions throughout the lifetime of the plant. It is the responsibility of the AEC to ensure that the license holder fulfills its legal duties, meets the regulations properly and carries out all the terms and conditions as specified in the licenses. Application for the construction or operating license of a nuclear power installation must be approved by the AEC as described in Subsection 6.2.3.1 of this report.

During the construction stage, the AEC will comprehensively review the safety of the design of a nuclear power plant and the capability of the applicant to design, construct, and safely operate a nuclear facility. In the meantime, the AEC will carry out various inspections to enhance the safety review as well as to make sure the construction and its quality are in compliance with the requirements of the construction license.

The operating license applicant of a nuclear power plant shall receive pre-operational inspections from the AEC to verify that the nuclear power plant is constructed as previously approved in the Construction License. After fulfilling all the requirements for the initial fuel loading as specified in Subsection 6.2.3.1 and obtaining an approval from the AEC, the applicant can start the nuclear fuel loading. Then, with the completion of all the pre-operational tests and startup tests (or power tests), including the systems' functioning, criticality and power ascension tests, and an approval from the AEC, the applicant will receive an operating license for commercial operation. After the operating license is issued, the AEC will continue, by the use of all kinds of regulatory means including reviews and inspections, to make sure that the licensee shall ensure the operation safety of its nuclear power plants.

The operation of a nuclear power plant shall receive periodic inspections from the AEC to assure that the performance of the plant maintains conformity with the technical standards prescribed in the relevant provisions, and that other performances, including the protection against pressure and radiation, maintain what they were during the pre-operational inspection.

The licensee is required to submit reports on operation, radiation safety and environment monitoring, emergency events, radioactive waste production, in-service inspections and tests, and the dose evaluation on the residents neighboring the plant to the AEC regularly or within a required period upon the occurrence of an event. By reviewing these reports, the AEC will be able to better understand the safety conditions of a nuclear power plant.

Furthermore, the AEC regularly holds a regulatory meeting with the TPC's high ranking staff from the headquarters such as the Vice President of Nuclear, managers from the DONG and/or DNS to discuss and exchange opinions on nuclear safety issues interested mutually or by either party. This kind of regulatory meeting between the regulatory body and the license holder are believed to be beneficial to the promotion of nuclear operational safety.

If a violation of the Regulations does take place, the AEC will immediately request the license holder to take corrective and complementary measures so as to secure the safety of the nuclear power plant. For example, if the operator of a nuclear power plant failed to meet the license conditions, the AEC may order the revocation of the license or the suspension of the license for a given period of time. Failure of a nuclear power plant to conform to the conditions imposed on the construction or operating license would subject the licensee to enforcement actions, which may include the receiving of a formal violation notice, the license being modified, suspended, or revoked and/or the receiving of a fine notice. The AEC may also order particular corrective actions or transfer the violation case to the court to ask for penalties including the criminal prosecution or a fine.

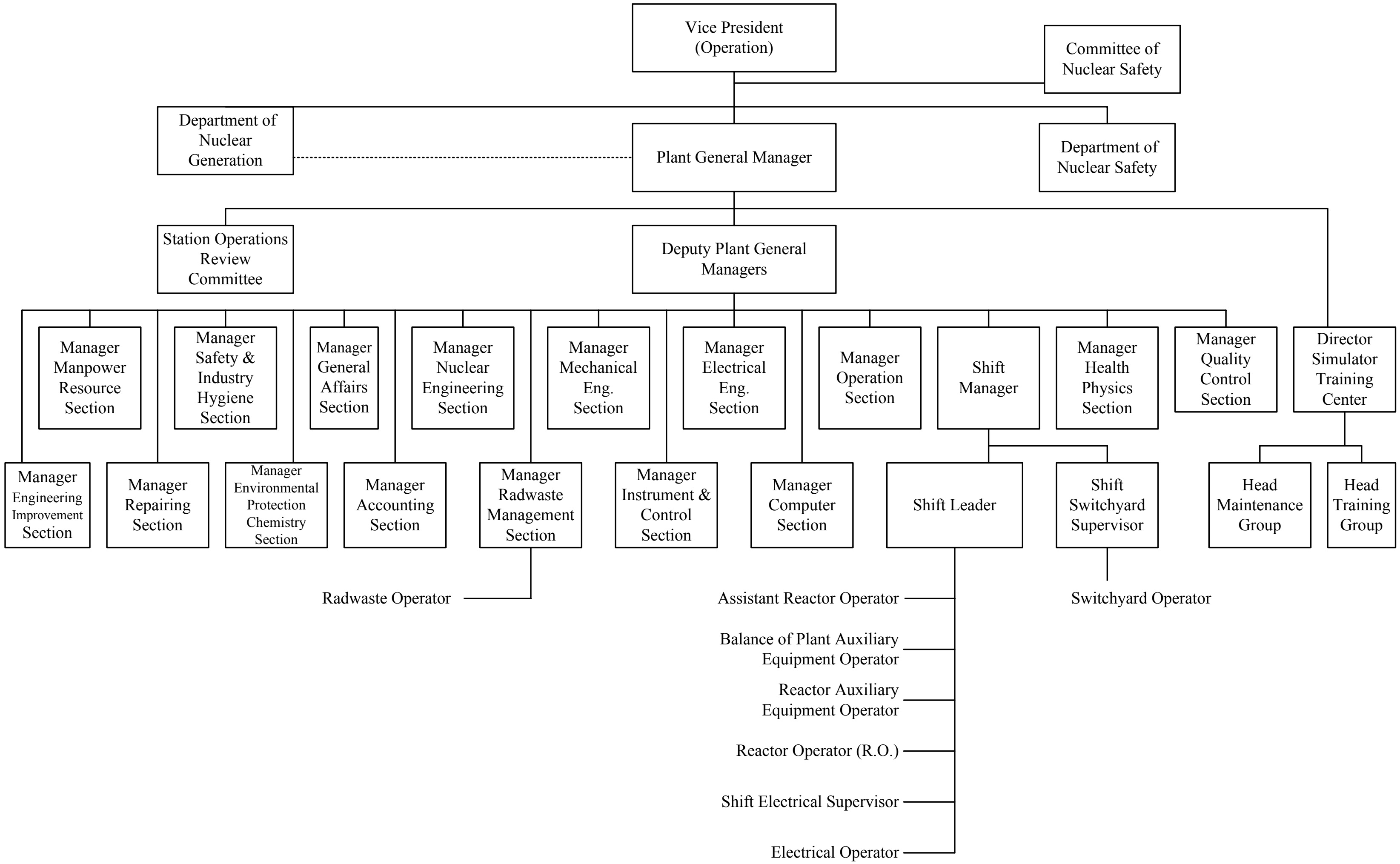


Figure 9.1 General Organization Chart of the TPC Nuclear Power Plant

ARTICLE 10. PRIORITY TO SAFETY

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

10.1 Safety Policy

The Atomic Energy Council (AEC) was founded as a ministerial level government entity in 1955 in order to handle the international affairs in the field of nuclear energy. Then, on May 7, 1968, the "Atomic Energy Act" was promulgated and in accordance with this Act (which was amended later in 1971), the AEC became a ministry-level organization under the Executive Yuan (the Cabinet) established by Law. To serve as the Nation's governing authority of atomic energy related affairs, the responsibilities of the AEC include safety regulation of nuclear facilities and radiation workplaces, and protection of the public and the environment from the adverse effects of radiation associated with nuclear materials and facilities. To ensure the safety of atomic energy applications, the AEC strictly implement the laws for nuclear safety control, radiation protection, environmental detection, and management of radioactive wastes. For the regulation of nuclear and radiation safety, the AEC upholds the principles of "safety first, reasonable regulation, and convenience to the people". The AEC will continue to strengthen the ability of response to nuclear incidents, and to reinforce safety control for the application of ionizing radiation in domestic medical, agricultural, academic and industrial sectors, so as to ensure the health and safety of radiation workers and the general public. In addition, the AEC will never stop the efforts of enhancing the transparency of nuclear safety information to the public.

10.2 Safety Culture

Although there is no regulation requiring the Taiwan Power Company (TPC) to implement the safety culture in its nuclear power plants, the AEC, responsible for the nation's nuclear safety, keeps reminding and encouraging the TPC to pay attention to the safety culture. After the Chernobyl accident in 1986, the International Nuclear Safety Advisory Group (INSAG) of the IAEA declared that the safety culture should be well established, understood and respected throughout the organizations of nuclear installations. A lot of discussions and developments in this area have been made under the leadership of the IAEA since then. Following this international trend, the TPC has developed its own safety-culture fostering program with reference to the associated IAEA reports since 1988.

After many processes of instruction by consultants, organization changes, trainings, meetings and discussions, the TPC established a "Safety Culture Implementation Plan" in 1993 and had implemented this plan since 1994. In this plan, the safety culture is promoted with the following 5 principles:

(1) Duty

(a) The responsibility for each position and the transfer system of information must be specified clearly.

(b) Everybody is responsible for his assignment.

(c) Good working procedures and practical ways of following procedures have to be established.

(2) Training

(a) Both “know how” and “know why” have to be emphasized in training.

(b) Performance evaluation systems must be set up.

(c) A nuclear technology training center must be established.

(3) Discipline

(a) A good working environment including promotion, training, reward and punishment, communication, and leadership must be created.

(b) Evaluation systems must be set up and problems have to be discovered and solved in advance.

(c) All ranks of superiors have to watch the repeated errors seriously.

(4) Regulation

(a) A self-regulating system must be set up to evaluate the daily works.

(b) Problems must be discovered by utilizing auditing, operational indicators, trend analyses and evaluation techniques properly.

(5) Execution

(a) Safety culture is a top-down process. All ranks of superiors have to make themselves models for the staff.

(b) All ranks of superiors have to set up goals and follow up the performance of execution.

In addition, the approaches that the TPC adopted to promote nuclear safety culture include:

(1) Inviting personnel from relevant units to participate in the nuclear safety culture seminars,

(2) Setting up a nuclear safety culture column in the Nuclear Monthly, a journal published monthly by the TPC, to introduce and convey the readers domestic and foreign professional nuclear techniques and the experience of handling nuclear incidents,

(3) Promoting the state-of-the-art knowledge of nuclear safety in every sector of generation to make all employees deem nuclear safety a part of their life, and

(4) Keeping on following-up the promotion status of the nuclear safety culture and

sending the results to the Ministry of Economic Affairs (MOEA) and the AEC quarterly. Accepting on-the-spot inspection of the performance of the safety culture conducted by the MOEA every year.

In July of 1996, the TPC issued the "Prevention Measures for Human Errors in Nuclear Power Plant" to reduce incidents caused by the operators. Furthermore, the TPC started a "Nuclear Safety Culture Reinforcement Plan" in 1997 to promote the safety culture up to a higher level. The following 6 targets consist of the cornerstones of this reinforcement plan.

Target 1: Declare Safety Commitment and Make Safety Culture Real Practice

(1) Sign the "safety commitment" by the members of all groups (i.e. plant, department, and all sections) and hang the signed document on the walls of the offices of each group,

(2) Promote the performance of employees and the level of safety culture by way of the organization's activities and propaganda, and

(3) Require all employees to think much of safety and enforce patrol of working places and the superiors of all ranks to enforce "Walking Management" in the working places.

Target 2: Practice the Following 10 Preventive Measures for Human Errors

(1) Conservative decision making,

(2) Potential risk assessment,

(3) Tool box meeting,

(4) Self-checking,

(5) Procedure adherence,

(6) Double checking,

(7) Communication enhancement between operation and maintenance,

(8) Error prevention of the contractor's employee,

(9) Experience feedback, and

(10) Human error root cause analysis.

Target 3: Follow the Procedures

(1) To develop procedure-adopting practice. All the adopters and users are responsible for the completeness and correctness of the procedures

(2) Superiors of all ranks have to teach their subordinates to follow the procedures

Target 4: Promote Self-Evaluation Capability

(1) Develop the self-evaluation practice, and

(2) Perform the assessments of nuclear projects aggressively.

Target 5: Increase Equipment Reliability

(1) Perform equipment reliability trend analysis to find out problems in advance,

(2) Use the scheme of "group discussion on plant system" to evaluate the potential system problems, so as to prevent the occurrence of unexpected incidents, and

(3) Perform the "root cause analysis" according to the procedures, correct the errors thoroughly and follow up the corrective action until its completion.

Target 6: Promote Training Performance

(1) Push the personnel qualification and certification program to work,

(2) Improve the planning of training program, and promote the quality of teaching, and

(3) Establish an unbiased and objective evaluation system for training performance.

In addition, there are 12 indicators divided into 2 categories being selected to evaluate the performance of "Nuclear Safety Culture Reinforcement Plan". These indicators are:

Category 1: Quantitative Indicators for the Results

- Number of incidents caused by human errors,
- Number of incidents violating the regulation of the 4^{th} degree and above,
- Number of unplanned auto scram in 7000 hours of critical condition (number per unit),
- Performance of safety systems (% of total time in service),
- Number of industrial safety (number of incidents per 200 thousand man-hours per plant), and
- Percentage of unplanned capacity loss.

Category 2: Quantitative Indicators for the Processes

- Number of walking-management (man-times per quarter),
- Number of supervision of tool-box meeting (man-times per quarter),
- Number of supervision of self-assessment (man-times per quarter),
- Number of safety condition improvement (man-times per quarter),

- Percentage of completion of request for equipment repair (%), and
- Number of meetings for system discussion (times per quarter).

Along with the above programs, the TPC also conducted the safety culture assessment to evaluate the effects of these programs on safety performance and to pinpoint the weakness. This assessment included two parts: one was the safety culture indicators review and the other was the performance evaluation. For the latter part, a team consisting of members from the TPC head offices and the three nuclear power plants went to each plant site for the safety culture performance evaluation. In addition, the Commission of National Corporations (CNC) of the MOEA, which is the supervisory organization of the TPC, would also organize a team with experts from universities, government agencies and news media to assess the plant safety culture annually. All findings by these teams were fed back to the plants for the improvement of safety culture.

10.3 Commitment to Safety

AEC is the governing authority for the regulation of all atomic energy related affairs in the country. “Safety” has always been the top priority in the AEC’s line of responsibilities. To effectively carry out its mandates, the AEC will keep on exerting her greatest efforts in the development of relevant Acts and regulations, improvement of regulating mechanisms, enhancement of technical expertise and professional capability, and fostering of the safety culture. To earn the public trust, the AEC is committed to ensure the highest standards of nuclear safety and radiation protection.

For the nuclear power plants, in order to enhance the safety of nuclear power generation, the TPC announced a “Nuclear Power Operational Safety Policy” at the beginning stage of the safety culture fostering period. The Policy Statements, revised on August 27, 2004, include the following contents:

The TPC follows the nuclear regulations and considers the continuously safe operation of nuclear power units as the most important prerequisite to provide a long-term stable power supply. The operation of nuclear power plants must avoid reactor core damage and abnormal release of radioactive materials to ensure the safety of the public health and property. In order to achieve these safety goals, the nuclear safety management is the first priority task of nuclear power operation. In order to establish consensus to achieve and maintain the safety of nuclear power operation cooperatively, the TPC promulgates “Nuclear Power Operational Safety Policy” as follows:

(1) The nuclear power operational safety is the responsibility of every one involved in the nuclear power generation business. All the workers, managers, and regulators related to safety have to be responsible for their own missions respectively.

(2) The Acts, regulations, standards, specifications and operating procedures related to the nuclear power operation have to be strictly followed. Applications for corrections or exemptions of them need to be proposed in advance, if difficulties to fulfill any requirement arise. The existing rules must be strictly followed until these applications are approved by the appropriate authority.

(3) Safety culture has to be promoted; Self-evaluation and self-regulation strategy have to be established. In order to find and solve abnormal issues, daily operations will be assessed by auditing, performance index review, trend analysis, and evaluation techniques.

(4) To conform to the regulatory requirements, the TPC has to do its best to prevent the occurrence of abnormal events and violations of regulations. In addition, the TPC has to perform evaluation and establish preventive strategies for those potentially dangerous test, inspection and maintenance.

(5) The TPC's goal is not to conform to the lowest regulatory requirements only, but to pursue highest operational safety.

In order to promote the performance of the nuclear power operational safety, all colleagues in the nuclear operation departments have to cooperatively develop good nuclear safety culture with aggressive and responsible altitude, and sense of mission and honor. The safety culture has to be demonstrated in every daily practice as well. It is expected that all these efforts will make the TPC's nuclear safety operation step from good to excellence.

10.4 Regulatory Control

The Atomic Energy Act is the basic Act that provides the legislative and regulatory framework of the utilization of nuclear energy. The objectives of this Act are to promote the research and development of the nuclear energy science and technology, and the development and peaceful usage of the natural nuclear resources. The Atomic Energy Act was first promulgated in 1968 and then modified in 1971. The Article 3 of this Act stipulates that the "Responsible Agency" for the Act shall be the AEC.

The AEC was founded in 1955 at the ministerial level under the Executive Yuan. The principal mission of the AEC in the initial founding period was limited to the management of international affairs concerning atomic energy and the promotion on the peaceful applications of the atomic energy in the country. In recent years, the most important tasks of the AEC have been shifted to the safety regulation, radiation protection, radwaste administration, and R&D for the nuclear technology and civilian nuclear applications. The legislative and regulatory framework, Acts, regulations, and requirements associated with the nuclear safety are described in Article 7, while the structure and responsibilities of the AEC are introduced in Article 8 of this report.

10.5 Voluntary Activities and Good Practices Related to Safety

Among many voluntary activities related to the nuclear safety, the first one worth mentioning is the experience feedback. In order to learn from the past experience, the worldwide operational as well as regulatory information are constantly collected and studied by the AEC and the TPC. Causes of abnormal events are investigated to check if similar situations exist in domestic facilities. Good practices are learned and propagated among working staff. In addition, safety issues experienced by any domestic plant would be reported to the other plants, so that similar mistakes can be avoided and good safety measures can be shared. To share the important operating and maintenance experiences among plants, the TPC worked out a program, namely, the Operation Experience (OE) program, to be applied to all its nuclear installations. It turns out that the OE program is a sharp tool to seek ways of improving the performance of the nuclear power plant. Besides the experience feedback program, a lot of additional efforts have been made to enhance the safety of nuclear facilities. Some examples are delineated as follows.

(1) Regulatory Conference

The AEC and the TPC hold periodical meetings to discuss topics such as recent nuclear activities in the other countries, the malfunction and abnormal occurrences, safety improvement measures, and the new plant status, etc. The purpose of these meetings is to reach consensus about the nuclear safety concerns and their remedies.

(2) Investigation of Reactor Scrams and Forced Outages

All of the six operating nuclear power units in Taiwan, including four BWRs and two PWRs, are designed and manufactured by the United States vendors. Therefore, all activities essential to the nuclear power plants (NPPs), such as design, purchasing, fabrication, handling, shipping, storage, cleaning, erecting, installation, inspection, testing, operation, maintenance, repairing, refueling, and modification, are subject to the codes and standards similar to those issued by the USNRC. For this reason, the permission to restart the unit after refueling outage was not necessary for the AEC to approve in the earlier years of operations of these NPPs. However, for reducing the frequency of nuclear unit scrams and forced outages, the AEC had decided to regulate the nuclear unit restart after refueling outage to assure the maintenance quality of structures, systems, and components (SSCs) of the facility and to improve the plant performance since 1987.

Besides, in case of a reactor scram, the TPC must report to the AEC about the consequence and probable root causes of the scram within two hours after its occurrence. The AEC may agree to restart the reactor only when the root causes are clarified, safety assessments are satisfactory, and necessary corrective actions have been implemented. If an operating unit requires a safety-related design modification or equipment change, the TPC has to submit the application in advance with necessary documents about its causes, procedures of modification, safety assessment and so on. The AEC will review these documents and monitor all the related activities until the modifications are satisfactorily completed.

(3) Investigation of Plant Abnormal Occurrence

Within 30 days of the occurrence of an abnormal event, the TPC has to investigate the root causes, propose remedy measures and submit a report to the AEC. The AEC will review the remedy actions and dispatch inspectors for field inspection if necessary. The implementation of the measures will be followed up by the AEC until the issue is effectively resolved.

(4) Investigation of Plant Equipment Malfunction

If an equipment malfunction was identified as significant to safety, the TPC has to investigate the root causes, propose remedy measures and submit a report to the AEC for review. The implementation of the measures will be followed up by the AEC until the issue is effectively resolved.

(5) Development of Severe Accident Management Guidelines

After the Three Mile Island (TMI) accident, the nuclear industry performed a large-scale severe accident research to understand the phenomena and develop the analysis code for improving the prediction capability. The goal of the severe accident research is to develop a Severe Accident Management Guidelines (SAMG) for the plant staff to mitigate the severe accident. At the end of 2003, the TPC has established its own SAMGs specific to the Chinshan, Kuosheng and Maanshan nuclear power plants, respectively. The development of the SAMG included the evaluation of the system status (hardware capability), plant control parameters (instrumentation availability), establishment of the interface between the emergency operating procedures (EOP) and the SAMG, verification of the SAMG and training of the operators. According to the SAMG of the TPC, an Accident Management Team (AMT) has been established in each operational nuclear power plants. The members of the AMT consist of the operation section manager, supporting shift manager, quality control section manager, and nuclear engineering section manager. The responsibilities of the AMT are providing Technical Support Center (TSC) with appropriate suggestions for responding to the severe accidents. In the Kuosheng plant, an information sharing system has been developed to perform those actions required by the SAMG. With this system, reading the flow charts becomes easier, the efficiency of group discussions becomes better, the information for decision making becomes more transparent, and the contents of management guidelines become more complete. It is believed that the SAMGs of the three existing nuclear power plants can enhance the severe accident management capability of the plant staff.

(6) International Technical Evaluation and Peer Review

The TPC has invited several international nuclear groups, such as the Institute of Nuclear Power Operations (INPO) and the World Association of Nuclear Operators (WANO), for safety review and discussion. For example, during the last few years, the TPC had the following relevant activities:

(a) On May 30 to June 1, 2001, two specialists from Kansai Electric Power Company of Japan visited the Maanshan NPP to share the experiences gained by this plant in the Station Blackout event, occurred at Unit One.

(b) On May 27 to June 14, 2002, a team of twenty-one specialists organized by WANO-Tokyo Center (WANO-TC) visited Chinshan NPP for the WANO peer review.

(c) On August 8-17, 2002, US INPO sent four specialists to Kuosheng NPP for technical exchange visit.

(d) On November 10-14, 2003, US INPO sent two maintenance experts to the TPC to give a “Maintenance Supervisor Professional Development Seminar”, with the “Equipment Reliability” as the main theme.

(e) On November 7 to 24, 2005, a team of 23 specialists organized by the WANO-TC visited Maanshan NPP for the WANO peer review.

(f) On December 6 to 19, 2007, a team of 14 specialists organized by the WANO-TC visited Kuosheng NPP for the WANO peer review. This team consists of specialists from USA, Janpan, South Korea, India and France. The areas of review included the organization and administration, operation, maintenance, engineering support, and operating experience.

A lot of improvements have been made according to the suggestions of the above-mentioned international specialists. For example, the following activities have been performed in Chinshan NPP with reference to the suggestions of the WANO peer review:

(a) Improvement of the self-assessment function,

(b) Improvement of the working environment, such as management of barriers in the walking passage, addition of emergency stop button for rotating machine, management of laying up of stuff and equipment,

(c) Reinforcement of the protection measures for human errors, such as adherence of procedures, self-assessment, and verification of instructions,

(d) Improvement of the notice for operation, such as elimination of hand-writing notice, periodical check of notices on control panel, and addition of notice correction action in the application form for set point of instrumental and electrical equipment,

(e) Improvement of the housekeeping, such as periodical check for cleanness conditions of equipment, timely removal of greasy dirt, check for the completeness of valve tags and improvement of laying up of stuff and equipment in the plant buildings,

(f) Establishment of the control requirements and the check lists for cranes and fittings to enhance the safety of crane operation,

(g) Improvement of the environmental conditions of the warehouses for spare parts, such as improvements of humidity control and corrosions on roofs and walls,

(h) Improvement of the determination methods for degraded and nonconforming equipment,

(i) Improvement of the methods for root cause analyses to avoid repeated occurrence of similar events,

(j) Improvement of the methods for contamination removal to avoid expansion of contaminated areas,

(k) Improvement of the control of chemical laboratories and storage of chemical stuff, and

(l) Improvement of the utilization of external operating experiences to develop corrective actions to preclude recurrence for important plant events.

As for Kuosheng NPP, the following activities have been performed with reference to the suggestions of the WANO peer review:

(a) Addition of the anticipated alarms during normal operation into procedures. The purposes of this improvement are to reduce the disturbance to operators and let the operators concentrate on those unpredictable alarms.

(b) Categorizing all valves in the plant and putting a tag with appropriate notice on each valve. The purpose of this practice is to decrease the human errors associated with valve operations.

(c) Addition of graphs with detected dose rates including high-dose spots in front of each area with high dose rate

(d) Improvement of the radiation safety information system in the control station of the plant by constructing a radiation control system in accordance with the philosophy of As Low As Reasonably Achievable (ALARA)

(e) Establishment of the control requirements for laying up of stuff and equipment inside plant buildings. The ways of laying up in all plant buildings were improved and relevant personnel were trained according to these requirements. The purposes of these activities are to avoid impacts of improper laying up on plant safety and reliability, and to avoid that safety systems being not able to perform their safety functions during earthquake events.

(f) Establishment of the control requirements and the check lists for cranes and fittings to enhance the safety of the crane operation

(g) Establishment of Foreign Material Exclusion (FME) area and guard in the lower fuel pool to prevent the fuel from damages caused by the invasion of foreign materials

(h) Setting up observation standards for working management areas and training the persons responsible for these areas to do trend analyses for errors and to implement performance rating practice

In addition, the following activities have also been performed in Maanshan NPP with reference to the suggestions of the WANO peer review:

(a) Reduction of the rusty debris in the steam generator by addition of ethanolamine (ETA) into the feedwater of the steam generator

(b) Establishment of the reactivity management policy to set up the concepts, responsibilities, and requirements of the reactivity management and to meet the goal of nuclear safety

(c) Establishment of the corrective action program to integrate the walking-management, near-miss events, employee's suggestions, flaw reports of components, work orders, etc. into a single-point-of-contact process

(d) Establishment of the high-level guidance for the walking-management and separation and evaluation guide lines for plant affairs assignment

(e) Categorization of the experience feedback sources of industry and assignment of appropriate sections or divisions to search, study, and share the feedback information for each category on monthly basis

(f) Reduction of the threshold of human performance enhancement system (HPES) to deal with the human errors having not caused significant adverse results, and hence, to further reduce the risk caused by human errors

(7) Corrective Action Program

With reference to the WANO's guidelines, the Mannshan NPP started to implement a Corrective Action Program (CAP) at the beginning of 2007. In this program, all corrective actions required from 17 sources, including findings by regulatory auditing, superior management persons, working staff, self-assessments and so on, are integrated and investigated. Then the problems associated with these actions are divided into 7 different areas, such as errors of system, implementation, human, design, management, house keeping and others, so that responsible divisions can be assigned accordingly. Importance levels of these problems are also evaluated according to the significance of their impacts on plant safety and operability. Resources for corrections are then allocated according to the importance levels.

The Kuosheng NPP also implemented a CAP starting from November of 2007. The requiring sources of the corrective actions are similar to that of the Maanshan NPP. A data base and information analyses system has been developed in the plant intranet. With this system, reporting, trend analyses, statistical evaluation of the problems associated with corrective actions can be performed through network.

In order to integrate CAPs of different NPPs, the Headquarters of the TPC started to develop a unified CAP for all existing NPPs in 2009. This program is to integrate individual problem-solving mechanisms in each existing NPP, so that the identification, categorization, correction, tracking, analysis, and resources integration for operational and maintenance problems can be implemented effectively. The root cause analyses, common cause analyses, trend analyses, evaluation of the effectiveness of corrective actions, and performance indicators associated with the NPP operation will be reinforced through this program. In addition, the management at the TPC Headquarters and information and resource sharing associated with the corrective actions can be effectively improved as well.

(8) Performance of the Maintenance Rule

In order to regulate the effectiveness of the NPP maintenance, USNRC promulgated "Maintenance Rule" in 1991 and required all US NPPs to implement this rule in 1996. The operational safety and performance of US NPPs were improved significantly since the implementation of the Maintenance Rule. To accompany with activities for promotion of the operational performance such as License Renewal, self-regulated on-line maintenance and maintenance optimization, the TPC required its three operational NPPs to implement the maintenance rule on August 2004. The major goals of this requirement include:

(a) Monitor and control the effectiveness of Maintenance,

(b) Evaluate the maintenance mechanism periodically, and

(c) Assess and manage the risk associated with the maintenance.

In the year of 2008, the Chinshan NPP had 537 items being included in the scope of the Maintenance Rule. Among these items, there was only one item which was determined as the (a)(1) category (i.e. the monitoring category). While for the Kuosheng NPP, there were 202 system functions being included in the scope of Maintenance Rule in the same year, and three of them were determined as category (a)(1). A Maintenance Integrated Risk Utilities (MIRU) computer system had also been developed for the predictive maintenance, arrangement of 13-week cycle schedule, and the on-line maintenance according to requirement (a)(4) of the Maintenance Rule in the Kuosheng NPP. As for the Maanshan NPP, the MIRU system was established in 2007 and applied for the maintenance risk management as suggested by procedure (a)(4) of the Maintenance Rule. The numbers of items belonged to the scopes of the Maintenance Rule and the category (a)(1) in the Maanshan NPP are 242 and 6, respectively, in 2008. In conclusion, the advantages of implementing the Maintenance Rule in the existing NPPs in Taiwan include:

(a) Implementation of Maintenance Rule is one of the requirements for the application of license renewal.

(b) The weak points in maintenance can be effectively identified by the quantitative monitoring measures of the Maintenance Rule, and hence, the reliability of equipment can be effectively improved.

(c) The effectiveness of performing the Maintenance Rule can be continuously improved by periodical performance evaluation as required in procedure (a)(3) of the Maintenance Rule.

(9) Improvement of the Technical Specifications

Originally, the Technical Specification (TS or Tech. Spec.) used by the Chinshan NPP was called “Customer’s TS” and those for the other domestic operational NPPs were “Standard TS”. The following shortcomings were found through the implementation of these TSs:

(a) Too many information were included in the TSs.

(b) The bases and explanations of requirements were unclear.

(c) Some requirements were not specific or even irrational.

(d) The formats and contents of the TSs did not conform to the human engineering, and hence, resulted in difficulties of utilization by the operators.

To correct these shortcomings, Chinshan, Kuosheng, and Maanshan NPPs adopted the so called “Improved TS” (ITS) on February 2002, January 2008, and September 2004, respectively. In these ITSs, fire extinguish equipment, radiation protection and environmental monitoring, snubbers, equipment lists, meteorological instruments, and water chemistry are removed and controlled by the technical manual or specific programs. The improvements of the formats and contents make the operators appreciate the meanings of the TS much easier. The safety of operations was enhanced accordingly. In addition, the administrative burden associated with the implementation of TS was reduced, and hence, the operational performance was improved.

(10) Investigation of Near-miss Events

The near-miss events are those events that have some component failures or malfunctions but the severity of which having not reached the level of abnormal events. These events are divided into 8 categories including: 1. work safety, 2. operation, 3. maintenance, 4. radiation safety, 5. nuclear safety, 6. traffic, 7. work process related, and 8. others. In order to further improve the NPP safety, the TPC established a "Near-Miss Team" to deal with this kind of events in 2002. The Near-Miss report form is available on the intranet system of the TPC, and the employees and contractors can initiate the near-miss reports and submit them to the Near-Miss team through intranet. The team will review the reports, investigate the root causes, and provide corrective actions when necessary. Rewards according to the benefit obtained will be given to the person or persons who propose the near-miss reports. In addition, the near-miss experience feedbacks are available in the intranet. Everyone in the TPC can study this valuable information and prevent similar errors from occurring again.

(11) Reactor Oversight Processes (ROP)

In order to provide the public an easy way to understand the safety levels of the operating NPPs, the AEC has referred to the ROP of the USNRC and developed the domestic ROP in 2005. The performance indicators associated with initiating events, mitigating systems, barrier integrity, emergency preparedness, and nuclear security are evaluated in this system regularly. The results of evaluations are translated into green, white, yellow or red color to reflect the different levels of safety concern for each existing NPP. The public can easily tell how safe the plants are from the website of the AEC. These colors are also important references for the AEC to decide the frequencies and scopes of inspections for each NPP.

10.6 Measures to Enhance Transparency of Nuclear Safety Information

Communication is a very important mechanism for effective regulation. The AEC holds periodic regulatory meetings with the licensee to enhance the reactor safety. Meetings with stakeholders are also held whenever new laws are enacted, regulations promulgated or policies announced. For the public outreach, the AEC holds press conferences monthly to inform the general public, through media, of its major activities such as regulatory decisions, inspection results, etc. Information that is of interest to the public is routinely posted on the AEC's Website.

To enhance the transparency of nuclear safety information, the AEC took one step further to make selected real-time data of the NPPs available on its Website. Currently, several types of the plant operational and environmental monitoring data are transmitted to the AEC's Nuclear Safety Duty Center, a 24-hour and all year round working, centralized reporting system for abnormal events and an inter-ministerial communication gateway within the framework of the national disaster prevention. During the Year 2004, some of this information became available to the general public at the AEC's Website. First of all, the real-time color-coded data of selected parameters from the Safety Parameter Display System is posted, with reader-friendly presentation of the operational status of each nuclear power station. Second in line are the real-time environmental radiation monitoring data, such as High Pressure Ionization Chamber (HPIC) readings, updated every hour from data at site boundaries of all NPPs. In addition, the area gamma radiation updated every hour for 30 sampling stations in the entire Taiwan area are also available.

Nuclear technology and applications are widely recognized as of international nature. There is growing international cooperation in the nuclear communities, safety regulations and R&D to enhance the safety of nuclear activities. Although Taiwan is not a member state of the United Nations (UN), there have long been bilateral cooperative relations of Taiwan with advanced nuclear countries such as France, Japan, Sweden, Switzerland, UK, and USA, in various aspects of nuclear programs.

The AEC also takes part in some of the cooperative activities and training seminars sponsored by the OECD's Nuclear Energy Agency and the IAEA, regarding such topics as reactor safety, decommissioning and decontamination, environmental monitoring, and nuclear safeguards, and will continue to seek opportunities for such participations. In the area of international nuclear safeguards, the IAEA conducts safeguards inspections in Taiwan following the spirit of the United Nations' Nuclear Non-Proliferation Treaty and an Additional Protocol with the IAEA.

The AEC is an active member of the World's Nuclear News Agency. Press release associated with major regulatory decisions, nuclear safety issues, annual operational data, or new development/status of major nuclear related projects in this country is sent regularly to the Agency for reporting on "NucNet News".

The Nuclear Energy Society, Taipei (NEST), an assembly of representatives of the nuclear and radiation related societies and associations, has provided another channel for Taiwan to communicate with the international nuclear communities on the subjects of nuclear safety enhancement. The NEST has been an active member of the Pacific Nuclear Council (PNC) and the International Nuclear Societies Council since 1990, and also hosted the 8th Pacific Basin Nuclear Conference (PBNC) in 1992. The NEST has held member meeting twice a year and sent representatives to attend the biannual PBNC regularly for more than two decades.

ARTICLE 11. FINANCIAL AND HUMAN RESOURCES

1. Each Contracting Party shall take the appropriate steps to ensure that adequate financial resources are available to support the safety of each nuclear installation throughout its life.

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

11.1 Financial Resources

11.1.1 Requirements

According to the Nuclear Reactor Facilities Regulation Act (Articles 5 and 6) of 2003, the Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities (Article 3), the Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities (Article 14), and the Enforcement Rules of the Atomic Energy Act (Article 15), the license applicant of a nuclear power plant is required to have, in addition to the technical and management capabilities, the sufficient financial resources for the safe operation of the nuclear facility throughout its lifetime as a prerequisite for granting a construction permit (or construction license for recent applications), an operating license, or the nuclear fuel license. The Nuclear Damage Compensation Act of 1971 (as amended in 1997) also requires the license holder to have the responsibility for compensating the persons whose health (or life) and property were damaged by a nuclear accident. The compensation liability is limited to a total amount of 4.2 billion New Taiwan dollars (NT$ or NT dollars).

11.1.2 Financial Resources of the Licensee

According to the Article 2 of the Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities of 2004, a qualified applicant for a construction license of a nuclear reactor facility must be a legal company with assets more than one hundred billion NT dollars.

The TPC, a government invested public utility company, is the sole operator of nuclear power plants in Taiwan. It is established with a mission to stably and safely fulfill the electric power demand in this nation through effective energy source development and power management programs.

The total TPC's assets were worth about NT$1,587 billions (~US$50 billions) at the end of the year 2009. An access to adequate funds for the safe construction, operation, decommissioning, and final disposal of nuclear spent fuels and radioactive wastes is a necessity for the licensee to protect the public health and safety. Thus, the Nuclear Reactor Facilities Regulation Act requires the licensee to have sufficient financial resources to properly construct and safely operate the nuclear power plants. Although there does not appear a consistent relationship between a licensee's financial condition and its safety performance, it is evident that a financial pressure will certainly limit the resources available for corrective actions, improvements, upgrades, and other safety-related expenditures. Furthermore, a nuclear power unit must be in operation in order to collect the funds for the backend management including the eventual plant decommissioning and the final disposal of the spent nuclear fuel and radioactive wastes. Thus, any early shutdown of a nuclear power plant before sufficient funds have been accumulated could potentially hinder the implementation of the decommissioning, spent nuclear fuel and the radwaste disposal programs of the plant.

There were once some proposals in the government sectors to make the TPC becoming a private utility and to shutdown some operating nuclear power plants before their current operating licenses were due. Although these proposals did not come true (at least so far), the AEC will constantly and closely monitor the proposed TPC's privatization program and any possible early closure of the three existing nuclear power plants in operation.

11.1.3 Financing of Safety Improvements

The TPC has established a betterment plan for the safe operation and reliability improvement of each nuclear power plant and planned to secure the required research and development fund on its own. The TPC has replaced and/or reinforced its facilities under the Mid- and Long-term Betterment Program. For example, the TPC has completed the replacement and/or upgrading of the systems and equipment such as the simulators, feedwater control system, reactor protection system, plant monitoring system, turbine rotors, field instruments, wide range neutron monitoring system, automatic voltage regulator and power system stabilizer, hydrogen water chemistry system, ultrasonic flow measurement system, emergency circulating water system, spent fuel pool re-racking, and automatic seismic trip system for both BWR and PWR units. Thus, a significant investment has been made in the betterment plan of the nuclear units. Table 11.1 gives the total number of design change requests resulted from the betterment programs of the three operating nuclear power stations in recent years.

The AEC also performed necessary regulatory research and development as part of its Mid-Term and Long-Term Nuclear Energy Research and Development Programs for maintaining the safe operation of nuclear power plants and revising regulations to take into consideration the state-of-the-art nuclear technology and the ever-increasing environmental requirements. To this end, the Atomic Energy Act (as amended in 1971) stipulates that the AEC should be responsible for funding the research and development programs to promote the nuclear science and technology.

11.1.4 Financial Provisions for Decommissioning and Radioactive Waste Management

The Radioactive Waste Management Policy of 1988 as amended in 1997 stipulates that the license holder of a nuclear power plant shall establish a nuclear backend fund for the decommissioning of the nuclear installation and the final disposal of the spent nuclear fuels and low-level radioactive wastes. The TPC estimates the total cost for these nuclear backend activities on the basis of the installed capacity, projected quantity of the radioactive waste, the commodity price index and the international experiences. This fund has been collected on the basis of the amount of electricity generated by the nuclear power plants since 1987. In the first two years, the sharing rate of electricity for the fund was set at NT$ 0.14/kwh, which was gradually raised to NT$ 0.17/kwh in 1993 and then to NT$ 0.18/kwh in 1998. In 1999, the "Rules for Control and Application of the Nuclear Backend Fund" was promulgated and became effective. The management of the backend fund and the collection rate for this fund were based on this Rule. The rate, which is currently NT$0.17/kwh (about 5 US mills per kwh) in 2010, is adjusted annually so that it would adequately accommodate the cost inflation. The net amount of the backend fund reached NT$205.7 billion as of August 31, 2010. The estimated total cost of the TPC's nuclear backend management programs is updated whenever necessary to be commensurate with the development status of the TPC's nuclear power programs, industrial technologies and government regulations. The latest update was completed in 2010. Under the scenario of operating each of the existing six nuclear power units for 40 years, the estimated total nuclear backend cost is about NT$335.3 billion at the currency value of 2008.

An ad hoc committee, established under the Ministry of Economic Affairs, is responsible for the management of this fund. This Committee, the Nuclear Backend Fund Management Committee, is comprised of 14 members from the government organizations, universities or colleges, and research institutes. In the meantime, the AEC has been closely monitoring the fund-related activities since the establishment of this fund.

On the other hand, the cost for the treatment of radioactive wastes generated from the plant operation, volume reduction for the waste, improvement of the waste treatment facilities, operation and maintenance of the on-site waste storage facilities, and on-site transportation is included in the maintenance cost of the plant.

The Department of Nuclear Backend Management at the TPC headquarters is responsible for the planning and implementation of the radioactive waste disposal programs and future decommissioning of the TPC's nuclear power plant. The Radwaste Management Section of each nuclear power plant is responsible for the treatment and storage of the radioactive wastes generated from its own plant.

11.1.5 Financial Protection Program for Liability Claims Arising from Nuclear Accidents

The Nuclear Damage Compensation Law, enacted in 1971, as amended in 1977 and 1997, governs the financial protection program associated with a nuclear accident. It provides the financial and the legal framework to compensate those who suffered bodily injury or fatality and/or property damage as a result of the accidents at the nuclear facilities covered by this Law, including mainly the nuclear power plants.

The Nuclear Damage Compensation Law was enacted to meet two basic objectives:

- Remove the deterrent to the participation of both domestic and foreign private industries in the nuclear energy activities in this country presented by the threat of potentially enormous liability claims in the event of a catastrophic nuclear accident, and

- Ensure that adequate funds are available to the public to satisfy liability claims if such an accident should occur.

In enacting this Law, the AEC sought for the balance between the needs of the industry and those of the public. Specifically, this Law requires that all nuclear power plant licensees purchase specified amounts of liability insurance at a maximum level of NT＄4.2 billions or possess other equal financial protection against the risk of a nuclear accident. The total liability claims for one nuclear accident is limited by this Law at a maximum amount of NT$4.2 billion dollars. Should the amount received from the liability insurance or financial guarantee not suffice to cover the liability, the government shall loan the balance to the licensee to cover its complete liability, but only to the limited amount of NT$4.2 billions. However, the licensee shall indemnify the government for the loan.

The above financial protection, indemnification and liability limit applies not only to the liability of the licensee but also to the aggregate sum of all liability for all persons who may be held liable. This "omnibus coverage" effectively channels the financial responsibility for all damages up to the liability limit of the licensee. In so providing, this Law indemnifies the suppliers, contractors and others in the nuclear power industry as needed in the event of an accident, and assures the availability of reasonable compensation to the harmed persons no matter what caused the accident.

One important feature of this Law is, if a nuclear accident does occur, the claimants need only to prove that the accident did cause their injury/fatality or property damage in order to receive compensation for damages from any accident with significant offsite releases of radiation, i.e. an "extraordinary nuclear occurrence." No proof of fault is necessary, nor what caused the accident. Therefore, the public is significantly protected by this feature of the Law.

After the occurrence of a nuclear accident which caused damage to the public, the AEC will organize an Advisory Committee on Nuclear Accident Investigation and Evaluation to investigate the accident, find out its causes, evaluate the damage, and make recommendations to the governmental authorities about the compensation, recovery, and measures to improve the radiation protection in the nuclear installation.

11.2 Human Resources

11.2.1 Human Resources of the Licensee

11.2.1.1 Manpower of the TPC

The TPC is composed of more than 20 departments of general management, operations and business. Up to December 2009, the total number of employees in the TPC was about 26,900 and, among these, 2,650 were working in the nuclear sectors including the construction and operation of the nuclear power plants. The TPC has three nuclear power plants in operation, each with two units, with a total installed nuclear operating capacity of 5,144 MWe and one plant with two units under construction with a combined installed capacity of 2,700 MWe.

As shown in Figure 11.1, the nuclear-related organizations in the TPC headquarters include the Department of Nuclear Generation, Department of Nuclear Safety, Department of Nuclear Engineering, Department of Nuclear Backend Management, Department of Fuels, Department of Industrial Safety and Environmental Protection, Department of Nuclear and Fossil Power Projects, etc. There are also the Taiwan Power Research Institute, which is an affiliated research institute of the TPC, and the Taipower Institute of Training , which consists of three training centers in addition to the institute headquarter and is responsible for the training of the TPC employees.

The Committee of Nuclear Safety in the headquarters of the TPC is an organization for deliberating and decision-consulting on nuclear safety issues. To organize this Committee, the TPC invites experts from universities/colleges, research institutes, and industries as the Committee members in addition to those composed of the nuclear-related department directors. While in each nuclear power plant, there is a Station Operation Review Committee organized to advise the Plant General Manager on matters concerning nuclear safety.

As shown in Figure 9.1, in each nuclear power plant, there are various sections including, for example, the Operation, Mechanical Engineering, Electrical Engineering, Repairing, Nuclear Engineering, Health Physics, Environmental Protection Chemistry, Computer, Instrument & Control, Quality Control, Radwaste Management, Engineering Improvement, and General Affairs Sections as well as one Simulator Training Center.

The numbers of reactor operator license holders in December 2009 in each of the three operating nuclear power plants are shown in Table 11.2.

11.2.1.2 The Taipower Institute of Training

Under the Department of Human Resources of the TPC, there is a Taipower Institute of Training (TIT) responsible for the training of the TPC's employees. The TIT consists of three training centers as well as one headquarters. The training at the institute headquarters is mainly for on-the-job training of the TPC's management and professional personnel. On the other hand, the training conducted at the training centers is more techniques-oriented. Among these three centers, the Lin-kuo Nuclear Training Center is the place where the training on major maintenance techniques for the nuclear or fossil power plants is conducted.

In 1998, the TPC established a comprehensive Nuclear Power Maintenance Training Facility within the Lin-kuo Nuclear Training Center for the purpose of enhancing the development of technical manpower. This training facility is fully equipped with large mock-ups of various equipment and facilities, including spent fuel pool, fuel-loading facility, reactor coolant pump, and so on, to enhance the maintenance capability of the personnel.

Besides the Taipower Institute of Training, there is a plant training center at each nuclear power plant site, equipped with a full-scope simulator and small-scale mock-ups. These plant training centers are mainly for the training of reactor operators.

11.2.2 Other Human Resources

In addition to the existing manpower resources within the TPC, domestic supporting manpower for nuclear operational safety may come from research institutes, universities and the industries. The Institute of Nuclear Energy Research (INER) has about 1,000 formal employees and more than 300 contracted employees, which is an important technical supporting manpower pool to assure the operational safety nuclear power plant.

The National Tsing Hua University (NTHU) located at Hsinchu City, which is about 90 km south of Taipei, offers comprehensive undergraduate and graduate level education of nuclear engineering. NTHU is one of the top universities in Taiwan. The students admitted to the university are among the top 5% of the high school graduates. The undergraduate education of nuclear engineering is under the Department of Engineering and System Science (ESS) and the graduate program of nuclear engineering is under the Institute of Nuclear Engineering and Science (NES).

The Department of Engineering and System Science was founded in 1964, originally named the Department of Nuclear Engineering. The department offers Bachelor, Master, and Doctoral degree in engineering field. The department ensures multidisciplinary training in mechanical, electrical and material engineering, as well as the capability of system integration. To emphasize the importance of system integration in modern engineering and promote the diversity in teaching and research, the department has been renamed to its current name since 1997. In response to the renaissance of nuclear power around the world, the university established the Institute of Nuclear Engineering and Science at 2007.

The ESS Department has about 400 undergraduates, 140 and 100 Master and Ph.D. students, respectively. The NES Institute has about 50 and 20 Master and Ph.D. students, respectively. The Taiwan Power Company offers fellowship to students of ESS Department and NES Institute with major in nuclear engineering. The fellowship was established in 2004 and about 70 students were awarded with the fellowship up to this year.

The ESS Department has 25 full professors, 3 associate professors, and 4 assistant professors. Eighteen of them held Doctoral degree in Nuclear Engineering from famous universities around the world. NES Institute has 29 adjunct faculty members; 23 with ESS Department, 5 with BEES Department (Department of Bio-Engineering and Environmental Science) and 1 with Nuclear Science and Technology Development Center of NTHU.

For the past 40 years, number of students graduated from the Department: Bachelor 2500, Master 1330, and Ph.D. 122. A lot of graduates of the Department develop their career in the fields other than the traditional nuclear engineering. Nevertheless, the graduates of the Department constitute the major manpower in the development of nuclear power in Taiwan over past thirty years. Some major managing positions of nuclear institutions in Taiwan are held by the alumni of the Department.

Besides the NTHU, some other universities are getting interested in giving courses on nuclear engineering. For example, both the National Taipei University of Technology and the private Lunghwa University of Science and Technology gave nuclear related courses such as “Introduction to Nuclear Engineering” in 2009. In addition to this, a non-profit association in the private sector, the Nuclear Science and Technology Association (NuSTA) whose current manpower comes mainly from the retired specialists of the AEC and INER started a nuclear technology training program to give courses and lectures of nuclear technologies for the public and industries who are interested in understanding nuclear energy or obtaining more about the up-to-date information of nuclear technology.

11.2.3 Regulatory Requirements for Personnel Qualification, Training, and Retraining

The Atomic Energy Act of 1968 (as amended in 1971) in the item 3 of its Article 26 and the Enforcement Rules of this Act (Articles 38 to 43) stipulate that only the relevant license holders approved by the AEC can operate the nuclear reactor or handle the radioactive materials, radioisotopes, or machines that generate radiations. In early 2000, the Nuclear Reactor Facilities Regulation Act (2003), the Ionizing Radiation Protection Act (2002), and the Nuclear Materials and Radioactive Waste Management Act (2002) as well as their Enforcement Rules were promulgated.

The Nuclear Reactor Facilities Regulation Act of 2003 stipulates that without a construction license granted by the AEC, the construction of a nuclear reactor facility can not be started. After the completion of the construction work of a nuclear reactor facility, only if an operating license has been issued, can the facility be formally operated. Furthermore, to operate a nuclear power reactor, all the control room operators must have a reactor operator license in advance.

The Ionizing Radiation Protection Act of 2002 requires that to do a business of handling the radioactive materials, operating an ionizing radiation generating equipment, or conducting a radiation practice, one must have a certificate of permission or an approval for registration (hereafter in this section and the following subsections, the one with a certificate of permission or an approval for registration will be called the licensee). This Act further requires the licensee of these radiation related business must set up a radiation protection control organization or have the licensed radiation protection personnel in order to implement the radiation protection practice. To operate the radioactive materials or an ionizing radiation generating equipment, one must have a certificate of radiation safety, while to operate a production facility of radioactive materials, one must have an operator license.

11.2.4 Regulatory Requirements for the Reactor Operators

According to Appendix 1 of the Regulations on Nuclear Reactor Operators' Licenses (as amended in December 2009), the minimum number requirements for reactor operators on-duty in the main control room (MCR) of an operating nuclear power plant with twin units, which are also specified in the technical specifications of the FSAR, are as follows:

(1) Twin-Unit Plant with one Single MCR (e.g., the Kuosheng NPP)

- No unit in operation: 1 Shift Manager (SM) plus 2 ROs (with 1 RO for each unit),
- One unit in operation: 1 SM plus 1 Shift Leader (SL) plus 3 ROs (with 1 SL and 2 ROs for the operating unit and 1 RO for the shutdown unit),
- Two units in operation: 1 SM plus 1 SL plus 3 ROs.

(2) Twin-Unit Plant with Two MCRs (e.g., the Chinshan and Maanshan NPPs)

- No unit in operation : 1 SM plus 2 ROs (with 1 RO for each unit),

- One unit in operation : 1 SM plus 1 SL plus 3 ROs (with 1 SL and 2 ROs for the operating unit and 1 RO for the shutdown unit),

- Two units in operation: 1 SM plus 2 SLs plus 4 ROs.

As mentioned in Section 9.1 of this report, both the SM and SL are required to have valid SRO licenses.

In order to be qualified for applying the reactor operator's license, the candidate must be at least a high school graduate or the equivalent and satisfy the following requirements:

(1) Experience Requirements:

(i) At least 2-year working experience in a power station and among them at least 1 year in a nuclear power plant, and

(ii) At least 6-month working experience in the nuclear power plant which he is applying for the RO license and among them at least 3 months on duty for operation.

(2) Training Requirements:

(i) At least 1-year training including at least 3 months of simulator training, and

(ii) Completion of the required reactor operator training courses with no less than 300 hours.

(3) Physical Condition Requirements:

Passing the physical health check-up required.

The educational prerequisite for the application of an SRO license directly is that the applicant must be at least a college or university graduate. Other requirements include:

(1) Experience

He needs to have:

(i) at least 2-year working experience in a nuclear power plant, and

(ii) at least 6-month working experience in the nuclear power plant which he is applying for the SRO license and among them at least 3 months on duty for operation.

(2) Training

He needs to have:

(i) at least 1-year training including at least 3 months of simulator training, and

(ii) completion of the required reactor operator training courses with no less than 300 hours.

(3) Physical condition

He has to pass the physical health check-up required.

A reactor operator can also apply for an SRO license. In this case, he must have: (1) an RO license of the same nuclear power plant which he is applying for the SRO license and (2) at least 2-year reactor-operating experience, in addition to a qualified physical condition.

To obtain a reactor operator license, the candidate must pass the relevant regulatory examinations which will be held in two stages. Application for these regulatory examinations for the operator license must be submitted through the licensee of the nuclear power plant.

In the first stage, the examination is a written test about the fundamental theories including the components, reactor theory and thermal-hydraulics. After passing the first-stage test, then the license candidate can take the second-stage examination which consists of a written test about the plant characteristics and an operation test. The plant characteristics test covers the operation in emergency and abnormal conditions, plant systems, operation maneuvering, equipment control, radiation control, emergency response procedures and emergency response plan. The operation test comprises: (1) an individual operation on the simulator, (2) a team operation on the simulator, and (3) an oral, plant walk-through examination. Only if an applicant had successfully passed these two-stage examinations and was then successfully trained on-the-job for at least three months, a reactor operator's license can then be granted.

11.2.5 Licensee's Training Programs for Its Employees

The Nuclear Reactor Facilities Regulation Act of 2003 stipulate that only the reactor operator license holder approved by the AEC can operate a nuclear reactor. On the other hand, the Ionizing Radiation Protection Act (IRPA) of 2002 requires that to handle the radioactive materials or to operate an ionizing radiation generating equipment (e.g., the X-ray machine), the person has to have a radiation safety certificate and to operate a radioactive material production facility he must have an operation personnel certificate. The IRPA further requires that depending on the scope and business of a radiation practice the licensee must set up a radiation protection management organization or have radiation protection personnel.

The licenses and certificates required above are summarized as follows:

- License for senior reactor operator,
- License for reactor operator,
- Certificate for radiation protection personnel,
- Radiation safety certificate, and
- Operation personnel certificate.

Licenses for reactor operators and senior reactor operators are issued to applicants who have engaged in the relevant fields with sufficient experience and successfully passed the examination administered by the AEC. The total number of nuclear related license (or certificate) holders employed by the TPC to work in the NPPs as of December 2009 is 436, as shown in Table 11.2. At regular intervals, the holders of these licenses (or certificates) must take retraining programs conducted by the TPC for the specific types of licenses.

The Nuclear Reactor Facilities Regulation Act and its Enforcement Rules stipulate that the licensee of a nuclear power plant shall provide its employees with the educational and training opportunities. Accordingly, the TPC provides its employees with the opportunity of professional knowledge and technology update by the annual educational programs.

In general, the personnel technical training programs in the nuclear power plant can be categorized into the following areas:

- Reactor operator training,
- Training for holders of licenses other than reactor operators,
- Training for non-licensed plant technical staff, and
- General employee training.

In the following paragraphs, the training and re-training programs for the reactor operators will be further discussed.

11.2.5.1 Reactor Operators Training

Generally the AEC regulations such as the Regulations on Nuclear Reactor Operators' Licenses as amended in December 2009, the USNRC Regulatory Guide 1.8 "Personnel Selection and Training", and ANSI/ANS-3.1-1993 "Selection, Qualification, and Training of Personnel for Nuclear Power Stations," shall be followed for the training programs to the plant staff.

It is the responsibility of each NPP to select qualified employees to attend the reactor operators training program developed and conducted by the NPP itself. After completing this training program, the operator trainees have to pass the TPC in-house operator qualification examination to get their certificates of reactor operator candidates for applying the license. Then, these candidates have to pass the AEC operator license examinations to get their operator licenses. In accordance with the AEC regulations, a retraining program for the licensed reactor operators shall be developed and conducted by the NPP to assure that the licensed operators receive adequate, appropriate, and required training. To renew a reactor operator license, the license holder must fulfill the regulatory requirements of retraining.

The reactor operators/senior reactor operators (ROs/SROs) training program is developed and conducted in accordance with the AEC requirements of reactor operators and with reference to the content of the AEC examination for operator license, as described in the Appendix 3 of the Regulations on Nuclear Reactor Operators' Licenses (as amended in December 2009). The major training items are as follows:

(1) Classroom Training

(a) Fundamental Theory for Nuclear Power Reactors, including

- Theory of reactor operation,
- Design features of the reactor,
- Operational characteristics of the nuclear power reactor,
- Fundamental theory for core transient analysis (including heat transfer, thermodynamics, and fluid dynamics),
- Instrumentation and control (I&C) of a nuclear reactor,
- Radiation safety and control, and
- Nuclear regulations.

(b) Nuclear Power Reactor Systems, including

- I&C systems of the nuclear power reactor,
- Safety, emergency and fire-fighting systems,
- Mechanical systems of the primary and secondary sides,
- Electrical systems,
- Plant auxiliary and supporting systems,
- Fuel handling system,
- Waste treatment systems, and
- Overall plant operations and accidental transient response.

(2) Simulator Operation

- Normal and emergency operating procedures,
- Operational transients,
- Judgment and analysis of an accident,
- Operations at normal, abnormal and emergency conditions of the unit, and
- Changeover of the operation shift.

(3) Plant Observation (or Plant Walkdown) Training

- Designs and layouts of major equipment and components,

- Functions and operational characteristics of equipment,
- Remedies to recover from the malfunctions of equipment and components,
- Implementation of special operating procedures,
- Radiation safety and protection,
- Familiarization of and operation on the main control room panels,
- Responsibilities of the reactor operator and senior reactor operator,
- Fuel management and control,
- Remote shutdown operation, and
- Administration control processes.

11.2.5.2 Licensed Reactor Operator Retraining Program

The Regulations on Nuclear Reactor Operators' Licenses (as amended in 2009) require, in its Article 10, the licensee of a nuclear power reactor should submit a retraining program based on the Appendix 4 of this Regulation for his reactor operators to the AEC for approval. The retraining guidelines as mentioned in the Appendix 4 of this Regulation require that the retraining program should contain a training of at least 90 hours on courses in the classroom and 30 hours on simulator every year. All reactor operators should be retrained according to the approved retraining program and must pass the licensee's annual re-qualification tests including written and oral examinations in order to be legally qualified to continue their operation of the reactor.

The license of a reactor operator is valid for a period of 6 years. To apply for the renewal of a reactor operator license, the licensee of the nuclear power plant has to submit the following documents about the operator to the AEC 30 days before the expiration date of the license:

- Certificate of qualifying in the physical health examination and the examination report, and
- Retraining records and certificate for passing the retraining examination.

The reactor operator retraining should include the following contents: classroom lectures and operation on simulators as well as on the plant site. Details of these contents are listed in the Appendix 4 of the above-mentioned Regulation.

(1) Classroom Lectures

The classroom lectures training should be at least 90 hours annually and is divided into two parts: fixed courses and adjustable courses. The contents of the lectures should take into consideration the fundamentals and operational proficiency topics.

(a) Fixed Courses

Training on fixed courses must be over 60 hours every year. It includes the fundamentals and the skillfulness of operation. The scope of these training courses must be covered within two years.

(b) Adjustable Courses

Training on adjustable courses in combination with that on fixed courses must be over 90 hours every year. These adjustable courses are mainly operation-related, such as the feedback of operating experiences.

(2) Operations on the Simulator and on the Plant Site

Training on simulator must be no less than 30 hours every year. The scope of these training courses is as follows:

(a) Exercise on normal plant operations,

(b) Exercise on abnormal conditions of the nuclear steam supply systems,

(c) Exercise on abnormal conditions of the balance of plant systems, and

(d) Emergency events that challenge the critical safety functions.

The TPC conducts its annual retraining program in accordance with the regulations' requirements for its licensed reactor operators who are rotating in a six-group, three-shift system to maintain the proficiency of plant operation skill. The retraining program is conducted by the plant itself on a regular and continuing basis. Mechanisms was established to assure the licensed operators remain cognizant of changes to the facility, procedures, governmental regulations, and quality assurance requirements, as well as the industry operating experience, licensee's event reports (LERs), and human errors as applicable to their area of responsibility.

11.2.5.3 Training for Non-Licensed Plant Technical Staff

Non-licensed plant staff include:

(1) Non-licensed on-shift operators of plant system/equipment radwaste control and process system/equipment switchyard, pump house, gas turbine, etc.,

(2) Each category of maintenance and engineering support engineers such as mechanical maintenance engineers, electrical maintenance engineers, instrumentation and control engineers, nuclear engineers, chemistry/radiochemistry engineers, health physics engineers, quality assurance and quality control engineers, and computer engineers, and

(3) Each category of maintenance and engineering support technicians for maintenance and engineering support, such as mechanical maintenance technicians, electrical maintenance technicians, instrumentation and control technicians, chemistry/radiochemistry technicians, health physics technicians, and quality assurance and quality control technicians.

The training for the non-licensed operators, the engineers and the technicians for maintenance and support must be no less than 40, 30 and 15 hours, respectively, every year.

Initial and continuing training programs shall be implemented for the non-licensed personnel to assure that they are qualified for the job. This is achieved by using a systematic approach to training (SAT) method, which is a performance-based method containing elements such as analysis, design, development, implementation, and evaluation. The training programs shall be developed after determining the job performance requirements through the process of the job and task analysis for the personnel of each category. The training program shall be updated to reflect the results of program evaluations, changes of regulations, changes in the facility, and lessons learned from the industry experiences. A system for periodic review of initial and continuing training programs was established to assess the instruction and program effectiveness in helping trainees to meet performance requirements. The training programs for the non-licensed NPP technical staff are as follows.

(1) Initial Training

For every category of plant personnel, an initial training (or the so-called pre-job training) program shall be established to develop or enhance the skills, knowledge, and ability of personnel to perform their job assignments. The initial training programs are developed for individuals with entry-level qualifications. Some individuals can be exempted from that specific training based on their prior education, experience, and training.

(2) Continuing Training

For every category of plant personnel, continuing training programs shall be implemented to maintain and enhance their proficiency of the plant. These programs shall include the following topics which are important to the employees' performance:

- Significant plant system and component changes,
- Applicable procedure changes,
- Applicable industry operating experiences,
- Selected fundamentals with emphasis on knowledge and skill necessary to nuclear safety, and
- Other training needed to correct performance problems of the position incumbent.

The continuing training programs will also include provision for retraining that maintain the proficiency of skills and knowledge required for acceptable performance. Mechanisms will be established to assure that individuals in the NPP who perform safety-related functions remain cognizant of changes to the facility, procedures, governmental regulations, quality assurance requirements as well as industry operating experiences, and personnel applicable to their area of responsibility.

(3) BWR/PWR/ABWR System Technology Training

According to the reactor type of the plant in which they are working, all non-licensed plant staff including engineers and technicians shall also take BWR, PWR or ABWR system technology training with different training period as required. This training will be conducted by the NPP itself, and the training material will be in Chinese and developed by the staff who have completed the manufacturer's system technology training.

11.2.5.4 General Employee Training

All persons employed in the NPP and the people hired by the TPC contractors who access the NPP to work shall be trained in the following areas commensurate with their job duties:

- General description of plant and facilities,
- Job related policies, procedures, and instructions,
- Radiological health and safety programs,
- Station emergency plans,
- Industrial safety program,
- Fire protection program, and
- Security program.

11.2.6 Regulatory Related Training

The AEC provides its staff with a systematic training program to maintain their professional capabilities up-to-date as to meet the ever-increasing regulatory challenges. For example, a course of 12 or 24 weeks on BWR or PWR technology and simulator training is a pre-requisite for a resident inspector at the nuclear power plant. Following this, a training on the advanced technology is then required in order to enhance the capability of the inspectors. In addition, regularly several selected staff members of the AEC are dispatched to the overseas regulatory agencies and/or research institutes in the nuclear advanced countries for on-the-job training. Also, the AEC regularly trains its inspectors and reviewers with the TPC's design change requests (DCRs) which had been implemented or were planned as the classroom discussion topics.

In May 2005, the AEC internally required its inspectors and RO/SRO examiners to be licensed in order to strengthen their technical ability and to further enhance the quality of inspections and reactor operator examinations. There are two kinds of inspector licenses: Inspector and Senior Inspector. Qualifications for being a candidate of the inspector are:

- College graduate with a major in engineering or the equivalent,
- Classroom training on PWR, BWR or ABWR systems for more than 2 weeks,
- Fundamental training for nuclear power reactor inspector for 4 days, and

- Self-reading the assigned on-the-job training courses for an inspector.

On the other hand, qualifications for one to become a candidate of the senior inspector are:

- Being an inspector for more than 6 years,
- Classroom training on PWR, BWR or ABWR systems for more than 4 weeks,
- Training on the relevant simulator for more than 2 weeks,
- Advanced training for nuclear power reactor inspector for 4 days,
- Training on quality assurance for 5 days,
- Training on the practical applications of probabilistic risk assessments (PRA) for 3 days,
- Performance on human relationship and effective communications, and
- Self-reading the assigned on-the-job training courses for a senior inspector.

Both licensed inspectors and senior inspectors are required to take the retraining courses on systems and simulators for more than 30 hours every 2 years. These licenses are effective for 6 years.

Qualifications for a candidate of the RO examiner are as follows:

- Being a senior inspector or qualified for a senior inspector, and
- Having been trained on and pass the specialist training for the RO examiner.

Table 11.1 Number of Design Change Requests Completed in 2005 - 2009

	CS0	CS1	CS2	KS0	KS1	KS2	MS0	MS1	MS2
2005	6	38	38	35	24	58	36	62	132
2006	14	14	40	20	62	17	26	102	103
2007	3	37	12	30	64	71	28	96	38
2008	11	43	34	31	16	13	20	26	72
2009	7	12	55	44	76	73	21	62	65

Note: 0: common for units 1 and 2
1: unit 1 only
2: unit 2 only
CS: Chinshan Nuclear Power Plant
KS: Kuosheng Nuclear Power Plant
MS: Maanshan Nuclear Power Plant

Table 11.2 Number of Nuclear-Related License (or Certificate) Holders Employed by the TPC

As of December 2009

Type of License/Certificate	Chinshan	Kuosheng	Maanshan
Senior Reactor Operator	30	34	29
Reactor Operator	24	26	19
Radiation Protection Personnel	32	41	32
Radiation Safety	14	75	80
Subtotal	100	176	160
Total	436		

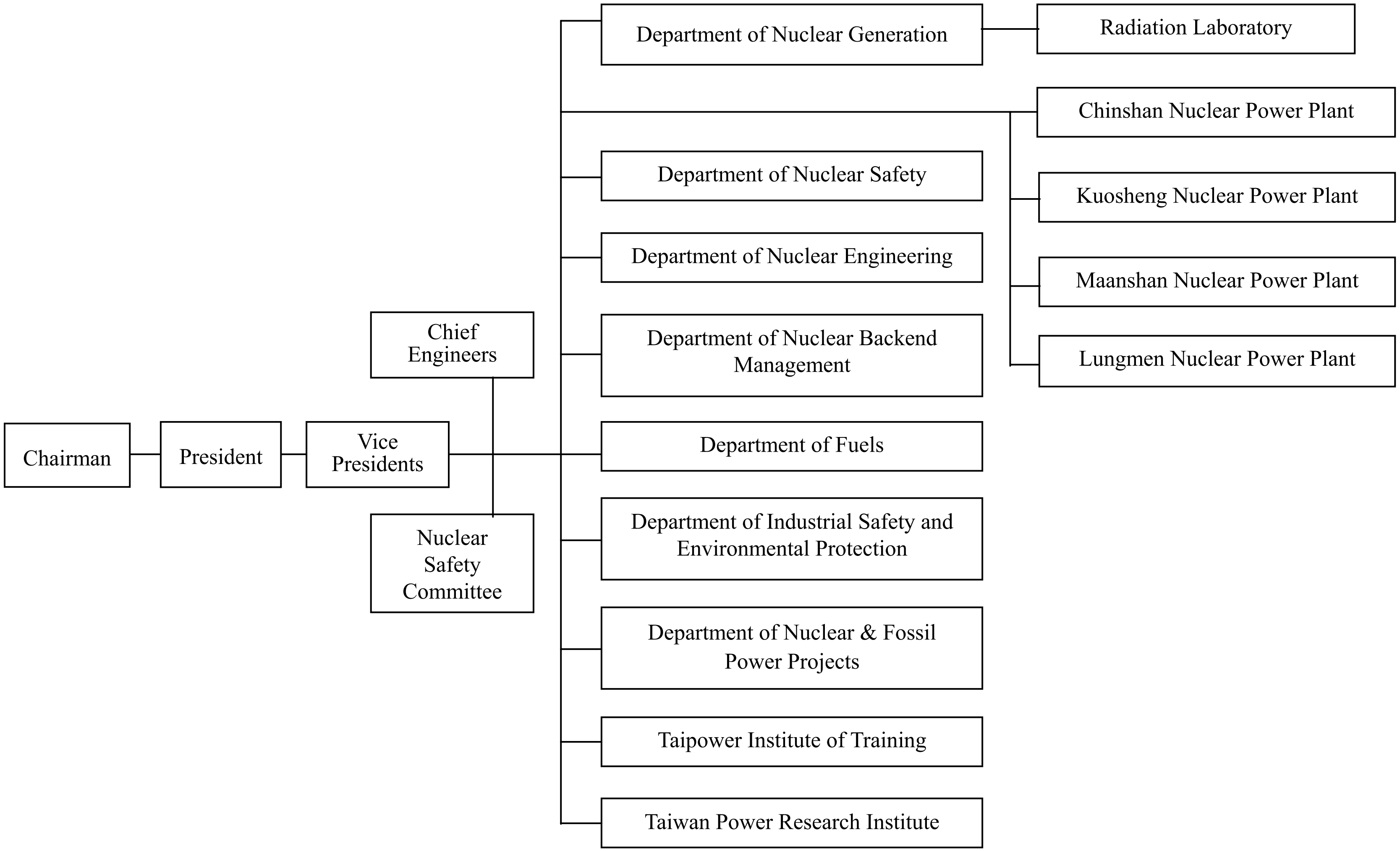


Figure 11.1 Organization Chart of the Nuclear Sector in the TPC

ARTICLE 12. HUMAN FACTOR

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.

12.1 Methods to Prevent, Detect, and Correct Human Errors

The ways carried out by the nuclear communities in this nation to prevent, detect and correct the human errors are as follows:

(1) In order to prevent the occurrence of abnormal events due to human errors, the AEC requires that human factors and man-machine interface have to be taken into account in the system and equipment designs of nuclear power stations. At the construction stage, the AEC examines whether the plant construction and equipment installation meet the requirements for the consideration of human errors in the preliminary safety analyses report (PSAR). During plant operation, the AEC oversees the human performance through site inspections, safety reviews, and regulatory meetings.

(2) To minimize the misjudgment of and the erroneous operation by reactor operators, the TPC continuously carried out long-term and short-term training programs for the operators. An operator-to-be trainee needs to learn the basic knowledge of nuclear installation through the in-house training curriculum, followed by operating practice with the full-scope main control room (MCR) simulator. (In Taiwan, every nuclear power stations has its own simulator). After passing all the examinations associated with these training courses, the trainee will become a candidate of the reactor operator (RO) and be assigned to an operating shift of the related nuclear power plant (NPP) for the on-the-job training under the guidance and supervision of a senior reactor operator (SRO). The AEC will grant an operator license to the trainee if he passes the the AEC's written examination and operating test for the reactor operator. For those licensed operators, regular re-training courses are arranged to maintain their capabilities of dealing with normal and abnormal operating conditions.

(3) In order to maintain the quality of maintenance works, the TPC has established a Maintenance Training Center for the training of its plant maintenance staff and workers from the contractors regularly. The maintenance personnel are trained according to their levels of knowledge and skill. The training courses include the basic principles, mock-up training, on-the-job training, and the experience feedback seminars.

(4) For the purpose of reducing human errors, ten preventive measures are reiterated in the TPC's safety culture enhancement program, in which the operating experiences are regarded to actively prevent occurrence of repeated events in domestic NPPs. Through this practice, lessons are leaned from such documents as General Electric Service Information Letter, Westinghouse Technical Bulletins, and information from BWROG, WOG, INPO/WANO Networks, NRC bulletin, and the TPC's Reportable Event Reports (RER). Through the process of event screening, evaluation, and analysis, the conclusions will be shared by the applicable TPC nuclear power plants

via an operating experience (OE) feedback system. The relevant nuclear power plants will follow the documents and reflect countermeasures into plant procedures, training, or equipment conditions. The WANO and INPO documents; such as significant operating experience reports (SOERs) and significant event reports (SERs) are respected as essential sources in the learning of international operating experiences. In particular, identified root causes, relevant corrective actions and recommendations given by the SOERs and SERs are very valuable and will be used by the plant operators to take the advantage of them for event prevention beforehand.

(5) Ten preventive measures shown as follows to reduce human errors were implemented:

- Double check,
- Potential risk evaluation,
- Tool box meeting,
- Self checking,
- Adherence to procedure,
- Conservative decision making,
- Enforcing the coordination within the operation and maintenance (O&M) Group,
- Reducing the human errors of vendors and contractors,
- Experience feedback and training, and
- Root cause analysis of human error type events.

These preventive measures form the “barriers for the prevention of human errors” as shown in Figure 12.1.

(6) To prevent the occurrence of a severe accident, emergency operating procedures (EOPs) of the three operating NPPs were established by the TPC. Furthermore, the severe accident management guidelines (SAMGs) of the three operating NPPs were also developed in 2003 for the accident management team (AMT) to mitigate the severe accident. The corresponding training on both EOPs and SAMGs are performed to reduce human errors. To minimize misjudgment and erroneous operation by the AMT, the TPC has developed a severe accident engineering simulation code (a TPC version of MAAP4) for training purpose. The AEC will also audit/inspect the associated performance via emergency preparedness drills.

(7) In order to evaluate the plant safety, the INER and the TPC have collaboratively developed the living probabilistic risk assessment (PRA) models on all three operating NPPs in 1996. The human reliability analysis (HRA) is an important issue within these models. According to such factors as man-machine interface, complexity of task, working environment, stress, timing, training, procedure,

experience, etc., the HRA is adopted to evaluate the human error probabilities (HEPs) for the human actions defined by the model analyst. The HEPs include the incorrect-calibration probability of the instruments, the misalignment probability of flow paths, and the mitigation actions after a postulated event. The findings from the HEP assessment are also reflected in the associated training courses and found very valuable for the actual reduction of human errors in plant operation.

(8) For contractors, the TPC has implemented preventive measures as follows to ensure the safety of contracting works.

(a) Plant Orientation Training – Workers hired by the contractors should take the following training courses before entering the plant to do their jobs as requested by the plant procedures:

(i) Security and entrance control,

(ii) Industrial safety and sanitation,

(iii) Radiation protection,

(iv) Environmental and radioactive waste management,

(v) Quality control, and

(vi) Emergency plan.

Contracted workers have to pass the test of the training before qualified for the work. The qualification is valid for one year.

(b) Pre-job training

Basically, the training will be based on the trainee's work scope to setup the courses and usually includes a mock up training.

(c) Trainee's qualification and license confirmation required

(d) Onsite management

The contractor should assign a foreman with engineering and management experiences to supervise on-site.

(e) Self verification and experience feedback

(f) Evaluation of the contracted work by the TPC

(g) Penalty and warranty terms

(9) Alcohol and Drug Test for Nuclear Reactor Operators were implemented. To prevent the occurrence of human errors caused by the alcohol and illegal drugs, the AEC has set up related requirements in "Regulations on Nuclear Reactor Operator's License". According to these regulations, random alcohol and drug tests will be performed for nuclear reactor operators on duty. The license of an operator will be suspended for 3

to 18 months if he fails to pass the alcohol and/or drug test. Should a reactor operator not pass the drug or alcohol test for the 2nd time, his license will be terminated.

(10) Human–System Interface (HSI) design was adopted in the Lungmen Project. The Lungmen Project is the 4th nuclear power plant under construction in this country. Two advance boiling water reactors (ABWR) are to be installed at this plant site. According to the final safety analyses report (FSAR) of this plant, the primary goal for the HSI design is to facilitate safe, efficient, and reliable operator performance during all phases of normal plant operation, abnormal events, and accident conditions. To achieve this goal, information, display, controls, and other interface devices in the control room and other plant areas are designed and implemented in a manner consistent with good human factor engineering (HFE) practices. An HFE design and implementation process contains four stages, i.e. planning, analysis, design and verification & validation (V&V). In the planning stage, the major task is the HFE program management. In the analysis stage, the major tasks include the operating experience review, functional requirement analysis and function allocation, task analysis, staffing, and human reliability analysis. In the design stage, the human-system interface design, procedure development, and training program development are important elements. Finally, the human factors will be verified and validated in the V&V stage. It is believed that, by adopting this HSI design, an efficient and reliable operation will be enhanced through application of the automated operation capabilities of the ABWR.

12.2 Managerial and Organizational Issues

In order to make sure that the managerial and organizational aspects of a nuclear power plant are properly addressed, the AEC requires the plant owner to describe the personnel organizations including reactor operators, maintenance personnel, and administrative staff in the PSAR and FSAR. This requirement is enacted in the Enforcement Rules of Nuclear Reactor Facilities Regulation Act. The TPC has to operate the nuclear power stations according to the organizations approved by the AEC. In addition, to minimize human errors of the reactor operators by reducing their workload and consolidating the educational and training programs, the TPC changed the reactor operator’s working system of all nuclear installations from a five-group three-shift scheme to a six-group three-shift scheme. In a six-group three-shift working system, three groups rotate for the reactor operation, while the other three groups will take training courses, day-offs, or routine works, respectively. Routine works may include evaluations and surveillance tests for the safety-related systems.

Whenever a human error event occurs, the plant operator needs to work out a human performance enhancement system (HPES) report and hold a system diagnostic meeting to find out which barrier for preventing human errors has been broken. Corrective actions will then be determined from the conclusions. Several R&D programs were performed to study ways of improving managerial and organizational aspects of a nuclear power station. For example, a systematic methodology was developed for evaluating the working procedures of the outage management and corrective maintenance in three nuclear power stations of the TPC in 1995. The major findings and recommendations of the evaluation were:

(1) Details of the outage working schedule and large boundary isolations were needed.

(2) Additional system coordinator was found necessary to enhance the communications among working groups.

(3) “Maintenance Management Computerization System” needs to be improved with respect to the short-term and long-term scheduling.

(4) Coordination and communication among working groups need to be strengthened for corrective and efficient maintenance.

(5) Root causes analysis, equipment failure analysis, and feedback of work experiences need to be emphasized.

The project for the study of improved maintenance performance has been accomplished in 1996. This study identified a set of ten O&M factors as shown in Table 12.1 that were judged to have influences on the maintenance of the nuclear power stations. Questionnaires related to these factors were solicited from 35 managers and 372 engineers on the sites. The results, somewhat to be surprised, showed that substantial differences existed in the choice of factors and their relative importance to the maintenance performance between the group of managers and the group of engineers. Therefore, the plant managers were recommended to pay more attention to those various areas and to foster a more consistent perception among plant employees. The TPC adopted these recommendations, as well as those resulted from the other R&D programs, and made corresponding corrective actions to improve human performance in its nuclear plants.

On the other hand, to help the rookies catch up with the pace of plant operation, the TPC has developed a “Mentor System” in each existing nuclear power plant. In addition to learn different techniques from different instructors, a newcomer of a plant has a senior plant staff with superior technique and excellent personal character assigned as his mentor. This mentor is responsible for the daily life caring, behavior instructing, problem consulting as well as performance evaluating of the newcomer. All newcomers have to submit progress reports describing lessons learned and progress of training to their superiors bimonthly. These reports have to be commented by the newcomers’ mentors before submission. All newcomers also have to deliver oral reports reviewing the training results to their superior every 6 months. It is expected that the efficiency and performance of the rookie training will be highly enhanced through this mentor system and human errors associated with the rookies will be reduced as well.

Furthermore, the TPC has developed a computerized recording system for plant patrol. A patrolman downloads all relevant checklists from a main server to his personal digital assistant (PDA) before he performs the patrol task. Then he records all system conditions by using his PDA during patrol. Finally, he uploads all the results of the patrol from his PDA into the main server. This system has the following advantages over the manual recording system:

- Probabilities of human errors associated with the patrol task were reduced.
- Records of the patrol were easier to store and to search.
- Space needed for storing the patrol records was reduced.

- Quantity of paper utilized was reduced which helped the environmental protection.

- Trend analysis of the patrol records was much easier than before.

- By the hi-low alarm and historical data stored in his PDA, a patrolman has much more information to do his job than using manual recording system.

12.3 Role of the Regulatory Body and the Facility Operator

12.3.1 Role of the Regulatory Body

The importance of human behavior in ensuring the safety of nuclear installations has been revealed in the accidents at TMI and Chernobyl. To prevent the occurrence of human errors in the nuclear power station, the AEC requires the TPC to include human factors in the stages of planning, design, construction, and operation of a nuclear power plant. Through the reviews of PSAR and FSAR, the AEC conducts safety examination associated with human engineering design. By way of plant inspections, the AEC ensures that all designs related to human factors are constructed according to the safety analysis reports. In the stage of operation, the AEC checks the human performance through resident inspection, outage inspection, regulatory meetings, and so on. To enhance the human performance, the AEC conducts a lot of special regulatory activities, such as the enforcement of incorporating the post-TMI actions to all the TPC's operating nuclear power plants so as to prevent the occurrence of similar human errors.

12.3.2 Role of the Facility Operator

To keep good human performance in the nuclear power plants, the TPC plays a key role in the prevention, detection, and correction of human errors. The AEC's requirements associated with the human factors are the baselines for the TPC to follow. In addition, the TPC spent a lot of efforts to prevent human errors and improve human performance. These efforts include fostering safety culture, preparing and revising operational manuals, better training of operators and maintenance personnel, and performing related R&D programs. The details of these efforts are described in the previous Sections 12.1 and 12.2.

Table 12.1 O&M Factors with Impacts on Nuclear Power Plant Maintenance

O&M Factors	Definition
Coordination of Work	Planning, integration, and implementation of maintenance work.
Learning and Experience Feedback	The manner how the plant encourages personnel to use knowledge, experience, and updated information to identify problems and propose improvement of maintenance work.
Training	The extent to which plant personnel are provided with the required knowledge and skills to effectively perform maintenance works. It also refers to personnel perceptions regarding the general usefulness of the training programs.
Formalization	The extent to which there are well-defined rules, procedures, and/or standardized methods for routine activities as well as unexpected occurrences.
Ownership	The degree to which plant personnel take the responsibilities and the consequences of their actions. It also includes the commitment to and the pride of the organization.
Resource Allocation	The manner in which the plant distributes its manpower and financial resources, including the actual distribution as well as the individual perceptions of this distribution.
Personnel Selection	The extent to which plant personnel are identified with the requisite knowledge, experience, skills, and ability to perform a given job.
Responsibility of Individuals	The extent to which plant personnel and departmental work activities are reasonably divided and matched.
Performance Evaluation	The extent to which plant personnel are provided with fair assessments of their work-related behaviors, including regular feedbacks with emphasis on future improvements.
Goal Recognition	The extent to which plant personnel get involved, understand, accept and agree with the cause and the purpose of the maintenance works.

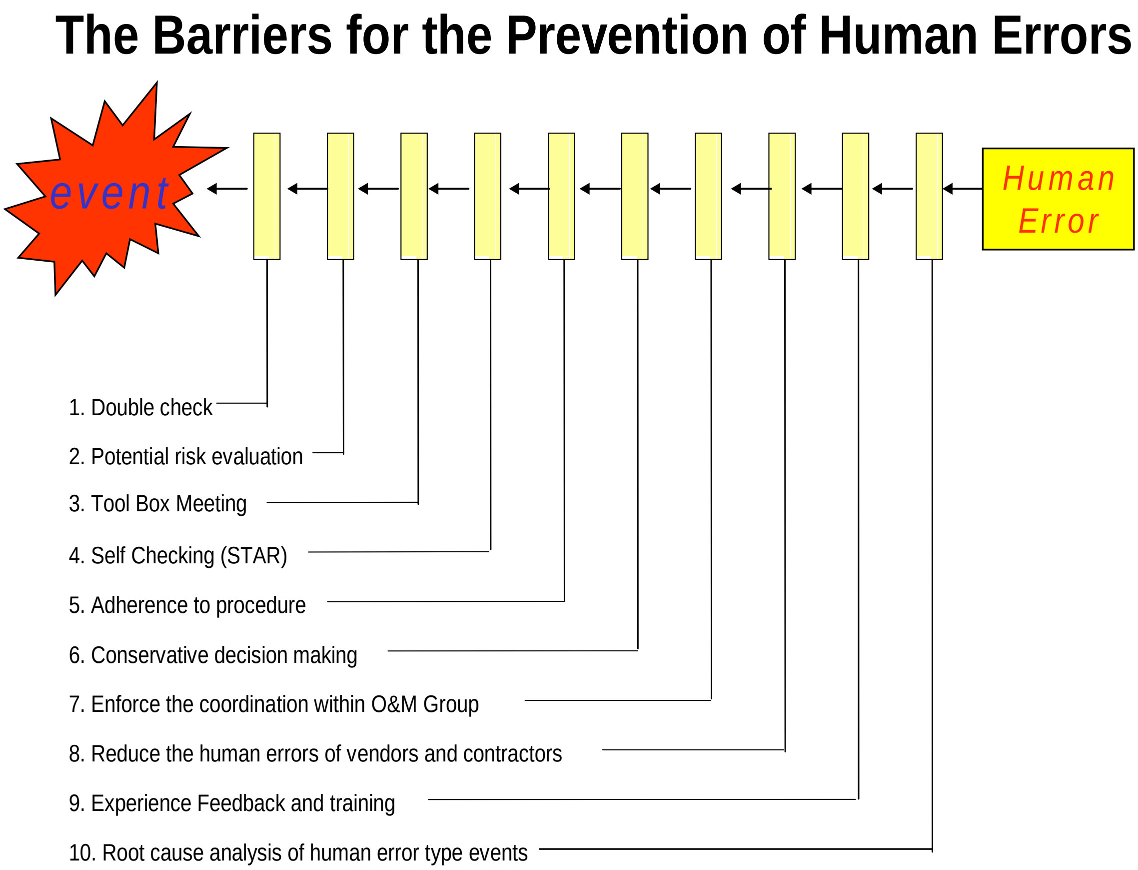


Figure 12.1 Barriers for the Prevention of Human Errors

ARTICLE 13. QUALITY ASSURANCE

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

13.1 Quality Assurance Programs

This chapter describes the quality assurance (QA) policy, requirements and programs which are implemented for the nuclear power plants in stages of design, procurement, manufacturing, construction, commissioning, operation and maintenance.

Each applicant for a construction license (or the so-called “construction permit” previously) of a nuclear power plant is required by the Nuclear Reactor Facilities Regulation Act to describe the quality assurance related activities including design, construction, and inspection programs in its PSAR. A construction license will be issued after the PSAR is reviewed and approved by the AEC. To verify the implementation of the QA program during the design and the construction stages, the AEC will perform onsite inspections in accordance with Article 12 of the Enforcement Rules for the Implementation of the Nuclear Reactor Facilities Regulation Act.

A policy statement for the quality assurance is submitted as part of the PSAR and FSAR to the AEC for review. Based on what had been provided in the total quality management policies, the TPC established its QA policy statements as follows:

(1) A total quality management system shall be established based on the national or international standard. The total quality management shall be undertaken with continuous improvement activities to enhance the service quality for the customer’s satisfaction.

(2) In addition to the policy described above, nuclear safety-related items and activities shall also be implemented based on a nuclear quality assurance program in accordance with the requirements of the Nuclear Reactor Facilities Regulation Act. All commitments to the regulation shall be fulfilled to assure the nuclear safety and public health.

Specifically, a nuclear project QA program is established before a nuclear facility is to be built. In the meantime, a nuclear operation QA program is established for the safe operation of a licensed nuclear power facility. Both QA programs adopted Appendix B to the 10 CFR 50 of U.S.A.

For a nuclear facility to be built, a Nuclear Projects QA Program shall be established first. This program applies to safety-related items and their associated activities from planning, design, procurement, fabrication, and construction, to preoperational testing for all new projects, as well as any specifically nuclear related works.

In the Lungmen project, which is under construction, all safety related structures, systems and components shall meet the quality requirements of the AEC as well as that of the country of origin. Currently, the requirements of the country of origin include 10

CFR 50 Appendix B and ANSI N45.2. In addition, the QA program for the Lungmen project meets the applicable United States Nuclear Regulatory Commission (USNRC) regulations and the ASME NQA-1 and NQA-1a requirements. Appropriate and practical quality requirements such as that in the ISO-9001 program will be applied to non-safety yet reliability-related structures, systems and components to meet the TPC's requirements of a total quality management program.

For each nuclear facility with an operating license, a nuclear operation QA program is established by the Department of Nuclear Safety of the TPC. All commitments made in the FSAR and other licensing application documents shall be strictly followed to assure the nuclear safety and public health.

13.2 Implementation and Assessment of Quality Assurance Programs

The implementation of the QA program is to assure the quality of the projects and to provide a solid foundation for nuclear safety and reliable power generation. To achieve the quality necessary for safety, the TPC employs the following management actions for its nuclear power stations:

- Develop and maintain an effective QA program,
- Audit and assess the effectiveness of the QA program, and
- Provide feedbacks to the management on quality of performance.

During the design and construction stages, QA activities are performed by the licensee, the reactor vendor, the balance of plant vendor, the component and equipment suppliers and various subcontractors. It is the licensee's responsibility to establish the QA program and to maintain the overall effectiveness of it. For the Lungmen Project, the TPC implements its QA program and at the same time supervises the implementation of it into the plant constructor's QA program. The latter will be through plant constructor's standard procedures supplemented with approved Lungmen project procedures and procedures addressing uniquely the TPC requirements.

Many activities are taken by the AEC to monitor the conformance of the construction activities to the quality requirements. These activities include mainly the resident inspection, periodic inspection, special taskforce inspection and examination of the key holding points for the construction. A more detailed description of the implementation of inspections is provided in Subsection 14.2.1 of this report.

In order to further improve the quality of procurement, the TPC has joined the Nuclear Procurement Issues Committee (NUPIC) since 2005. This committee was founded in 1989 and is represented by 33 US members and 12 international members recently . The NUPIC provides effective programs for the evaluation of suppliers furnishing safety related items and services to the nuclear industry. These programs are performed through Joint Audits and Surveys with cooperative efforts of the NUPIC members. The quality of procurement has been greatly improved for the members of NUPIC. As a member of NUPIC, the TPC has obligations to set up audit process and joint NUPIC auditing activities. the TPC also attends conferences held by the NUPIC and collects information about qualities of suppliers periodically. A data bank of information of suppliers in nuclear fields is set up on the intranet of the TPC. This data bank includes background information, record of quality assurance, experience of utilization, and internal review results of all suppliers for safety-related items in nuclear industry. The following advantages have been found by using this data bank:

- Complete and correct information of suppliers can easily be obtained.
- Quality of safety-related items can be assured.
- Reliability of safety-related items can be increased.
- Cost of procurement for safety-related items can be reduced.
- Time of procurement for safety-related items can be shortened.

13.3 Configuration Management

13.3.1 Purpose of Configuration Management

Configuration management is an essential tool for managing high quality engineering activities and many requirements contained in the quality assurance plan are closely related to it. The purpose of configuration management is to ensure the structure, system, component, and computer software are in compliance with the predetermined design requirements and to assure the physical and functional characteristics of a nuclear power plant are correctly incorporated in the appropriate documents. The configuration management plan of the Lungmen project, first time for the TPC's nuclear power plants, was based on the INPO 87-006 report. A computerized Information Management System (IMS) has been established to perform the document as well as the modification and change control during the stages of design, procurement, manufacturing, construction, commissioning, operation, and maintenance. Through this IMS, the required document and information can be quickly and correctly retrieved.

13.3.2 Configuration Management Plan

All principal vendors of the Lungmen project, including vendors of reactor, architecture engineering, turbine generator, and radioactive waste system, are required to establish their respective configuration management plans. The contents of the configuration management plan include:

(1) Purpose,

(2) Scope,

(3) Framework of configuration management,

(4) Design bases,

(5) Design and design change control,

(6) Design document control,

(7) Evaluation of the configuration management process,

(8) Interfaces and integration,

(9) Working procedures for configuration management plan, and

(10) Computer codes for design and configuration management.

A Configuration Management Program Procedure for the whole plant has been established. This procedure shall be followed by all the vendors during the process of the design, design document management and design change control. It should also be followed during the periodic update to maintain the design documents in the most current condition.

13.3.3 Design Documents in Configuration Management

The configuration management of the Lungmen project includes the design documents of NSSS, BOP, and related systems, equipment and services. The design documents consist of function requirements, design bases, design criteria, system design description, specification (including technical procurement specifications), manuals, drawings, interface requirements, design changes, etc. The preservation, maintenance, and integration should be performed in the stages of design, procurement, manufacturing, construction, commissioning, operation, and maintenance to ensure that the integrity of the documents can be maintained throughout the lifetime of a nuclear installation.

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 nuclear installations continues to be in assurance with its design, applicable national safety requirements, and operational limits and conditions.

14.1 Ensuring Safety Assessment throughout Plant Life

This section describes the documents and processes to ensure that systematic safety assessments are carried out during the lifetime of the nuclear installation.

14.1.1 Safety Assessment before Operation Stage

The licensing procedures for nuclear installations, pursuant to the Nuclear Reactor Facilities Regulation Act and described in Articles 5 and 6 of this Act, consist of two steps: applications for the construction license and the operating license. The applicant for a construction license or an operating license shall conduct comprehensive and systematic safety assessments to ensure that the public and environment are protected from radiation hazards which may accompany the operation of nuclear installations. The results of the assessments are documented into two reports, namely, the preliminary safety analysis report (PSAR) and the final safety analysis report (FSAR). Both of them need to be reviewed and approved by the AEC. In addition to these two reports, the applicant must also submit an environmental impact assessment (EIA) report to the Environmental Protection Administration (EPA) in order to fulfill the licensing requirements. More detailed descriptions of the requirements for the environmental impact assessment are provided in Subsection 17.2.1 of this report.

The “Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities” enacted to be pursuant to Article 5 of the Nuclear Reactor Facilities Regulation Act, describe the required contents in the PSAR. Similarly, the “Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities” describe the required contents in the FSAR. Because the content can be covered by the standard safety analysis report (SAR) of the country of origin of the supplier of nuclear steam supply systems (NSSS), the contents of PSAR and FSAR for Chinshan, Kuosheng and Maanshan Nuclear Power Plants are essentially the same as that required in the country of origin. For the Lungmen Nuclear Power Plant (LNPP), the contents of PSAR are based on that of the standard SAR of the ABWR. However, two more chapters and five more appendices are added due to the requirements of the AEC. These added contents are:

- Chapter 18 : Human Factors Engineering,
- Chapter 19 : Severe Accident Analysis,
- Appendix A : Probabilistic Risk Assessment,
- Appendix B : Integrated Reliability Analysis,
- Appendix C : Emergency Planning,
- Appendix D : Backend Program, and
- Appendix E: Experience Feedback.

The chapter of human factors engineering describes the human-system interface (HSI) design development, the HSI design goals and bases, the standard HSI design features, and the HSI design and implementation process, whatever applicable to the Lungmen project.

The severe accident analysis is performed to show that the regulatory requirements and the severe accident policy established by the USNRC for advanced LWRs can be met. The probabilistic risk assessments (PRA) of the LNPP showed that the plant had a significant capability to prevent such accidents and to mitigate their consequences.

A Level-3 PRA has been performed for the LNPP. The analysis covered power and shutdown operation, as well as risks from internal and external events. The external events evaluated include seismic, typhoon, internal fire and flooding.

The purpose of the integrated reliability analysis (IRA) program is to assure that the safety and reliability of the LNPP are maintained as designed during and after the procurement and construction phase. The IRA will demonstrate that the designed plant safety and availability performances are met through the design life of the plant. The plant safety performance includes the core damage frequency, the chance of an inadvertent reactor coolant system (RCS) depressurization, and the frequency of station blackout and reactor trip. The plant availability performance includes the plant production availability requirement, the frequency and duration of forced outages, the refueling duration capability, the duration of planned outages, and the frequency and duration of major outages.

The purpose of the emergency planning and its implementation procedures is to enable the plant personnel and/or the offsite authorities to handle any foreseeable emergency conditions in a safe and efficient manner. A more detailed description of the emergency plan is provided in Article 16 of this report.

The working scope of the backend program includes: (1) the nuclear power station decommissioning; (2) the transfer, interim storage and final disposal of spent fuels; and (3) the final disposal of low level radwaste from plant operation and decommissioning.

The purpose of the experience feedback is to collect and make good use of the experiences that have been gained in the stages of design, procurement, manufacturing, construction, commissioning, operation, and maintenance of domestic nuclear power plants.

14.1.2 Safety Assessment at Operation Stage

The safety assessment at the operation stage can mainly be divided into the following areas:

(1) FSAR Update

Licensees are required to update their final safety analysis reports periodically to incorporate the revised information and analyses that they submitted to the AEC. The description of the FSAR update is provided in Article 6 of this report. In response to the promulgation of the Act of Administrative Procedure, modification of relative regulations required by the Nuclear Reactor Facilities Regulation Act has been performed. The requirements related to the FSAR update are described in this section. In addition, the requirements for the periodic safety review are also provided in this section.

For an operating nuclear power plant, the first update of FSAR shall be completed within two years after the operating license is granted. The follow up FSAR updates shall be completed within six months after each fuel reload. If one FSAR is shared by multiple units, the reference date will be set by the second unit.

Periodic Safety Review

According to the Nuclear Reactor Facilities Regulation Act and its enforcement rules, the owner of a nuclear rector facility has to submit an “Integrated Safety Assessment Report” (ISAR) 6 months before the end of every 10 years of operation to the AEC for review. The major items in this report are as follows.

- Review of operating conditions, including operational safety, radiation safety and radioactive waste management,
- Review of items needed to be improved or reinforced, including review of reactor unit problems needed to be improved or reinforced and descriptions of commitments for improvement and reinforcement,
- Summary of the previous reviews and prospectiveness of items of paying attention to in the future and commitments of future improvements and schedules, and
- Other items requested by the AEC.

(2) License Renewal

According to Article 6 of the Nuclear Reactor Facilities Regulation Act, the valid period of the operating license shall be forty years at longest. When there is need to continue operation after the license is expired, an application for renewing the license thereof shall be filed by the licensee within the period prescribed by the competent authorities. Since the Chinshan nuclear power plant in Taiwan started its commercial operation in 1978, the TPC has performed necessary safety assessment and applied for license renewal in July 2009 for the Chinshan NPP. The important reference for safety evaluation is USNRC regulation 10 CFR 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants”. The major requirements include:

- Define the scope of safety assessment,
- Provide the following technical information:
  - integrated plant assessment (IPA)
  - time-limited aging analysis (TLAA)
  - updated final safety analysis report (UFSAR)
- Update technique specifications, and
- Provide environment information such as impacts of license renewal on surface water, underground water, land use, ecology, etc.

(3) PRA Update

The technology and application of PRA have been developed in this country for more than twenty years. The PRA models for the three operating nuclear power plants have been established through the cooperation among the AEC, the TPC and the Institute of Nuclear Energy Research (INER) since 1982. Up to the end of 1996, the so-called "living PRA" models for these plants were further completed by the TPC and INER. These models have provided very handy and quick tools for the operators to estimate the plant safety status. Several PRA application programs such as the establishment of "Taipower Integrated Risk Assessment and Management Model" and the establishment, maintenance and application of "Flood, Fire and Containment Safety Assessment Models" have also been performed by the PRA experts of both the TPC and INER. A project entitled "Standardization and Refinement of PRA Models on Operating Nuclear Power Plants" is currently performed by the TPC. In this project, the baseline PRA models (including models of power-operation, internal event, internal flood event) of the 3 operating nuclear power plants will be reviewed. A technically acceptable PRA complied with the American Society of Mechanical Engineers (ASME) PRA standard will be performed. Plant-specific data updated to the end of 2009 will also be updated into the PRA models. The modified PRA models will be applied to the future risk-informed analyses such as on-line maintenance application, outage maintenance scheduling, risk analyses of plant modifications, and so on.

(4) Safety Assessment for Power Uprate

Based on the magnitude of the power increase and the methods used to achieve the increase, the power uprate which is defined as the process of increasing the licensed power level at a commercial nuclear power plant, can be categorized into 3 categories. The measurement uncertainty recapture (MUR) power uprates result in power increases less than 2 percent and are achieved by implementing advanced techniques for feedwater flow measurement. The stretch power uprates (SPU) typically result in power level increases up to 7 percent and do not generally involve major plant modifications. The extended power uprates (EPU) result in power level increases greater than that of the SPU but less than 20 percent and usually require significant modifications to major plant equipment. For MUR and EPU, USNRC provides RIS 2002-03 and RS-001 as the review guides, respectively. While for SPU, the already approved cases and RS-001 are the major references for the USNRC review. In Taiwan, the TPC has submitted the

MUR application for the existing 3 nuclear power plants in operation during 2006 to 2008. The AEC reviewed these applications with reference to RIS 2002-03 and approved them during 2007 to 2009. The total power increase in 6 units is 55.64 MWe. It is expected that the TPC may submit the applications of SPU for Chinshan in December of 2010. The AEC has already prepared the related review guides for SPU with reference to RS-001 and the already approved foreign cases since 2009.

14.1.3 Design Changes

A nuclear power plant is required to be operated in accordance with the requirements described in its FSAR. Whenever design change or equipment overhaul is required, all works must be prepared in accordance with the Nuclear Reactor Facilities Regulation Act or relevant regulations. The AEC's approval is required before implementation of the following design changes:

- Change of the technical specifications,
- Resulting in more than a minimal increase in the frequency of occurrence or the consequence of an accident previously evaluated in the FSAR,
- Resulting in more than a minimal increase in either occurrence of a malfunction or the malfunction consequence of the structure, system, and component (SSC important to safety) previously evaluated in the FSAR,
- Creating a possibility for either an accident of a different type or a malfunction of an SSC important to safety with a different result than previously evaluated in the FSAR,
- Change of the design basis limit for a fission product barrier as described in the FSAR,
- Change of the evaluation method used in establishing the design bases and safety analyses as described in the FSAR, and
- Others as required by the regulatory body

The design change, which involves new safety issue as described above, is the item with any of the following conditions:

- The frequency of occurrence or the consequence of an accident previously evaluated in the FSAR is increased, or the frequency of occurrence of an equipment malfunction important to safety is higher than that previously estimated in the FSAR.
- The frequency of occurrence for an accident not previously evaluated in the FSAR is increased.
- The safety margin of the nuclear installation is decreased.

For design changes requiring the AEC's approval, an assessment report should be submitted to the AEC. Then, the AEC will review the report and the design change request (DCR) can't be performed until a satisfactory conclusion has been reached. Inspections will be conducted by the AEC during the work of design change. After the completion of the design changes, proof tests have to be performed to assure the quality of the changes.

14.1.4 Aging Management

For life extension of nuclear power plants, aging related issues receive more and more attention. Intensive world-wide efforts have been put into the development of life cycle management (LCM) for license renewal programs. Existing programs such as maintenance, surveillance, testing and inspection can be linked and integrated into the overall LCM strategies. The aging assessment for LCM together with the existing maintenance programs make timely improvement of the aging related problems and improve operational safety during the designed service life of the nuclear power plants.

14.1.4.1 Overview of the TPC's Nuclear Power Station Aging Management

The TPC has performed several aging related studies. The first step of these studies was to collect and review related documents. The components that are important to system function and subjected to aging were identified in this step. Since the aging assessment of an entire nuclear power plant is very complicated, therefore, a pilot study on certain selected systems was established. In a three year project, a BWR high pressure coolant injection (HPCI) system, a BWR residual heat removal (RHR) system, and a PWR condensate and feedwater system have been examined for demonstration purpose. The objective of this demonstration project is to develop general methodologies for aging assessment which can be used for other related systems in the future.

The aging of the reactor pressure vessel (RPV) and RPV internals are important issues. According to the surveillance program for the RPV material, test samples are removed from the RPV and assessed once every 10 years. The assessment of the toughness of the RPV material removed from the existing nuclear power plants in operation showed that sufficient safety margins were maintained. As for the RPV internal components, improvements on the material, water chemistry and stress distribution have been made. The visual inspection procedures for the RPV internal components have also been enhanced to minimize the probability of potential damages.

Before the year 2005, there were three aging management related projects performed by the TPC:

- Impact assessment of environmentally enhanced corrosion of the RPV.
- System development and application of technical assessment of aging management.
- Investigation of the stress corrosion cracking (SCC) of the steam generator secondary side for the Maanshan Nuclear Power Plant.

These projects have been replaced by Maintenance Rule (MR) associated with License Renewal program to take care of aging management of the three existing nuclear power plants in operation.

14.1.4.2 Future Perspective

An aging management system will be developed for each NPP regarding the planning, organizing, execution and control of the aging management process for the respective equipment and component. The scope of aging management will be identified and the assessment model will be established in this campaign for each plant.

14.2 Verification by Analysis, Surveillance, Testing and Inspection

Surveillance, testing, and periodic inspection requirements are included in the Technical Specifications of the FSAR of each NPP. Some examples are: surveillance requirements for in-service inspection and testing of the ASME Code Class 1, 2, and 3 components, airborne concentrations, the solid radwaste system, the shutdown margin, the moderator temperature coefficient, the reactor coolant temperature, the borating systems, the availability of shutdown flow paths, etc. Licensees are responsible to perform the required surveillance, testing and periodic self-inspection, and the AEC is responsible to audit the performance of these activities.

14.2.1 Nuclear Power Plant Inspection

The inspection of nuclear facilities is one of the most important tasks of the AEC. The following approaches are usually performed by the AEC to inspect an operating nuclear power plant:

- Resident inspectors perform daily monitoring and inspection on site of a nuclear power plant. The resident inspectors should be well informed and in good control of the plant operating conditions.
- Periodic or planned outage inspections are performed by means of group inspection to assure the quality of maintenance works.
- Unannounced inspections are performed without pre-notice to test the alertness of plant operators.
- Taskforce inspection for reviewing the complete plant operating condition is conducted at each site to assure nuclear safety. The scope of this inspection usually includes all primary items related to the safety of the nuclear power plant.

The inspections at construction stage are similar to that for an operating plant and those for the Lungmen NPP with two advanced boiling water reactors (ABWR) are described as follows:

(1) Resident Inspection

The responsibilities of the resident inspector during construction are: (i) to report the daily construction activities to the AEC headquarters, (ii) to monitor the implementation of the quality assurance program, (iii) to audit the conformance of construction activities, and (iv) to conduct a routine construction work inspection.

(2) Periodic Inspection

Periodic inspection is generally conducted every quarter. The frequency of the inspection will be increased depending on the nature of the construction work.

During the design stage, the design control function of the TPC, AE Company, and the Lungmen site office are inspected with reference of the guidance of the integrated design inspection program published by USNRC, and the quality assurance requirements.

During the manufacture stage, the inspection activities are focused on the manufacturing quality of the equipment and components important to safety. Examples are the reactor pressure vessel, reactor internal pump, fine motion control rod drive (FMCRD), and the liner of reinforced concrete containment vessel.

Three categories of inspections including the civil and structure, the mechanical and piping, and the instrumentation and electrical are conducted during the construction and installation stage. The civil and structure inspection activities are focused on steel structure welding, rebar processing, concrete quality control, concrete pouring control, administrative control, and quality control and quality assurance functions. The mechanical and piping inspection are focused on welding, non-destructive examination, and quality control and quality assurance functions for the installation of the mechanical and piping equipment. The electrical and instrumentation inspection items are focused on cable tray welding, conduit installation, instrumentation tubing installation, cable routing and identification, fire protection, seismic resistance, and quality control and quality assurance functions.

(3) Special Taskforce Inspection

For items that require special professional technology to do the inspection job, experts outside the AEC are invited to join the inspection team. The areas considered for special taskforce inspections include:

(a) Civil and Structure

The inspection activities are focused on structural design, concrete quality control, materials composition of concrete, and concrete pouring control.

(b) Special Process Control

The inspection works are focused on quality control of welding and non-destructive examination, material characteristics examination, defect disposition, re-evaluation of radiographic examination film, and witness of special process implementation.

(c) Human Factor Engineering

The inspection items are focused on the human-system interface design for the main control room (MCR), alternate shutdown panel, and local panel. The inspection will also check the instruction manual and procedures to confirm that the NUREG-0711 requirements are followed in the human factor engineering design.

(d) Fire Protection

The inspection activities are focused on the conformance of the design and installation of the fire prevention system and fire protection system to regulations, including fire hazard analysis, fire detection, fire confinement, fire resistant material applicability, and fire protection program.

(4) Observation of the Key Holding Points of Construction

For items that may impose a significant effect on the construction quality, a prior approval is required for the activities to proceed. The submittals for the approval shall include the quality control plan, procedures or test plan, and test results. The items for key holding point inspection are listed as follows.

(a) Reactor building base mat first concrete pouring,

(b) Reactor pressure vessel installation,

(c) Safety-related mechanical equipment installation initiation,

(d) Safety-related piping system installation initiation,

(e) Safety-related instrumentation and control equipment installation initiation,

(f) Safety-related electrical equipment installation initiation,

(g) Containment integrity functional test,

(h) Reactor protection system functional test initiation,

(i) Cold hydrostatic test,

(j) Simulator operator training initiation,.

(k) Pre-operational test initiation,

(l) System integration functional test,

(m) Initial fuel loading,

(n) Initial criticality and safety margin test,

(o) Turbine rolling and initial synchronization, and

(p) Power ascension tests including tests at 25%, 50%, 75%, and 100% rated power.

Following the completion of construction activities, continued inspection activities are taken by the AEC to monitor the conformance to the quality requirements.

Pre-operational test and startup test will be performed after completion of the construction activities. The pre-operational tests consist of post-construction test, hydrostatic test, system flushing, initial test run of rotary mechanical equipment, and system operational test. The inspection activities are focused on quality assurance of test programs, test procedures implementation, test result review, witness point implementation, non-conformance disposition, and system operability prior to initial fuel loading.

The startup tests encompass the initial fuel loading, initial criticality, turbine rolling, generator synchronization, and the 25%, 50%, 75%, and 100% rated power ascension tests. The inspection activities are focused on the quality assurance of test programs, test procedures implementation, test result review, witness point implementation, non-conformance disposition, and 100-hour continuous 100% rated power operation test.

Before entering into commercial operation, a review of operational readiness regarding the operation and maintenance (O&M) administrative management is conducted to ensure safe and reliable operation of the plant. The inspection items are focused on the training of the O&M personnel, operational safety review and audit function, in-service inspection and in-service testing program establishment, quality control and quality assurance program establishment and implementation.

(5) Resident Safety Team of the Department of Nuclear Safety of the TPC

The resident safety team of the Department of Nuclear Safety at an NPP site was established at the beginning of commercial operation of each NPP. Currently, there are one manager, three section heads and three engineers in each team. The three section heads are responsible for safety, regulation, and quality assurance respectively. The major tasks of this team include:

(a) General auditing and special evaluation associated with quality assurance,

(b) Review and verification of safety related affairs such as abnormal events, operational procedures, and QA specification for procurement,

(c) Collection of daily operational information for reporting to the superiors and related organizations, and

(d) Participation of the review activities performed by the headquarters of the TPC during outage maintenance. These activities include on-site verification of in-service test, review of shutdown and restart safety, design change report, maintenance, testing, radiation protection, working safety, and package of outage maintenance document.

Corrective action report or recommendations will be proposed by the resident safety team, if technical or equipment deficiency were found through general auditing and verification of daily operational conditions of nuclear units. “Special Evaluation Program” will be performed to search for potential adverse contributors, if important quality problems are identified in system, equipment, or control practices. For the adverse contributors founded in the evaluation program, practical corrective recommendations or corrective action report will be proposed for the related NPP’s reference.

14.2.2 Reload Safety Analysis

For each fuel reload, licensees are required to submit a reload safety analysis report (RSAR) for the BWR or reload safety evaluation report (RSER) for the PWR to the

AEC. This RSAR or RSER needs to be reviewed and approved by the AEC before the restart of the nuclear power unit for the next fuel cycle. Extensive review in the areas of fuel mechanical design, thermal hydraulic design, neutronic design, transient analysis, and other affected design or analysis will usually be undertaken in any of the following situations:

- Different fuel vendor from the original one is selected,
- New fuel type of the same vendor is introduced, and
- Revision or major modification of the reload safety analysis methodology is proposed.

As a result of the review, some additional inspection or test may be required. For example, the observation of crud thickness and the measurement of oxide layer thickness and internal gas pressure of the fuel rod were required when a new fuel type (e.g., ATRIUM-10) was proposed in both Chinshan and Kuosheng Nuclear Power Stations.

14.2.3 Preventive Maintenance

In order to keep the equipment and systems in good conditions and to ensure that the intended design functions of the equipment are maintained, periodic and planned maintenance should be performed. The maintenance activities such as inspections, measurements and adjustments shall meet the requirements of the quality assurance program. The preventive maintenance in a nuclear power station is classified into two categories: the daily preventive maintenance and the planned preventive maintenance during outage. The contents of these two categories are described as follows.

(1) Daily Preventive Maintenance

Two computerized maintenance management program called “Maintenance Management Computerization System (MMCS)” and “Maintenance Integrated Risk Utilities (MIRU)” have been developed. All the daily preventive maintenance activities such as work assignment, schedule, notice, performance and validation as well as information storage and tracking of delayed items are all handled by using MMCS and MIRU.

(2) Planned Preventive Maintenance during Outage

Items that are on the list of ten-year long-term maintenance program are reviewed before each outage. Among them, those items that the preventive maintenance was planned to be performed in a specific outage and items that required preventive maintenance as selected from the monitoring results will be put into the outage maintenance schedule. Preventive maintenance is then performed in accordance with the outage working procedures, equipment maintenance working procedures and other related procedures.

14.2.4 Other Safety Analysis at Operation Stage

In case the system parameters departed far away from normal values or there was a malfunction of the structures, systems, and components (SSCs), the licensee is required to justify for continued operation and to report it to the AEC. Depending on the degree of severity of the situation, a safety analysis may be required and an extensive review may be initiated. For example, cracks have been found in the welds of the cover plates of core shroud support access hole at Chinshan Unit 1 in 1990. The observed situation was similar to that in the cracks found in Peach Bottom Unit 3 as described in the NRC Information Notice 88-03. The cracks could potentially result in complete weld failure and increase of the core bypass flow. As part of the supporting material to justify continued operation, a safety analysis has been performed to show that there was no safety concern for the increased core bypass flow, because the postulated event was less severe than that of a recirculation pump seizure, which was covered in the original FSAR of the Chinshan NPP.

As for the spent fuel storage, all the spent fuels from the three operating NPPs are currently stored in the spent fuel pools located in the plant sites. Extensive review in the areas of fuel mechanical design, thermal hydraulic design, neutronic design and event analysis have been undertaken for the spent fuel re-racking and the subsequent second re-racking projects. The capacity of the spent fuel pools of the 1^{st} and 2^{nd} nuclear power plant in Taiwan will be used up in 2013 and 2016 respectively. The INER has won a contract from the TPC to build an independent spent fuel storage installation (ISFSI) with the dry storage cask at the 1^{st} Nuclear Power Station（i.e. Chinshan）on August 2005. The objective of dry storage is to safely store the spent fuel for an extended (20 year) period at a site. In the United States, the design of dry storage installation has to follow the requirements of 10 CFR Part 72. In this country, the TPC has to submit safety analysis report in accordance to the regulations set by the AEC for the establishment of spent fuel dry storage installations. These regulations include “On Site Spent Fuel Dry Storage Installation Regulation”, “Guidelines for On Site Spent Fuel Dry Storage Installation Safety Analysis Report”, and so on.

ARTICLE 15. RADIATION PROTECTION

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

15.1 Regulatory Framework and Protection of Radiation Workers

15.1.1 Law, Enforcement Rules, and Regulations for Radiation Protection

The Ionizing Radiation Protection Act (IRPA) was enacted in 2002 and came into effect on February 1, 2003. This Act consists of 5 chapters and 57 Articles. At the same time, 22 daughter regulations took effect for the implementation of the IRPA.

The purpose of the IRPA is to properly manage radioactive material, equipment capable of producing ionizing radiation, and radiation practices, so as to prevent the radiation workers and the public from the detriment of radiation.

The IRPA prescribes the basic radiation protection principles and the following nuclear power plant related topics are emphasized in it:

- Provisions for protective measures against radiation hazards that keep the radioactive material release and the occupational radiation exposure as low as reasonably achievable (ALARA),
- Provisions for safety measures related to operations stipulating the necessary actions for protecting human bodies, the public, and the environment from radiation hazards which may accompany the operations of nuclear power stations,
- Performance criteria for the personnel dosimetry service for radiation workers or persons having access to nuclear installations, and
- Training requirements for the persons working in radiation environment.

The Enforcement Rules of the IRPA consists of 25 Articles, to address the details and necessary supplement of the Act. The provisions related to nuclear power stations are the content of the radiation protection plan, the requirement for the monitoring of the radiation worker’s dose, the content of the safety assessment report for discharge of the gaseous and liquid effluents, the requirement to conduct the radiation training for the radiation workers and the content of the evaluation report for a possible accident.

There are 22 Regulations prescribe the technical requirements on radiation protection, and the following topics are emphasized for the nuclear power stations:

- Detailed provisions of the safety standards to protect the radiation worker and the public against the radiation,
- Detailed provisions for the safe transport of radioactive materials,

- Detailed provisions for the establishment of radiation protection organization in the nuclear power station,
- Detailed provisions for the radioactive materials and associated practices, such as designation of a controlled area,
- Detailed provisions for the monitoring of the radiation work places and the environment,
- Detailed provisions for the radiation protection personnel, and
- Detailed provisions of the performance criteria for the personnel dosimetry service.

15.1.2 ALARA for Occupational Exposure

15.1.2.1 Implementation of ALARA in the Design and Construction of Nuclear Power Plants

The TPC incorporates the following radiation protection principles in the design and construction of nuclear power stations, for assuring the criterion of ALARA and maintaining the radiation doses to workers and the general public within the applicable limits:

- Installation of equipment capable of producing ionizing radiation separately in shielded rooms with partition,
- Installation of shields to fully attenuate radiation from pipes and equipment containing large amounts of radioactivity,
- Use of remotely controlled equipment and automatic equipment in radiation controlled area,
- Installation of ventilation facility in area~~s~~ of potential air contamination,
- Installation of a continuously radiation monitoring system in nuclear power station, and
- Establishment of the appropriate radiation zone classification and access control.

15.1.2.2 Criteria for Radiation Exposure Control

The TPC in practice establishes a target dose limit for radiation workers at 90% of the official limits, as shown in Table 15.1, and controls radiation doses within the target dose limit. It is prescribed in the procedures that any person whose annual dose reaches the target value will be closely monitored on daily basis and any person whose annual dose reaches 80% of the official limit is not allowed to work in the high radiation area, unless approved by the plant general manager and proper measures are taken.

15.1.2.3 Management of Radiation Work

It is prescribed in the TPC's procedures that any person, intended to have access to the controlled areas for radiation works, shall obtain a radiation work permit in advance. This is prepared separately for the consideration of the type of the work, the radiation level, and the working conditions. For the issuance of this permit, the radiation protection personnel from the Health Physics Section have to evaluate the expected dose and, if necessary, to further impose special conditions on the worker.

15.1.2.4 Reduction of Occupational Radiation Exposure

The TPC has established and implemented respective targets for reducing the occupational radiation exposure, such as the annual collective dose, collective dose during planned refueling period or preventive maintenance period, and the job-specific collective dose. The TPC prescribes in the procedures that any radiation work shall follow the plan established beforehand. It is also prescribed that the ALARA Committee meeting shall be held at the planning stage to estimate and evaluate the radiation level and the expected collective dose. Furthermore, the TPC will evaluate the ALARA performance more than once a year for the major maintenance work, design modification, and replacement of equipment. When conducting radiation work, the technique of dose reduction shall be described in the radiation work procedure or the radiation work permit. It is required for the radiation workers to utilize the proven ALARA technique from the past experience of similar work.

The total occupational collective dose of the TPC employees in 2009 was 7.45 man-sievert. Among them, the major contributions were from the employees of the three NPPs in operation, namely 2.08 man-sievert from the Chinshan NPP, 3.05 man-sievert from the Kuosheng NPP, and 2.01 man-sievert from the Maanshan NPP, respectively. The collective dose distribution for different radiation work categories and the collective doses for the employees in three NPPs from 2004 to 2009 are shown in Tables 15.2 and 15.3, respectively. The occupational collective dose of the Chinshan NPP was 1.03 man-sievert in 2004, which was in the WANO's top upper quartile rank. It can be seen from these two tables that the collective dose of the employees in each NPP is decreasing in trend, except when refueling occurred in the same year for both units of the NPP.

15.1.2.5 Personnel Dosimetry Service and Its Verification

Every year there are approximately 30,000 workers associated with the occupational radiation exposure in Taiwan. The Atomic Energy Council has authorized the INER to establish the National Database Center of Occupational Radiation Exposures (NDCORE) to manage the operation.

All organizations with personnel dosimetry service, including the TPC, must obtain approval from the AEC before they conduct the service. The TPC distributes, collects and reads monthly the thermoluminescent dosimeters (TLD) carried by its employees and informs relevant personnel of the results. These results are also reported to the AEC on a monthly basis. Accuracy of the reading is maintained by the accreditation from the Taiwan Accreditation Foundation (TAF) Program of the Bureau of Standards, Metrology and Inspection and by inter-laboratory comparison.

15.1.2.6 Radiation Protection Training

The TPC prescribes in the procedures that radiation workers and any personnel having access to the nuclear power stations and radioactive waste treatment or storage facilities shall take appropriate radiation protection training courses. Workers acquire basic knowledge and handling skills needed for radiation work through this training. The curriculum is classified into the following courses:

- Course for personnel of temporary access,
- Course for personnel of occasional access,
- Course for radiation workers,
- Refreshing course, and
- Course for managers.

The specific training duration is assigned for each course. The basic subjects include fundamentals of radiation protection, health effects of radiation, access procedures to the controlled area, and emergency preparedness. Additional subjects include radiation exposure control, contamination control, waste management, and the use of instruments and protective equipment. Those who have taken the training courses shall be evaluated by written examination. After passing the evaluation, the trainee is then qualified to have access to or conduct works in the controlled areas.

15.1.3 Activities to Enhance the Regulatory Control

The AEC had conducted a series of projects since July 1996 to incorporate the ICRP-60 recommendations into the relevant Acts and regulations. The Ionizing Radiation Protection Act was enacted in January 2002, in which some of the radiation protection concept of ICRP-60 was incorporated. Major contents of the Regulation, "Safety Standards for Protection against Ionizing Radiation," promulgated in January 2003 and revised in December 2005 are the reduction of the dose limits and the introduction of an internal exposure assessment system following the abolishment of the maximum permissible dose concept. The AEC has established the Radiation Protection Control System for the efficient control of the personnel, radiation sources, equipment capable of producing ionizing radiation, and the radiation practice.

15.1.3.1 Safety Standards for Protection against Ionizing Radiation

The safety standards for protection against the ionizing radiation had been updated to follow ICRP-60 on December 30, 2005. The "effective dose" was adopted to replace the "effective dose equivalent" used in the previous version. The personal occupational dose for radiation workers shall not exceed 100 milli-sievert in five consecutive years defined by the AEC, in which the first five year cycle is year 2003 through 2007 and the current cycle is 2008 through 2012. The radiation weighting factor and the dose conversion coefficients recommended by ICRP-60 were incorporated into this version of the Standards. The effective dose for the general public shall not exceed one milli-sievert in one year. Six groups of inhalation and ingestion dose coefficient are adopted for internal dose evaluation with respect to different age groups of the public.

15.1.3.2 Utilization of Radiation Protection Control System

In order to implement the new Ionizing Radiation Protection Act and to realize the policy of e-administration so as to effectively control the utilization of radioactive materials and equipment capable of producing ionizing radiation, the AEC started the use of the Radiation Protection Control System (RPCS) on February 1, 2003 for the better protection of the public from radiation hazard. The RPCS puts the management of personnel qualifications, business operators' capability, import/export of radioactive materials and equipment, etc. into a computer-controlled management system.

15.2 Protection of Radiation Exposure for Members of the Public

15.2.1 Dose Constraints on Radioactive Effluents

The AEC refers the Appendix I to US 10 CFR Part 50, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents" to establish the maximum allowable concentration of gaseous and liquid effluents to be released into the environment from NPPs and the relevant dose constraints. According to the regulations, each NPP is allowed to discharge the gaseous or liquid effluent into the environment after confirming its concentration is within the allowable limit.

The dose constraints to members of the public for gaseous effluents, as prescribed in the regulations, are as follows:

- Air absorbed dose from gamma rays: 0.1 mGy/yr-unit,
- Air absorbed dose from beta rays: 0.2 mGy/yr-unit,
- Effective dose from external exposure: 0.05 mSv/yr-unit,
- Equivalent dose in skin from external exposure: 0.15 mSv/yr-unit, and
- Equivalent dose in organ from radioactive iodine and particulates:

  0.15 mSv/yr-unit

The dose constraints to members of the public for liquid effluents are as follows:

- Effective dose: 0.03 mSv/yr-unit, and
- Equivalent dose in organ: 0.1 mSv/yr-unit.

15.2.2 Assessment of Radiation Doses to the Population around NPPs

The radiation dose to and its effect on the population around NPPs are assessed quarterly according to the Regulation entitled "Criteria for Management of Radiation Workplaces and Environmental Radiation Monitoring outside Them". The assessments model is based on the radioactivity of liquid and gaseous effluents, the atmospheric conditions, dose conversion factors, and social data including agricultural and marine products of the local community within a radius of 50 km.

The 3 operating NPPs of the TPC evaluate the radiation dose to the population around the NPP every season to confirm the dose to the critical people near the NPP is within the regulatory dose limit. The actual radionuclides released through gaseous and liquid pathways are recorded as the source term for the evaluation. The hourly meteorological data of the site area include the wind direction and speed as well as the wind stability class are recorded based on the on site monitoring station. For the gaseous pathway, the relative atmospheric dispersion factor (X/Q) and deposition factor (D/Q) are calculated first at the interested position points around and within 50 kilometers of the NPP. Then the dose rate to the critical group of individual and the population dose for the population within 50 kilometers are evaluated and compared with the regulatory control requirement. The exposure pathways considered in the gaseous release situation include plume exposure, ground exposure, inhalation and ingestion of contaminated vegetable, beef and milk. The liquid release is evaluated in a similar way with the actual release recorded for all radionuclide. The pathways considered are ingestion of fish and the invertebrate and swimming and shoreline recreation activities.

Typical computer programs such as XOQDOQ-82, GASPAR and LADTAP-II are used for the evaluation. The dose conversion coefficients are replaced with the ICRP-60 recommended values to reflect the current requirement in the Regulation, “Safety Standards for Protection against Ionizing Radiation”. The effective dose and organ equivalent dose from the gaseous and liquid effluents for the 3 operating NPPs are summarized for the years from 2003 to 2009 and shown in Tables 15.4 to 15.6. As the actual quantities of released radionuclides are less than the designed values in FSAR, the dose to the general public surrounding the NPP is at the background level and thus meets the regulatory requirement.

15.2.3 Environmental Radiation Monitoring by the Licensee

The TPC conducts environmental radiation monitoring activities including the installation and operation of the TLD posts and environmental radiation monitors as well as the analysis of the radioactivity of environmental samples, in accordance with the Regulation entitled ”Criteria for Management of Radiation Workplaces and Environmental Radiation Monitoring outside Them“.

There are a total of 20 environmental radiation monitors installed within the 2 km radius of the Chinshan, Kuosheng, Maanshan and Lungmen nuclear power plants, with 5 monitors for each NPP. All the monitors are installed in consideration of topography, population distribution, and atmospheric dispersion factors. They monitor the gamma dose rate continuously at 1 m above the ground. The status of the monitoring system and the radiation dose levels can be confirmed, on real time basis, in the Radiation Laboratory of the TPC and the Health Physics Station of the nuclear power unit, where the monitors are connected on-line. TLD are installed on posts for assessing the cumulative quarterly gamma radiation dose of the area within a radius of 50 km around the nuclear power station. The numbers of TLD installed are 45 for Chinshan, 36 for Kuosheng, 32 for Maanshan and 40 for Lungmen nuclear power plant.

The environmental samples are air samples, waterborne samples (seawater, drinking water, ground water, underground water, precipitation), seabed samples (sediment, shore line sand), and food products (milk, vegetables, fruits, sweet potato, fishes, shellfish, seaweed). Different types of samples are measured at different periods as shown in Table 15.7.

15.2.4 Environmental Radiation Monitoring Network by the AEC

The Radiation Monitoring Center (RMC) of the AEC measures the radioactivity in airborne dust, fallout, rainwater, drinking water, underground water, livestock products, farm products, soil, and milk, and the background radiation levels throughout the nation. The RMC also installs and operates a nation-wide Environmental Radiation Monitoring Network (ERMN). This enables the RMC to quickly detect and properly respond to any abnormal situations in environmental radioactivity. The nationwide ERMN, as shown in Figure 15.1, consists of the following facilities: an environmental radiation monitoring center in the RMC, local monitoring stations at five major cities with large population, one monitoring post at the AEC Headquarters, and the monitoring posts at three nuclear power station sites, INER and around the nation. Currently a total of 30 radiation monitoring stations has been established in Taiwan, Kinmen island and Lan-yu island. The RMC has conducted annually national and international inter-laboratory comparisons on environmental radioactivity measurements for quality control.

Table 15.1 Currently Applicable Dose Limits

Category	Radiation Worker	General Public (Critical Group)
Effective Dose	50 mSv (any single year) 100 mSv (5 continuous years)	1 mSv per year
Annual Equivalent Dose in -The Lens of the Eye -The Skin, Hands, and Feet	 150 mSv 500 mSv	 15 mSv 50 mSv

Table 15.2 Collective Dose in Different Radiation Work Category

Unit: Man-Sievert						
	2004	2005	2006	2007	2008	2009
TPC	8.250	7.970	6.834	7.811	6.061	7.447
Medical	0.460	0.450	0.518	0.534	0.537	0.429
Industrial	0.820	1.420	1.798	1.232	0.922	0.768
Others	0.270	0.360	0.298	0.254	0.179	0.203
NRM	0.000	0.000	0.000	0.000	0.000	0.000
Total	9.790	10. 220	9.448	9.830	7.699	8.847
Unit: Man-Year						
Total Employee	35,774	39,243	40,882	43,447	44,120	42,966

Note: NRM- Natural Radioactive Material

Table 15.3 Collective Dose in Each NPP of the TPC

Unit: Man-Sievert

	2004	2005	2006	2007	2008	2009
Chinshan	2.067	2.931	1.936	1.970	2.364	2.076
Kuosheng	4.623	3.049	2.688	4.417	2.637	3.048
Maanshan	1.544	1.928	2.182	1.417	0.963	2.011
Others	0.016	0.062	0.028	0.007	0.097	0.313
Total	8.250	7.970	6.834	7.811	6.061	7.447

Table 15.4 General Public Dose Evaluation around the Chinshan NPP

Unit: mSv/yr*

Year	Gaseous Effluent		Liquid Effluent	
	Effective Dose[1]	Equivalent Dose[2]	Effective Dose[3]	Equivalent Dose[4]
2003	2.60×10^{-8}	1.11×10^{-3}	3.56×10^{-5}	1.38×10^{-4}
2004	3.86×10^{-7}	2.41×10^{-4}	1.87×10^{-5}	3.55×10^{-5}
2005	1.40×10^{-7}	2.28×10^{-4}	1.19×10^{-4}	2.35×10^{-4}
2006	1.20×10^{-6}	9.81×10^{-5}	4.02×10^{-5}	7.94×10^{-5}
2007	1.34×10^{-6}	1.23×10^{-4}	5.55×10^{-5}	1.05×10^{-4}
2008	1.37×10^{-6}	2.77×10^{-4}	5.87×10^{-4}	3.00×10^{-3}
2009	3.40×10^{-6}	2.68×10^{-4}	2.05×10^{-4}	4.43×10^{-4}

Note:1. From external exposure of noble gas (dose constraint 0.05mSv/yr-unit).

2. Equivalent dose of organ from radioactive iodine and particulate (dose constraint 0.15mSv/ yr-unit).

3. Dose constraint 0.03mSv/yr-unit.

4. Equivalent dose of organ (dose constraint 0.1 mSv/yr-unit).

* for two units.

Table 15.5 General Public Dose Evaluation around the Kuosheng NPP

Unit: mSv/yr*

Year	Gaseous Effluent		Liquid Effluent	
	Effective Dose[1]	Equivalent Dose[2]	Effective Dose[3]	Equivalent Dose[4]
2003	8.75×10^{-4}	1.47×10^{-3}	8.42×10^{-5}	3.80×10^{-4}
2004	1.30×10^{-3}	2.62×10^{-3}	5.92×10^{-5}	1.20×10^{-4}
2005	1.03×10^{-3}	7.84×10^{-3}	7.60×10^{-5}	1.51×10^{-4}
2006	1.21×10^{-3}	3.39×10^{-3}	5.52×10^{-5}	1.09×10^{-4}
2007	7.00×10^{-4}	1.48×10^{-2}	8.16×10^{-5}	1.82×10^{-4}
2008	7.62×10^{-4}	1.83×10^{-2}	1.87×10^{-4}	8.30×10^{-4}
2009	8.30×10^{-4}	3.20×10^{-3}	1.10×10^{-4}	2.94×10^{-4}

Note:1. From external exposure of noble gas (dose constraint 0.05mSv/yr-unit).

2. Equivalent dose of organ from radioactive iodine and particulate (dose constraint 0.15mSv/ yr-unit).

3. Dose constraint 0.03mSv/yr-unit.

4. Equivalent dose of organ (dose constraint 0.1mSv/yr-unit).

* for two units.

Table 15.6 General Public Dose Evaluation around the Maanshan NPP

Unit: mSv/yr

Year	Gaseous Effluent*		Liquid Effluent*	
	Effective Dose[1]	Equivalent Dose[2]	Effective Dose[3]	Equivalent Dose[4]
2003	4.84×10^{-4}	2.08×10^{-3}	7.00×10^{-5}	7.84×10^{-5}
2004	3.78×10^{-4}	1.71×10^{-3}	2.53×10^{-5}	3.30×10^{-5}
2005	2.75×10^{-4}	1.19×10^{-3}	2.56×10^{-5}	3.08×10^{-5}
2006	1.60×10^{-4}	1.29×10^{-3}	2.46×10^{-5}	2.69×10^{-5}
2007	1.25×10^{-4}	4.99×10^{-4}	2.11×10^{-5}	2.13×10^{-5}
2008	4.47×10^{-4}	1.20×10^{-3}	3.66×10^{-5}	1.58×10^{-4}
2009	2.63×10^{-4}	2.28×10^{-3}	2.30×10^{-5}	2.30×10^{-5}

Note:1. From external exposure of noble gas (dose constraint 0.05mSv/yr-unit).

2. Equivalent dose of organ from radioactive iodine and particulate (dose constraint 0.15mSv/ yr-unit).

3. Dose constraint 0.03mSv/yr-unit.

4. Equivalent dose of organ (dose constraint 0.1mSv/yr-unit).

* for two units.

Table 15.7 Environmental Radiation Monitoring in the Vicinity of NPPs

Unit: Number of Samples

Sample Items	NPP			Analysis Items/Analysis Frequency
	Chinshan	Kuosheng	Maanshan	
Direct Radiation				
TLD (Continuous) Stations	45	36	32	Gamma Dose Rate/ Quarterly
HPIC (Continuous) Stations	5	5	5	Gamma Dose Rate /hr
Air				
Particulates (Continuous) Stations	16	11	16	Gross β, γ Spectrum [1]/ Weekly, γ Spectrum/Quarterly, Sr-89,90[2]
Iodine (Continuous) Stations	16	11	16	I-131/Weekly
Fallout (Continuous)	1	1	1	γ Spectrum / Monthly
Water				
Sea Water (Quarterly)	9	9	10	γ Spectrum [3], H-3[3]/ Monthly, Sr-89,90[2]
Drinking Water(Quarterly)	7	6	7	γ Spectrum, H-3/ Quarterly, Sr-89,90[2], I-131[4]
River Water (Quarterly)	2	4	2	γ Spectrum, H-3/ Quarterly, Sr-89,90[2]
Pond Water (Quarterly)	5	3	3	γ Spectrum, H-3/ Quarterly, Sr-89,90[2]
Ground Water (Quarterly)	2	3	2	γ Spectrum, H-3/ Quarterly, Sr-89,90[2]
Precipitation I (Monthly)	2	2	3	γ Spectrum / Monthly, H-3/ Quarterly, Sr-89,90[2]
Precipitation II (Rain)	2	2	3	γ Spectrum, H-3[7]
Agriculture & Marine Products				
Milk: Cow/Goat (Quarterly)	—	—	1	I-131, γ Spectrum / Quarterly, Sr-89,90[2]
Rice (Semiannually)	2	3	3	γ Spectrum / Semiannually, Sr-89,90[2]
Vegetables (Semiannually)	6	5	5	I-131, γ Spectrum / Semiannually, Sr-89,90[2]
Tea (Semiannually)	5	—	—	γ Spectrum / Semiannually, Sr-89,90[2]

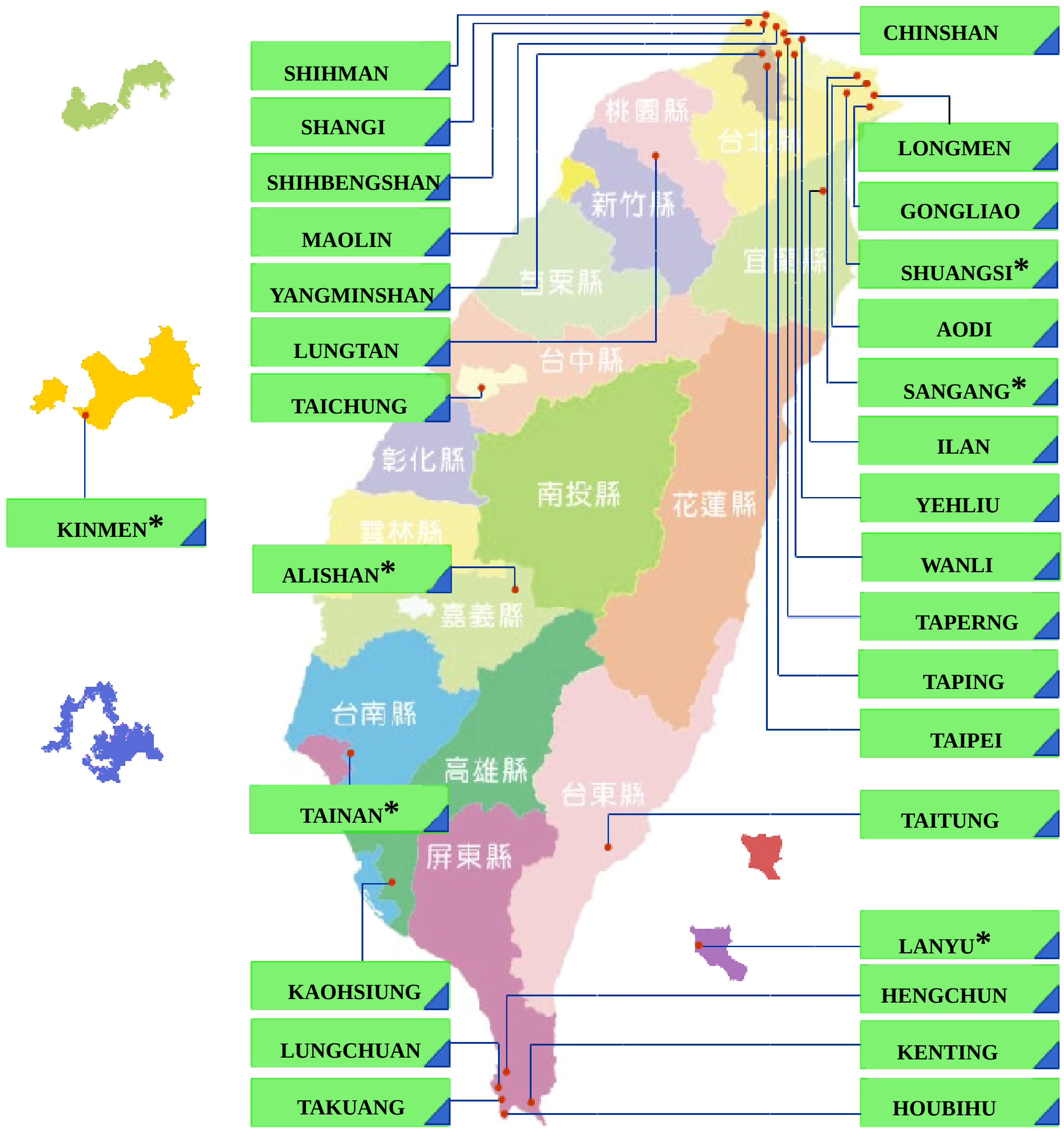
Table 15.7 Environmental Radiation Monitoring in the Vicinity of NPPs (continued)

Unit: Number of Samples

Sample Items	NPP			Analysis Items/Analysis Frequency
	Chinshan	Kuosheng	Maanshan	
Fruits(Annually)	2	2	1	γ Spectrum / Annually, Sr-89,90[2]
Vegetables (Root) (Annually)	3	3	2	γ Spectrum / Annually, Sr-89,90[2]
Sweet Potato (Annually)	1	1	—	γ Spectrum / Annually, Sr-89,90[2]
Vegetable Stem[5] (Annually)	1	1	1	γ Spectrum / Annually, Sr-89,90[2]
Poultry (Semiannually)	3	3	3	γ Spectrum / Semiannually, Sr-89,90[2]
Seaweed (Annually)	2	2	2	I-131, γ Spectrum / Annually, Sr-89,90[2]
Sea Fish & Shellfish (Quarterly)	5	5	6	γ Spectrum / Quarterly, Sr-89,90[2]
Index Organism				
Acacia (Land)(Monthly)	1	1	1	γ Spectrum / Monthly
Algae (Sea) (Annually)	1	1	1	I-131, γ Spectrum / Annually, Sr-89,90[2]
Land & Coast				
Beach Sand (Quarterly[6])	9	12	11	γ Spectrum / Quarterly
Soil (Semiannually)	14	14	11	γ Spectrum / Semiannually
Sea Sediment (Semiannually)	4	4	4	γ Spectrum / Semiannually

Note: 1. Conduct γ Spectrum analysis if weekly Gross β > 4 Bq/m^3.

2. Conduct Sr-89,90 analysis if Cs-137 exceeds limit set by AEC.
3. Conduct γ Spectrum and tritium analysis in all stations if monthly results from inlet, outlet and counterpart stations exceed the limit set by AEC.
4. Conduct I-131 analysis if I-131 is found in air.
5. First NPS: bamboo shoot; Third NPS: onion.
6. Within 600 meters from reactor cooling water inlet or outlet : Chinshan - outlet; Kuosheng - outlet; Maanshan - inlet and outlet.
7. Conduct tritium analysis during rainy period.



Note：

(1) There are 30 radiation monitoring stations in total (up to the end of 2009).

(2) Stations with "*" were established after 2004.

(3) Dose rate (in μSv/h) will be updated every one hour.

Figure 15.1 Environmental Radiation Monitoring Network in the ROC

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 on their territory, insofar as they are likely to be affected by 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.**

16.1 On-Site and Off-Site Emergency Preparedness

16.1.1 Laws, Regulations and Requirements

On the basis of the Nuclear Reactor Facilities Regulation Act, the safety of a nuclear installation is strictly regulated in every stage of design, construction and operation. Nevertheless, to assure the preparedness against the very unlikely occurrence of accidents with large release of radioactive materials, the “Nuclear Accident Emergency Response Plan (NAERP)” was promulgated in 1981 by the Executive Yuan, the highest administrative authority in this nation, and later revised in 1994, 1998 and 2002, respectively. The Nuclear Emergency Response Act was then promulgated by the President on December 24, 2003. In order to carry out the emergency response activities effectively for a nuclear accident, the response mechanisms have been established. The central government is responsible for the communications and decision-making for the public protection in case of a nuclear accident. The local government is responsible for the implementation of the related protective actions with the support of experts from the AEC and military units. The response organizations and their functional responsibilities are described in following section. The nuclear reactor facility licensee shall set up a dedicated Nuclear Emergency Response Unit (NERU) and the Nuclear Emergency Response Organization (NERO) within the Facility. The responsibilities and the activation timing of the dedicated NERU and the NERO, and the relevant operational procedures are to be proposed by the nuclear reactor facility licensee and submitted to the AEC for approval.

The Nuclear Emergency Response Act, the Enforcement Rules for the Implementation of this Act, and the Emergency Response Basic Plan (ERBP) (which was enacted in July 2005 by the AEC to replace the previous NAERP) cover the responsibilities of the competent organizations, accident categorizations, protective actions and recovery measures for nuclear emergency response. The major contents of the Emergency Response Basic Plan are summarized as follows.

16.1.1.1 Emergency Response Organizations and Their Missions

In case of a nuclear accident, the TPC is responsible for all the emergency response activities inside the plant, while the National Nuclear Emergency Response Center (NNERC) assumes those activities outside the plant. This center consists of delegates from the following organizations: AEC, Ministry of the Interior, Ministry of National Defense, Ministry of Economic Affairs, Ministry of Transportation and Communications, Council of Agriculture, Department of Health, National Communications Commission, Environmental Protection Administration, Coast Guard Administration, and Government Information Office. Several temporary centers will be organized and under the command of the NNERC if a nuclear accident occurs. The organization of NNERC is shown in Figure 16.1. These centers and their missions are described as follows.

(1) National Nuclear Emergency Response Center

The major missions of the National Nuclear Emergency Response Center (NNERC) are supervision of the response measures, evaluation of the accident, notification of the activation of Regional Nuclear Emergency Response Center (RNERC) and Nuclear Emergency Support Center (NESC), announcement of the protective actions for the public, releasing information associated with accident conditions and rescuing movements, clarification of false messages and notification of government announcements.

(2) Radiation Monitoring and Dose Assessment Center

The Radiation Monitoring and Dose Assessment Center (RMDAC) is composed of specialists from the AEC, the Central Weather Bureau and the TPC. Its major missions are information collection, siren broadcasting, accident consequence prediction, radiation detection and monitoring, public dose assessment, protection action suggestions, providing information and technical support to other centers, etc.

(3) Regional Nuclear Emergency Response Center

The Regional Nuclear Emergency Response Center (RNERC) is composed of staff from the local government. Its major missions are the notification and implementation of protective actions to the public, including traffic control, notification of in-house sheltering, distribution of iodine tablets, reception and accommodation of the public, medical cares for the injured people and comforting of the affected inhabitants.

(4) Nuclear Emergency Support Center

The Nuclear Emergency Support Center (NESC) is composed of experts from the military units. Its major missions are decontamination for the public, vehicles and environment, and assisting the implementation of the public protective actions, including evacuation, sheltering, medical cares, distribution of iodine tablets, traffic control, and safeguarding the affected area. This center also provides supports for the radiation detection and monitoring of the affected area.

16.1.1.2 Emergency Response Organizations of the Licensee and Their Missions

The licensee of a nuclear reactor facility shall set up a dedicated Nuclear Emergency Response Unit, and activate the Nuclear Emergency Response Organization within the facility upon the occurrence or possible occurrence of a nuclear accident. The Nuclear Emergency Response Unit is responsible for the support and coordination of the response activities inside the facility. Also this Unit has to evaluate the accident situations and possible radiation dose impact, and cooperate with NNERC, RMDAC, NESC and RNERC for proceeding of the related emergency response measures. This Unit shall report and keep in contact with the AEC and related Centers under the emergency condition, and request for support from possible resources outside the facility, if needed. The Nuclear Emergency Response Organization (i.e. the technical support center, TSC) is composed of the experts from the facility. The major missions of this organization are taking the compensatory measures to bring the nuclear power unit under control, supervising the effectiveness of the response measures inside the facility, and providing the information regarding the accident to the related organizations.

16.1.1.3 Categorizations of Nuclear Accidents

Not all the abnormal incidents occurred in a nuclear power station have a need for emergency response. Even when a nuclear accident does occur, there will be no threats to the general public if appropriate counter measures are taken by the plant operators. In order to effectively formulate the steps of emergency response and to determine proper protective actions for the public, nuclear accidents are divided into the following three categories according to the possible severity of their impact.

(1) First Category: Alert

When an abnormal event in a nuclear power unit deteriorates to the extent that specific safety parameter(s) seriously exceeds the Technical Specifications limit and probably threatens the safe operation of the unit, it is defined as an Alert event (AE). A small amount of radioactive materials may be released in this kind of event. However, the safety limit associated with the protection of public health is not exceeded. Therefore, there is no need to perform any protective actions for the public. In accidents of this category, the TSC will be activated to respond to the condition. The Nuclear Emergency Response Unit of the TPC will evaluate the possible impact of the accident and prepare for necessary response activities. The AEC will establish an emergency team, and notify the RMDAC, RNERC and NESC to standby, based on the accident situation and its possible impact. The AEC may notify the RMDAC for assembly and preparedness if needed.

(2) Second Category: Site Area Emergency

When a nuclear power unit encounters a major accident that causes severe damages to the safety systems and endangers the safety of the unit, a declaration of “Site Area Emergency accident” (SAEA) will be needed. In case of this accident, the TPC has to activate its whole in-house emergency response structure to perform the necessary response. The AEC will establish an emergency response team and inform the RMDAC to activate. The AEC may also notify the RNERC and NESC for assembly and preparedness. Protective actions for the offsite residents may be needed during this category of accident.

(3) Third Category: General Emergency

When a nuclear power unit encounters a major accident that may cause damage of the reactor core and failure of the containment integrity, a “general emergency accident” (GEA) will be declared. In this kind of accident, all the emergency response organizations need to be activated. The most proper protective actions for the offsite residents need to be performed immediately as well.

16.1.1.4 Emergency Planning Zone

In case of a nuclear accident for which the preliminary protective measures or evacuation of the local residents are required, a question of how large the affected area should be will need to be answered. Besides the public safety consideration, cost-effectiveness should be another vital factor. An emergency planning zone (EPZ) for the preparation of emergency responses for all nuclear accidents was determined for each of the existing nuclear power plants. The size of the EPZ is closely related to the type of the reactor, the population density around the plant, the local topography, the local weather conditions, etc. Being identified in the Enforcement Rules for the Implementation of the Nuclear Emergency Response Act, the guidelines to determine the EPZ boundary are as follows:

- The predicted radiation dose outside EPZ resulting from design basis accidents shall not exceed the evacuation intervention criteria stipulated in the Nuclear Emergency Public Protective Action Guides.

- The annual probability of the predicted radiation dose outside the EPZ resulting from a core meltdown accident exceeding evacuation intervention criteria stipulated in the Nuclear Emergency Public Protective Action Guides shall be less than three in 100,000.

- The annual probability of the predicted radiation dose outside the EPZ resulting from a core meltdown accident exceeding 2 Sv shall be less than three in 1,000,000.

The analysis of the accident dose and risk distribution for the EPZ was made by applying the computer code MACCS2 with the following input information: the possibility of radiation release, the weather conditions, the population distribution around the plant, and other related parameters. Based on the above-mentioned guidelines, methods and analysis results, and the dose limits of the “Nuclear Emergency Public Protective Action Guides” as well as following a detailed analysis with social and economic considerations, the EPZ for the three operating nuclear plants were all set as a circle with 5 kilo-meters radius from the center of the nuclear power station. The boundary of EPZ of each nuclear power plant will be re-evaluated by TPC every 5 years. The population and meteorological data will be updated in the re-evaluation.

Within the EPZ, all the preparedness must be ready at all time, including the public alert system, rendezvous points, evacuation routes, the reception and accommodation center, etc. Drills should be conducted periodically to evaluate the feasibility of the preparedness and response arrangements, to see whether the staff react according to the procedures, to check the functions of relevant hardware and software, and finally to find out whether the nearby residents are used to the practice so as to enhance the efficiency and effectiveness of the emergency response plan.

16.1.2 Planning, Training and Drill

16.1.2.1 Emergency Response Basic Plan and Public Protection Plan

Based on the Nuclear Emergency Response Act, the AEC shall consult all designated agencies to lay down the Emergency Response Basic Plan and the Nuclear Emergency Public Protection Action Guides. The contents of the Emergency Response Basic Plan include the missions of the response organizations, their routine preparedness measures, the exercise of the emergency response plan, notification and activation of each response organization, and the recovery measures after the nuclear accident. This basic plan provides the basic guides for the AEC, RMDAC, NESC and RNERC for their detailed planning to enhance their capability for emergency response.

The regional competent authority shall lay down the Public Protection Plan based on the Emergency Response Basic Plan and the Nuclear Emergency Public Protection Action Guides. The contents of this Plan shall include categorization of the nuclear accident, mission of the organization, facility layout of the response center, notification of the accident and activation of the organization, routine preparedness measures, and recovery measures after the accident.

16.1.2.2 Emergency Response Plan of the Nuclear Reactor Facility Licensee

The nuclear reactor facility licensee shall follow the provisions laid down by the AEC to define the EPZ in the surrounding area of the nuclear reactor facility. The area of the EPZ should be reviewed and revised every five years. The licensee shall periodically submit the analysis and planning of the public protective measures within the EPZ to the AEC for approval. The analysis and planning of the public protective measures within the EPZ includes the population distribution, radiation monitoring program, public alert system, and the assembly, evacuation and accommodation of the public. The licensee shall set up necessary places and equipment according to the approved analysis and planning. The licensee shall also draw up the Emergency Response Plan for the nuclear reactor facility. The contents of the plan include categorization of the nuclear accident with its justification procedure and method, the mission of the emergency response organization, the related routine preparedness measures and recovery measures after the accident. A newly constructed nuclear reactor facility shall define the EPZ, the proposed public protective plan within the EPZ and the emergency protection plan for the facility to be submitted to the AEC for approval before initial fuel loading.

16.1.2.3 Training and Routine Equipment Testing

To assure the knowledge and skill of the emergency response for the personnel involved in the response actions, periodical training courses together with the equipment testing and maintenance are held in each nuclear power station and the designated agencies. The scope of training includes emergency operating procedures (EOP), rescue of injured persons, and emergency repair of damaged equipment. Inspectors from the AEC are responsible for auditing the effectiveness of these courses. As for those from the central government, local government and military agency who are responsible for the emergency response, regular training courses in the areas of nuclear accident basics, emergency communication, radiation protection, disaster countermeasures, sheltering and evacuation, etc., are held by the NNERC. Special trainings on the decontamination of personnel, vehicles and roads are performed by the relevant military agency on regular basis as well. In addition, introductory lectures for the emergency response are given to the local residents living inside the area of the EPZ every time before an off-site drill was conducted. Primary and high school teachers in this area are trained to instruct their students about the knowledge of protective actions in case of a nuclear accident. Brochures as well as the audio and video compact discs about emergency response are also distributed in the EPZ area associated with each nuclear power station every year.

In order to insure all the facilities and equipment are in normal operating condition, the nuclear reactor facility licensee and the emergency response organizations shall perform the maintenance and functional testing according to the pre-approved maintenance program. Testing and maintenance of the emergency communication equipment is the important portion of the program.

16.1.3 Nuclear Safety Duty Center

The AEC is responsible for proactively monitoring the operational data and radiation status of the existing operating nuclear power plants at any time. The AEC also maintains close contact with domestic and foreign nuclear organizations for the purpose of communication, technical support, and/or accident reporting. The Nuclear Safety Duty Center (NSDC) is established in the AEC to integrate all the related functions to enhance the performance of the emergency response. The NSDC is operated on a basis of 24 hours a day and all year round to monitor the operating status of nuclear power plants, collect data, train the relevant personnel, and communicate with the general public regarding the nuclear safety. In case of a nuclear accident, the NSDC is immediately acted as the pivotal center for emergency response to perform the initial event analysis and dose evaluation. The main functions and capability of this center are illustrated as follows:

(1) Video Conference System: This is an integrated system serving as a remote control channel during accident or abnormal conditions. Normally, a weekly video conference test among the AEC, TPC headerquarters and NPPs will be performed.

(2) Safety Parameter Display System: This system provides the NPPs' safety parameters on a real time basis. When an emergency occurs, these parameters will provide the vital first-hand information for event analysis in the AEC.

(3) Environmental Radiation Monitoring System (ERMS): There are a total of 30 radiation monitoring stations around this country, providing the real time environmental radiation information nationwide. Through this ERMS, the radiation status around the Taiwan and Kinmen area are displayed at the center.

(4) NPP Site Boundary Radiation Monitoring System: There are five radiation stations at each NPP to monitor the radiation level at the plant site boundary area. All these information are real-time displayed at this center.

(5) Dose Evaluation System: Under the accident condition, this system is capable of performing dose evaluation based on the online radioactive gaseous effluents and meteorological data. The calculated results are utilized as an input to the accident management and event control. This system may also perform the dose evaluation based on the predicted gaseous release of the accident and the predicted meteorological information from the Central Weather Bureau.

(6) Meteorological Information: The real-time meteorological information is provided from the Central Weather Bureau. Of which the wind field is one of the key parameters in projecting the public dose under accident condition.

(7) Cable TV News Channel: There are twelve news channels equipped to allow simultaneous news acquisition from domestic and foreign reporting. The multi-sources of news enable the worldwide nuclear event to be alerted in its early stage.

(8) Satellite Communication System: In case of a total breakdown of all communication mechanism, the satellite telephone can be used as an important backup for the AEC to communicate with other organizations.

(9) Dedicated Hot Line Phone: Dedicated hot lines have been installed for immediate and direct contact with each nuclear power plant and the Taipower headquarters during emergency. The lines are tested everyday by the on-duty staff.

16.2 Notification and Protection of the Public

16.2.1 Implementation of Emergency Preparedness

(1) Notification and Activation of Emergency Response Organizations

The activation of the emergency response organizations is based on the nuclear accident categories and the possible release of the radioactive materials. The Nuclear Emergency Response Unit and the Nuclear Emergency Response Organization of the nuclear facility will be activated under the alert category of the nuclear accident. The RMDAC and NNERC will be activated and established under the site area emergency accident. The NESC and RNERC will standby as notified by the AEC under the alert accident, and will be assembled and established according to the notification by the NNERC. The notification and motivation of these emergency response organizations based on the accident category and possibility of release of radioactive materials are shown in Table 16.1.

(2) Notification of the General Public

In case of a severe nuclear accident that may affect the residents in the EPZ, the NNERC is responsible for providing the public the correct and complete information. In general, the ways of notifying the public include radio, TV, broadcasting cars and emergency siren systems set up at police stations in the EPZ.

(3) Protective Actions for the General Public

In case of a severe nuclear accident that may affect the offsite environment, the protective actions for the residents in the EPZ will include sheltering and evacuation. These actions are performed according to the criteria described in Table 16.2. Medicines for radiation dose reduction (e.g. Iodine Tablet) are prepared for all the evacuees. Accommodation stations will be established at some appropriate places outside the EPZ to accommodate the personnel evacuated from the EPZ. Personnel and vehicles need to be checked for radiation contamination before entering these stations. De-contamination processes will be executed wherever necessary. The RNERC is responsible for providing the evacuees water, food, medicines and other necessary assistance and the NESC will provide decontamination of the personnel, vehicles and road as well as the traffic control, relocation of the personnel, emergency medical care, and Iodine Tablet distribution.

(4) Protective Action Guides

In case of a radioactive material release to the offsite areas of a nuclear power station, the decision on whether the offsite residents need to take shelter or to evacuate or other protective actions need to be taken is based on the predicted radiation exposure as listed in Table 16.2, which forms the protective action guide (PAG) used by the AEC. As suggested in ICRP-63 and IAEA-115 reports, the projected dose and the avertable dose are used to define the dose limit for the intervention level for protective actions. The projected dose means the evaluated dose when no protective action is taken, while the avertable dose means the dose may be saved when protective actions are taken as compared with the projected dose. On the other hand, the decision of food edibility in the contaminated area is based on the control standards of the equivalent concentrations of radionuclides in milk, infant foodstuffs or drinking water and the ordinary food as listed in Table 16.3.

16.2.2 Exercise

To assure the effectiveness of the emergency response actions, both on-site and off-site emergency response drills are held periodically. For the on-site drill, once a year is required for each nuclear power plant. The items of the on-site exercise include notification and information transmission, activation and response of the emergency organization, rescue of the accident condition, accident impact evaluation, nuclear security, radiation monitoring, and dose evaluation. The scenario of each drill is planned in the TPC Headquarters and kept confidential beforehand. An evaluation group, consisting of scholars, government officials and civilian representatives, is organized to oversee the performance of the drill. Recommendations from this group are documented for the TPC for improvement.

For the off-site emergency response, a full scope exercise was held in this country every two years before 2001. However, the frequency has been changed to once a year after 2002 as required by the government. Currently the south part nuclear facility (i.e. Maanshan) and one of the north part nuclear facilities (i.e. Chinshan and Kuosheng) held the exercise by turns. It will be changed to a north-north-south cycle when the Lungmen NPP enters into commercial operation. The items of the off-site exercise include notification and information transmission, activation and response of the emergency organization, accident impact evaluation, protective actions for the public, radiation monitoring and dose evaluation, radiation decontamination, and related recovery measures. The participating organizations in this drill include all Ministries involved in the NNERC, RNERC, RMDAC, NESC, and TPC. In addition, about one percent of the residents in the area of the EPZ are invited to participate in the evacuation practice of each drill. The performance of each drill is assessed by a group of experts similar to the evaluation group for the on-site drill. The recommendations on further improvements will be followed up by the AEC.

16.2.3 Recovery Measures

In order to make the affected regions recovered promptly to normal conditions, the AEC shall call upon relevant government agencies of various levels and the nuclear reactor facility licensee to activate the Nuclear Emergency Recovery Committee to take recovery measures. The Committee consists of 19 to 23 members from the AEC, relevant government agencies, the nuclear reactor facility licensee and the public representatives from the affected regions. The responsibility of this committee includes determining recovery measures, supervising the implementation of these measures, notifying relevant government agencies of various levels and the nuclear reactor facility licensee to implement relevant recovery measures, coordinating the dispatched manpower and resources for recovery, announcing orders for public protective actions during the recovery period, issuing press release for recovery, and carrying out any other recovery measure. The missions of the relevant organizations are as follows:

(1) Ministry of Interior

The Ministry of Interior (MOI) is responsible for: (a) supervising the local government to assist the public in the affected regions for temporary relocation or permanent accommodation, reconstructing the community, and searching the missing personnel; (b) supervising the supply and storage of necessary daily stuffs for the public in affected regions, and maintaining the necessary police and fire protection force for the affected regions; and (c) planning and conducting the recovery of the contaminated national park near the affected regions.

(2) Ministry of National Defense

The Ministry of National Defense (MND) supervises the military force to support the radiation monitoring, to support the local government for area control and transportation of the public, to conduct the decontamination of personnel, vehicle and road in affected regions, and to arrange vehicles for the recovery related measures.

(3) Ministry of Finance

The Ministry of Finance (MOF) is responsible for reduction or deferring of the land tax and customs duties in affected regions, and adjusting the rate of the import tax or the amount of the quota as needed by the condition of the disaster.

(4) Ministry of Economic Affairs

The Ministry of Economic Affairs (MOEA) supervises the nuclear reactor facility licensee to perform the recovery measures, and the affiliated organizations to control the contaminated water resources and adjust the water supply, and to regulate the electricity and the necessary stuffs for the public’s livelihood.

(5) Ministry of Transportation and Communication

The Ministry of Transportation and Communication (MOTC) supports the evaluation and planning of the road required for the recovery measures and acquisition of vehicles required for the recovery measures, and planning and conducting the recovery of contaminated national scenic spots near the affected regions.

(6) Directorate-General of Budget Accounting and Statistics

The Directorate-General of Budget Accounting and Statistics (DGBAS) provides the local government of the affected regions the financial support required to perform the recovery measures.

(7) Government Information Office

The Government Information Office (GIO) collects the accident recovery and emergency response information, and issues the press release about the recovery.

(8) Department of Health

The Department of Health (DOH) supervises the medical care for the public in affected regions, planning and dispatching medical supplies for the recovery measures, and evaluating the radiation injuries. It is also responsible for the health insurance and medical care related items for the affected public.

(9) Environmental Protection Administration

The Environmental Protection Administration (EPA) evaluates the non-radiological environmental impact and environment protection, supports for the recovery of the contaminated environment, and make the recommendation for the transport, processing, and disposal of the contaminated waste.

(10) Financial Supervisory Commission

The Financial Supervisory Commission (FSC) coordinates the deferring or reduction of the insurance fee for the affected public, and provides assistances of the insurance compensation or preferential financial measures for the public.

(11) Atomic Energy Council

The Atomic Energy Council (AEC) provides the technical consultation for the recovery measures, and supervises the licensee for the radiation detection and protection, radiation dose and contamination evaluation, etc., as needed in the recovery measures. The AEC shall also plan the decontamination measures, including transportation, processing and disposal of the contaminated waste, coordinate the technical support from foreign countries, identify the radiation affected regions based on actual radiation detection, assist the public for the nuclear damage compensation related cases, summarize the damage situation, and issue the contamination certificate.

(12) Council of Agriculture

The Council of Agriculture (COA) coordinates the supply of agricultural produces in affected regions, summarizes and reports their damage situation, supports the control and recovery of the agriculture in the affected regions, coordinates the organizations of financial support for the recovery measures, supports for the recovery measures to deal with the contaminated agricultural produces, and plans for their protection afterwards.

(13) National Communications Commission

The National Communications Commission (NCC) coordinates the communication organizations for the normal communication in the affected regions, and provides the emergency communication measures as needed.

(14) Local Government

The local government compiles the recovery plan to coordinate and assist the re-construction and notification of the affected public, to conduct temporary relocation and permanent accommodation for the affected public, to handle the non-radiological waste to protect the public, and to enhance the public security and traffic control.

(15) Nuclear Reactor Licensee

The nuclear reactor licensee shall recover the damaged nuclear facility, perform radiation monitoring, dose evaluation, and protective measures needed in affected regions, and assist decontamination and transport, processing and disposal of contaminated waste.

16.2.4 Compensation for Nuclear Damage

The financial compensation program for the liability claims arising from nuclear accidents is described in Subsection 11.1.5 of Article 11 of this report. However, some important requirements associated with the compensation for nuclear damage are emphasized in this section. The Nuclear Damage Compensation Law with the latest version promulgated on May 14, 1997 is enacted according to the Article 29 of the Atomic Energy Act (amended in 1971). This Law applies to the compensation for the nuclear damage resulting from the peaceful use of atomic energy. When a nuclear incident occurs in a nuclear installation or during the transport of nuclear materials belonged to the installation, the operator of the installation thereof shall be liable for the compensation of the resulted damage. This liability is regardless of whether the incident is caused through intention or negligence, except when it is caused directly by international armed conflicts, hostilities, domestic rebellion, or grave natural calamity. In case the operator can prove that the occurrence or expansion of nuclear damage was caused by the victim's intentional action or negligence, the court may reduce or dispense with the compensation.

The liability of a nuclear installation operator for nuclear damages arising out of each single nuclear incident shall be limited to four billion two hundred million NT Dollars (4.2 billion NT Dollars). A nuclear installation operator shall maintain liability insurance or financial guarantee sufficient to cover the maximum amount of nuclear damage compensation liability. However, this stipulation is not applicable to the nuclear installations of the central or local government and their research organizations. In respect of operation of a nuclear installation or transport of nuclear material, applications may be filed with the AEC for the reduction of the amount of liability insurance or financial guarantee within a certain limit. Should the amount received from the liability insurance or financial guarantee not sufficient to cover the finalized nuclear damage compensation, the government shall loan the balance to the nuclear installation operator to cover its complete liability; but only to the maximum amount that the operator is liable.

According to the Article 28 of Nuclear Damage Compensation Law, claims of compensation for nuclear damage shall be extinguished if an action is not brought within three years after knowledge of the damage and of the nuclear installation operator liable for the damage; however, the period shall in no case exceed ten years from the date of the nuclear accident. After the occurrence of a nuclear accident, the AEC may organize an Advisory Committee on Nuclear Accident Investigation and Evaluation to perform the duties and exercise the rights as follows:

(1) Determination of the extent of a nuclear accident and investigation of the cause thereof,

(2) Investigation and evaluation of the nuclear damage,

(3) Recommendation on compensation, relief and rehabilitation measures for the nuclear accident, and

(4) Recommendations on improvements of safety of the nuclear installation.

Reports of the aforementioned investigation, evaluation, and recommendations shall be prepared for public announcement. When the victims of a nuclear accident seek compensation by way of a judicial proceeding, the court may take into account these reports.

16.3 International Framework and Relationship with Neighboring Countries

To promote the domestic technology of the emergency preparedness and to enhance the capabilities of personnel involved in the activities, the AEC actively engages in the cooperation with relevant international organizations. Several important activities in this area, completed or still ongoing, are described as follows:

(1) As a member of the Emergency Notification System of the IAEA, regular communication tests between the AEC and the headquarters of the IAEA have been performed for several years, and participation of the emergency drills held by the IAEA have also been fulfilled regularly.

(2) Since all nuclear power units in this country were imported from the United States, many groups of engineers have been sent to American organizations for training in the areas of emergency medical care, assessment of the EPZ, planning of the emergency response, etc. On the other hand, many experts from the governmental agencies, national laboratories and utilities of the United States were invited here to exchange the information of the emergency preparedness with local officers and engineers. In addition, participation in the international research projects organized by the USNRC on severe nuclear accidents, such as CSARP, COOPRA and CAMP, were proved very fruitful.

(3) To cooperate with the neighboring country in the area of emergency preparedness, a bilateral emergency support agreement on nuclear accidents has been signed by the AEC and the Japanese Atomic Industrial Forum. A lot of activities, such as safety seminars, information exchange and exchange of experts and governmental officials and staff, have been performed through this agreement.

Table 16.1 Notification and Activation of Emergency Response Centers

	Alert*		Site Area Emergency*		General Emergency
	A	B	A	B	
Nuclear Emergency Response Organization	Activated	Activated	Activated	Activated	Activated
Nuclear Emergency Response Unit	Partially Motivated	Activated	Activated	Activated	Activated
RMDAC	Standby	Assembled and Prepared	Established	Established	Established
NNERC	—	—	—	Established	Established
RNERC	—	Standby (Notified by AEC)	Assembled and Prepared (Notified by AEC)	Established (Notified by NNERC)	Established (Notified by NNERC)
NESC	—	Standby (Notified by AEC)	Assembled and Prepared (Notified by AEC)	Established (Notified by NNERC)	Established (Notified by NNERC)

* A: Without Release of Radioactive Material
B: With or Potentially With Release of Radioactive Material

Table 16.2 Intervention Levels for Protective Actions

Dose Limit	Protective Actions
Avertable Dose of 10 mSv in 2 days	All residents need to take sheltering inside the house
Avertable Dose of 50 to 100 mSv in 7 days	Residents to be evacuated from EPZ
Avertable Thyroid Equivalent Dose of 100 mSv	Take Iodine Tablet
Projected Dose of 30 mSv in 30 days	Temporary relocation (To be terminated when Projected Dose below 10 mSv in 30 days)
Expected Lifetime Dose Greater than 1 Sv, or Temporary Relocation over 1 Year	Permanent Relocation

Table 16.3 Food and Drinking Water Control Standards

Radionuclide	Action Level (Kilo-Bq/Kg)	
	Food	Milk, Infant Foodstuffs, Drinking Water
Cs-134,Cs-137,Ru-103,Ru-106,Sr-89	＞1	＞1
I-131	—	＞0.1
Sr-90	＞0.1	—
Am-241,Pu-238,Pu-239	＞0.01	＞0.001

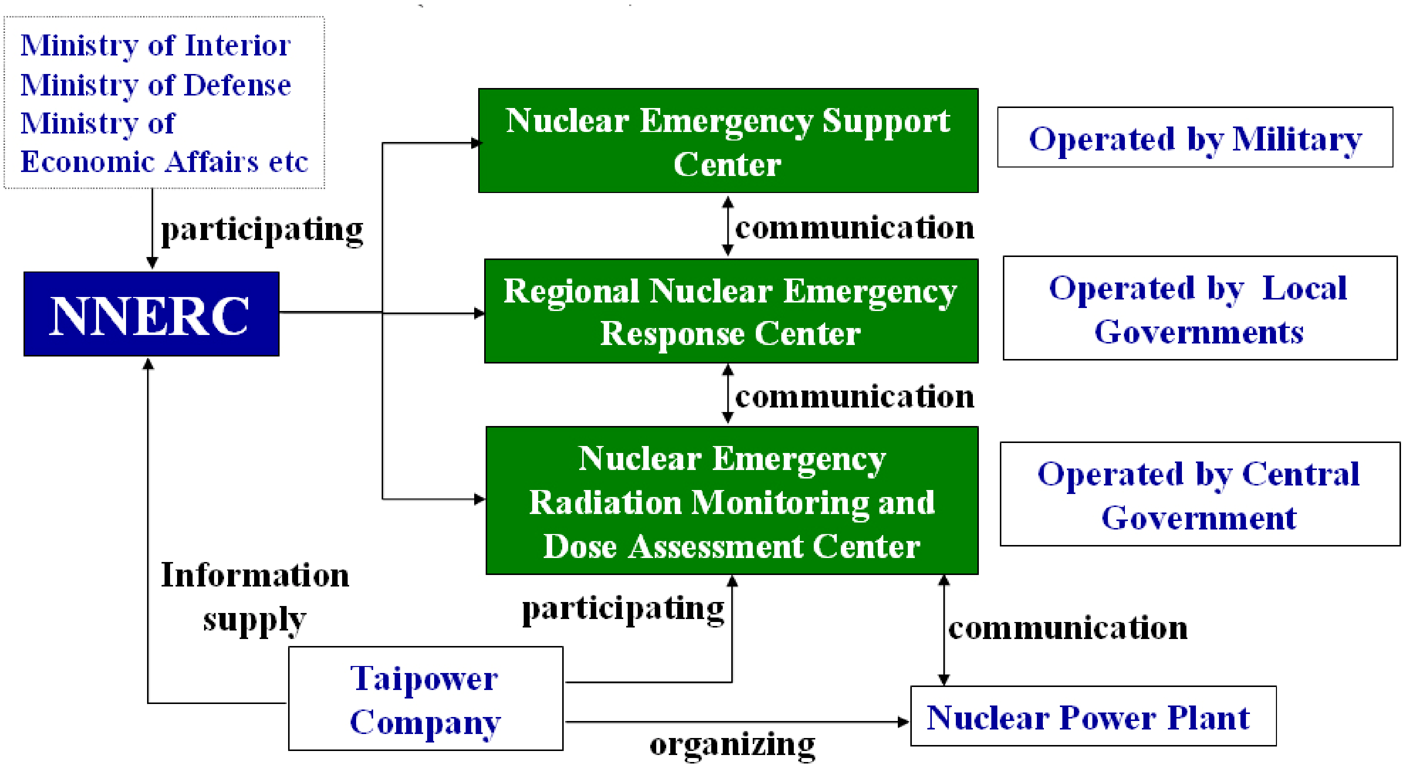


Figure 16.1 Organization of National Nuclear Emergency Response Center

(NNERC, Operated under AEC)

ARTICLE 17. SITING

Each Contracting Party shall take the appropriate steps to ensure that appropriate procedures are established and implemented for

(i) evaluating all relevant site-related factors that are likely to affect the safety of a nuclear installation for its projected lifetime

(ii) evaluating the likely safety impact of a proposed nuclear installation on individuals, society, and the environment

(iii) 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) 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

17.1 Evaluation of Site-Related Factors

The siting requirements are mainly contained in the Enforcement Rules of the Nuclear Reactor Facilities Regulation Act. Other major Codes and Standards for the site selections required by the country of origin (here referred to USA) are listed as follows:

- 10 CFR Part 100 -- Reactor Site Criteria
- USNRC Regulatory Guide 4.7 -- General Site Suitability Criteria for Nuclear Power Stations
- USNRC Standard Review Plan 2.5.1 -- Basic Geologic and Seismic Information.

These Codes and Standards required by the country of origin are considered by the AEC as important reference documents. However, the requirements set forth in these Codes and Standards are not mandatory. In order to fulfill the regulatory siting requirements, the evaluations of site-related factors that are likely to affect the safety of the plant are documented in the PSARs and FSARs of the existing domestic NPPs. The important considerations of these evaluations are described in the following sections.

17.1.1 Nearby Industrial and Military Facilities and Transportation

According to the relevant regulatory requirements, the applicant of a new reactor site should provide information on the nearby industrial and military facilities as well as the transportation routes, and evaluate the potential external hazards. The applicant should also identify any situations in the vicinity of the plant which have potentials for accidents, such as explosions of hazardous materials, delayed ignition of flammable vapor clouds, liquid spills and release of toxic vapors, fires, accidents at sea, etc., and assess the potential effects of these situations on the safe operation of the nuclear facility.

To fulfill theses requirements, important and relevant evaluations performed by the licensee of the existing domestic NPPs include river traffic accidents, explosions, fires, accidental releases of toxic gases, airplane crashes, airborne pollutants, and so on. Generally speaking, the potential hazards caused by the nearby industrial and military facilities and the transportation means are negligible to these NPPs.

17.1.2 Meteorology

The evaluations of site-related factors associated with meteorology include heavy precipitation, typhoons, thunderstorms, tornadoes, strong winds, and tsunami. In order to collect data for meteorological evaluation, each NPP performs an "Onsite Meteorological Measurement Program" before and after the commercial operation. In this program, the meteorological variables under observation before operation include wind, temperature, precipitation, sunshine rate, elevation of sun, insolation, evaporation, cloud conditions, atmospheric pressure, humidity, wind aloft, and temperature gradient. While after operation, the meteorological monitoring system to be kept includes wind speed and direction, temperature gradient, and humidity (or dew point).

17.1.3 Hydrology

The site-related factors to be evaluated in association with hydrology include probable maximum flood, probable maximum precipitation, precipitation losses, coincident wind-wave activity, combination of natural events, probable maximum tsunami flooding and so on.

17.1.4 Geology and Seismology

The evaluations associated with geology and seismology are required to determine site suitability and to provide reasonable assurance that a nuclear power station can be constructed and operated at a proposed site. The structures, systems and components (SSCs) of safety systems shall be designed to withstand appropriate seismic forces. The major considerations for these evaluations include:

(1) Basic Geology and Seismic Data

The data associated with regional and site physiography, regional geology and tectonic, site geology, structural geologic map, geologic profiles (presenting the relationship of the foundations of the nuclear power plant to subsurface materials), history of groundwater fluctuations, subsurface investigation, seismic and velocity surveys, static and dynamic rock properties, and excavation and backfill are collected and analyzed for geology and seismology evaluations.

(2) Vibratory Ground Motion

The analyses associated with the vibratory ground motion include those on: regional and site tectonic structures, prior earthquake behavior of surficial and subsurface materials, static and dynamic soil properties, previous regional earthquake data, correlation of epicenters with tectonic divisions, active faults, vibratory ground motion at the site for structure related earthquakes, vibratory ground motion at the site for site tectonic province related earthquakes, maximum ground acceleration at the site and design basis earthquake, operating basis earthquake, etc. The design for the Design Basis Earthquake is intended to assure that:

- The integrity of the reactor coolant pressure boundary is not compromised;
- The capability to shut down the reactor and maintain it in a safe condition is not compromised; and
- The capability to prevent or mitigate the consequences of accidents which could otherwise result in potential offsite exposures comparable to the limiting exposures of the Enforcement Rules for the Implementation of Nuclear Reactor Facilities Regulation Act, is not compromised.

17.1.5 Experience of Site Selection from the Lungmen Nuclear Project

Originally there were four potential sites selected for the project of the Lungmen nuclear power plant. Many factors have been considered in the selection, which could be classified into two categories -- the most important factors and the important factors:

(1) Most Important Factors

- Geology
- Seismology
- Foundation conditions
- Environmental impact

(2) Important Factors

- Accessibility
- Land use and acquisition
- Power transmission
- Population
- Meteorology
- Oceanography
- Hydrology
- Site development
- Radiation dose consideration.

An assessment of the most important siting factors of the four candidates has been made and the results are shown in Table 17.1 with rating from 1 to 4. Besides this, a weighting factor is added on top of the ratings such that a factor of 4 is assigned to the “most important factor” while a weighting factor of 2 is assigned to the “important factor”. By combining the score and the weighting factor, Yenliao was selected as the site of the Lungmen nuclear project as shown in Table 17.2.

17.2 Evaluation of Safety Impact on Individuals, Society, and the Environment

17.2.1 Regulatory Requirements for Environmental Impact Assessment

According to Article 5 of the Environmental Impact Assessment Act, an environmental impact assessment (EIA) shall be conducted for the development activities which are likely to have adverse impacts on the environment. Therefore, the exploitation of nuclear energy and the construction of radioactive waste storage or treatment facilities are required to conduct the EIA. During the planning stage, the project developer of the development activities shall conduct, in accordance with the Working Guidelines for EIAs, a Phase I EIA and prepare an environmental impact statement ("EIS"). When applying for a permit related to the proposed development project, the project developer shall submit the EIS to the responsible agency for the enterprise associated with the project, which will then transfer the EIS to the Environmental Protection Administration (EPA). The EPA shall, within fifty (50) days of receiving the EIS mentioned above, publish its conclusions related to the EIS review and notify the responsible agency for the enterprise associated with the project, as well as the project developer, of these conclusions. The review period may be extended for another fifty (50) days under unusual circumstances. If the conclusions reached by the EPA show that the development activities are likely to have significant adverse impacts on the environment, the project developer must conduct a Phase II EIA. According to Article 11 of the Environmental Impact Assessment Act, the nuclear project developer shall prepare a draft environmental impact assessment report [("Draft EIA Report")] and submit it to the AEC. The contents of the Draft EIA Report are provided in Table 17.3.

Within thirty (30) days of receiving the Draft EIA Report, the AEC shall, in conjunction with the EPA, members of the Environmental Impact Assessment Review Committee and other relevant agencies, invite experts, scholars, non-governmental groups and local residents to conduct an on-site inspection and hold a public meeting to explain the development activities. The inspection record, public meeting minutes and the draft EIA Report shall be submitted to the EPA.

The EPA shall conclude its review within sixty days and provide the conclusions to the AEC and the project developer. The project developer shall revise its Draft EIA report in accordance with the EPA's review conclusions and prepare an EIA report [("Final EIA Report")] for approval by the EPA (in accordance with the review conclusions). Upon approval by the EPA, the Final EIA Report and a summary of the review conclusions shall be published in the EPA register. The review period may be extended for another sixty days under unusual circumstances.

During the construction period, the environmental impacts considered in an EIA report include air quality, noise, transportation flow, water quality, terrestrial ecology, aquatic ecology, solid waste, and historic and archaeological resources. While after operation, the impacts will include radiation and thermal pollution. The EIA report of the Lungmen Nuclear Project was approved in 1991.

17.2.2 Evaluation of Radiological Consequences

According to Article 3 of the Enforcement Rules for the Implementation of Nuclear Reactor Facilities Regulation Act, the area surrounding the nuclear facility shall be divided into the two following regions based on the possible damage resulted from the design-basis nuclear accidents:

(1) Exclusion area: An exclusion area (EA) is the area surrounding the reactor that an individual at its boundary for two hours immediately after the onset of a postulated fission product release would not receive a total radiation dose to the whole body in excess of 250 mSv (25 rem) or a total radiation dose in excess of 3 Sv (300 rem) to the thyroid from iodine exposure.

(2) Low population zone: A low population zone (LPZ) is the area surrounding the exclusion area that an individual at its outer boundary who is exposed to the radioactive cloud during the entire period of its passage would not receive a total radiation dose to the whole body in excess of 250 mSv (25 rem) or a total radiation dose in excess of 3 Sv (300 rem) to the thyroid from iodine exposure.

In addition to the dose criteria, Article 4 of the Nuclear Reactor Facilities Regulation Act requires that the distance from the nuclear facility to the nearest boundary of a densely populated center with 25,000 or more residents must be at least one and one-third times of the radius of the low population zone. Furthermore, except highway, railroad, or waterway, the licensee shall obtain the full ownership control of the land within the exclusion boundary during the intended utilization period. Every site must have a designated low population zone immediately surrounding the exclusion area boundary so that there is a reasonable probability that appropriate protective measures could be taken in a serious accident. According to Article 4 of the Act, residence within the low population zone is generally permitted. However, for a newly established school, works, jail, hospital, long term nursing institute, or recuperation and convalescent institute (charity) for the aged, protective measures shall be provided, referring first to the response plan of civil security and protection of that local area, and submitted to the AEC. After the AEC invites the government of municipality under the direct jurisdiction of the Executive Yuan and the county (city) government to review and approve those protective measures, these facilities can then be constructed and operated in accordance with the relevant laws and decrees.

In accordance with the previous requirements, data about the population within 10 Km from the reactor, population between 10 and 40 Km, transient population (mainly resulted from both seasonal variations in beach park, temple, church, and fishing and daily workday variations), population center, and the public facilities and institutions have been collected. To evaluate the range of the exclusion area and the low population zone, domestic NPPs adopt justifiable parameters for the following: fission product release fraction from the core, expected leak rate from the containment, and the meteorological conditions for the site. In addition, investigation of the atmospheric diffusion characteristics and provision of the bounding relative atmospheric dispersion factor (X/Q) were also performed for evaluating radiological consequences of the postulated design-basis accident to ensure that the safety limits are not exceeded.

Besides the dose analyses necessary to support reactor siting, all domestic NPPs have also performed evaluation of the potential increase in the consequences of accidents and radiological release that might result from the modification of the systems, structures, and components of the facility after construction. As part of the accident analysis in the FSAR, the changes in dose resulted from the design basis accidents such as large break LOCA, small break LOCA, fuel handling accidents, etc. were also performed to ensure that these changes will still comply with the dose criteria.

Table 17.1 Comparison of the Siting Factors for the Yenliao, Laomei, Kuanyin and Tawu Sites

<table>
<tr><th>Site / Factor</th><th>Yenliao</th><th>Laomei</th><th>Kuanyin</th><th>Tawu</th></tr>
<tr><td>Seismology</td><td colspan="3">•Yenliao, Laomei, and Kuanyin Sites meet USNRC geology and seismic siting criteria.
•For conservative seismic design, a peak horizontal ground acceleration of 0.4g is appropriate for SSE at these sites.</td><td>•Tawu site is very marginal and cannot meet USNRC Siting Criteria.
•SSE would be higher than 0.55g.</td></tr>
<tr><td>Geology</td><td colspan="3">Geology of Yenliao, Laomei, and Kuanyin areas are favorable sites for nuclear power station.</td><td>Tawu site is less desirable because of its proximity to a major active plate boundary and possibly capable faults.</td></tr>
<tr><td>Foundation Conditions</td><td>•Underlain by competent rock close to the ground surface and has the best foundation conditions.
•Covered by 0 to 10 meters of alluvium.</td><td>•Laomei is also underlain by sound rock, but covered by up to 30 meters of alluvium.
•Has slope stability problem.</td><td>•Has a marginal foundation for a nuclear power station.
•11 to 15 meters of overburden overlies a soft, weak rock formation.
•Has liquefaction potential.</td><td>•Tawu is the least desirable site with up to 53 meters of overburden on top of rock.
•Serious slope stability problems.</td></tr>
<tr><td>Other</td><td>•Located on the northeastern coast of the island.
•Close to the Fulong Beach.</td><td>•Extremely difficult to construct 345 kV transmission lines.
•Close to Baisa Beach</td><td>•Close to Kuanyin Beach.
•Difficulty of fresh water supply.</td><td>The population and population growth rate are very low around the site.</td></tr>
</table>

Table 17.2 Site Rating Chart for the Lungmen Nuclear Project

Factor	Weight	Rating of Site				Weiqhted Rating			
		Yenliao	Laomei	Kuanyin	Tawu	Yenliao	Laomei	Kuanyin	Tawu
Most Important Factors:									
Geology	4	4.0	4.0	3.3	2.1	16	16	13	8
Seismology	4	4.0	3.8	3.5	1.0	16	15	14	4
Foundation Conditions	4	4.0	3.0	1.8	1.0	16	12	7	4
Environmental Impact	4	1*	2*	3*	2*	4	8	12	8
Important Factors:									
Accessibility	2	4	4	3	2	8	8	6	4
Land Use ＆ Acquisition	2	2	2	3	4	4	4	6	8
Power Transmission	2	4	1	3	2	8	2	6	4
Population	2	3	3	3	4	6	6	6	8
Meteorology	2	4	4	3	2	8	8	6	4
Oceanography	2	3	3	2	4	6	6	4	8
Hydrology	2	3	4	2	2	6	8	4	4
Site Development	2	3	2	3	1	6	4	6	2
Radiation Dose Considerations	2	3	2	3	4	6	4	6	8
Total						110	101	96	74

* Rating applies only if the plant incorporates design features for minimizing impact on environment.

Best = 4 Better = 3 Good = 2 Poor = 1

Table 17.3 Contents of the Draft Environmental Impact Assessment Report

1. Name and business or office address of the project developer, 2. Name, residence or domicile and identification number of the representative of the project developer, 3. Signatures of the person(s) who conducted a comprehensive evaluation of the EIS and the person(s) contributed their opinions to particular items in the EIS, 4. Name and site of the development activities, 5. Description of the purpose and nature of the development activities, 6. Description of the environmental status, and the primary and other possible impacts of the development activities and all related plans, 7. Prediction, analysis and evaluation of the environmental impacts (of the proposed project), 8. Description of measures to mitigate or prevent adverse impacts to the environment caused by the development activities, 9. Description of alternatives to the proposed development plan, 10. Description of the comprehensive environmental management plan, 11. Description of actions taken in response to the comments of relevant agencies, 12. Description of actions taken in response to the comments of local residents, 13. Conclusions and recommendations (of the project developer), 14. The budget for implementing environmental damage mitigation measures, 15. Summary of measures to prevent and mitigate adverse impacts to the environment caused by development activities, and 16. Bibliography of references.

ARTICLE 18. DESIGN AND CONSTRUCTION

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

(i) 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 material, with a view to preventing the occurrence of accidents and to mitigating their radiological consequences should they occur;

(ii) the technologies incorporated in the design and construction of a nuclear installation are proven by experience or qualified by testing or analysis;

(iii) 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.

18.1 Protection against the Release of Radioactive Materials

18.1.1 Licensing Process and Regulatory Requirements

According to the Nuclear Reactor Facilities Regulation Act, the important processes to construct a nuclear power plant include:

(1) To define an exclusion zone (EZ) and a low population zone (LPZ),

(2) To submit the PSAR to get a Construction License,

(3) To submit the FSAR to get an intial fuel loading permit and then Operating License, and

(4) To define an Emergency Planning Zone (EPZ) and perform an emergency drill before the initial fuel loading.

The important requirements associated with the statement of “protection against the release of radioactive material” in these processes are explained as follows.

(1) The whole-body dose and thyroid dose shall be less than 250 mSv and 3 Sv, respectively, for a person at the boundary of the EZ within 2 hours in a postulated nuclear accident occurred in the plant. The owner of the plant has to purchase all land inside the EZ.

(2) The whole-body dose and thyroid dose shall be less than 250 mSv and 3 Sv, respectively, for a person at the boundary of the LPZ within 24 hours in a postulated nuclear accident occurred in the plant.

(3) Measures of protection against the release of radioactive material shall be described in the PSAR and FSAR clearly for review. Important chapters of the PSAR and FSAR associated with this topic include:

- Chapter 11 Radioactive Waste Management,
- Chapter 12 Radiation Protection,

- Chapter 15 Accident Analysis,
- Chapter 19 Severe Accident Analysis,
- App. A Probabilistic Risk Analysis, and
- App. C Emergency Plan.

(4) According to the Nuclear Emergency Response Act and its Enforcement Rules, the criteria for the EPZ assessment are as follows:

- The predicted radiation dose outside the EPZ resulting from design basis accidents shall not exceed the evacuation criteria stipulate in the Nuclear Emergency Public Protective Action Guide.
- The annual probability of the predicted radiation dose outside the EPZ resulting from a core melt accident exceeding evacuation intervention criteria stipulate in the Nuclear Emergency Public Protective Action Guide shall be less than three in 100,000.
- The annual probability of the predicted radiation dose outside the EPZ resulting from a core melt accident exceeding 2 Sv shall be less than three in 1,000,000.

The intervention level of evacuation is that the averted dose for a person near an NPP is in the range from 50 mSv to 100 mSv within the first 7 days of an accident.

Detailed requirements about the construction and operating licenses are described in the “Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities” and the “Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities” respectively.

18.1.2 Implementation of Defense-in-Depth Concept

Irrespective of reactor types, the design of all structures, systems, and components (SSCs) of a nuclear power plant should take into consideration the following internal and external events, as specified in the Nuclear Reactor Facilities Regulation Act, its Enforcement Rules, General Design Criteria (GDC) for Nuclear Reactor Facilities, and the related Regulations:

- Internal events: loss of coolant accident, main steam and high-energy line breaks, internal missiles caused by a turbine rotor, fire, flooding, and so on.
- External events: earthquakes, floods, typhoons, inflammables, poisonous gas, explosions, other anticipated man-made disasters, and so on.

The nuclear power plant shall be designed by applying the defense-in-depth principle as a safety design concept against the internal and external events as mentioned above. The major contents of this principle are as follows:

- Sufficient safety margin is secured in the design so that the probability of any design basis accident is minimized. Safety systems are designed with independency, redundancy, and diversity so that the consequences of accidents are minimized.

- Plants are designed so that even if any abnormal condition occurs due to equipment failures, operator errors or combination thereof, the reactor protection system operates automatically after detecting the abnormal condition and initiates the reactor shutdown system to prevent the abnormal condition to proceed into a severe accident.

- Plants are designed with multiple barriers, such as the fuel pellet, the fuel clad, the reactor pressure vessel (RPV) and the reactor coolant pressure boundary, and the containment building, to prevent the release of any radioactive material into the environment.

In order to assure the safety of the nuclear power plants, the TPC applies the defense-in-depth principle to the design, construction and operation of the nuclear power plants. The following basic concepts are considered for the implementation of the defense-in-depth principle into all safety related systems:

- Securing sufficient design safety margins,

- Fail-safe concept,

- Interlock concept,

- Securing independency, redundancy, and diversity,

- Multiple barriers concept, and

- In-service testability.

18.1.3 Prevention and Mitigation of Accidents

The requirements about prevention and mitigation of accidents are specified in the Nuclear Reactor Facilities Regulation Act, its Enforcement rules, General Design Criteria for Nuclear Reactor Facilities, and the related Regulations. In accordance with these regulations, the following practices are implemented in the design of the TPC's nuclear power stations.

- The reactor core is designed so that in the power operating range, the prompt inherent nuclear reactivity characteristics tends to compensate for a rapid increase in power (i.e. negative power coefficient). The reactor core is also designed to assure that power oscillations, which can result in conditions exceeding specified design limits, are not possible or can be readily suppressed.

- The reactor coolant pressure boundary is designed to have an extremely low probability of abnormal leakage and gross rupture. If any leakage of the reactor coolant system takes place, it is promptly detected to prevent against proceeding to a severe accident. It is also designed to allow periodic inspection and testing of the system to assess the structural integrity and leak-tightness.

- The emergency core cooling system (ECCS) is designed for automatic core cooling following any loss of reactor coolant at a rate such that any fuel damage that could interfere with continued effective core cooling is prevented. Even if the off-site power is lost, the necessary power of the ECCS system is to be supplied from emergency diesel generators installed in the nuclear power plant. The residual heat removal system is designed to remove the core decay heat.

The reactor protection system is installed to detect accident conditions and to maintain the reactor at a safe state by automatically initiating the reactor shutdown system and the engineered safety features (ESFs). The reactor protection system is designed with redundancy, diversity, and independence to assure that no single failure of any equipment or channel of the system results in the loss of the intended safety functions.

The following practices are incorporated into the design of nuclear power plants to mitigate any accidents including a severe accident:

- The reactor containment is designed so that if any accident occurs, the radioactive material released from the reactor coolant pressure boundary is confined and reduced over a long period. A system is installed in the containment to control the concentration of any combustible gas as it accumulates inside. The engineered safety features including the containment spray system and fan coolers are incorporated to lower the pressure inside the containment and to minimize radioactivity release.

- The emergency response facility (ERF) is installed so that if any radioactive material is accidentally released outside the nuclear power plant, the radiological effects on nearby inhabitants and the contamination to the environment are minimized. The ERF consists of the technical support center (TSC) and the operating support center (OSC). The safety parameter display system (SPDS) is installed in the following locations: main control room (MCR) of the plant, the TSC, and the TPC Headquarters, so that the major safety parameters are promptly recognized.

The main control room is designed so that even if a serious accident occurs, the operators can safely remain inside the MCR to take the necessary post-accident actions. It is possible in the MCR to monitor the operating parameters, the radioactivity inside and outside the reactor containment, the radioactive material releasing passage, and the radioactivity around the nuclear power plant in order to get control of the accident conditions and to take appropriate actions.

For example, in the Lungmen Project, the Lungmen nuclear power plant under construction in this country, two Advanced Boiling Water Reactors (ABWR) are being installed at this plant. The probabilities of radioactivity release from the containment are significantly reduced with the following features:

- The containment is filled with nitrogen gas during normal operation to reduce the probability of hydrogen burn or explosion.

- Ten passive flooders connecting suppression pool and reactor cavity are installed to assure molten core debris be cooled if the reactor vessel was melt through.

- Basaltic concrete instead of limestone concrete is used in the floor of the reactor cavity to reduce the production of non-condensable gases if there were corium-concrete interaction.

- A containment overpressure protection system (COPS) is installed to protect the containment from overpressure failure. This system releases steam flow from the upper space of the wetwell. Therefore, the radioactive level of the released flow is significantly reduced.

18.2 Application of Proven Technologies

In order to assure the safety of nuclear reactor facilities, proven engineering practice is usually required in the nuclear industry. The essential elements in the proven practice include:

- All technologies are proved by testing and experience.

- All processes of the design, construction, and operation follow approved codes and standards.

- All design and construction are performed by qualified manufacturers and constructors under their QA Program approved by TPC.

Since all nuclear reactors in this nation were imported from foreign countries, proven technologies are always the top tier requirements in the bidding processes performed by the TPC. All nuclear power reactors imported to Taiwan were required, as elaborated in the bid specifications, to design with technologies proven by operating experiences inside or outside this country. It is also required that these reactors have to be licensable in the exporting country. These requirements are usually important for the applications of construction and operating license of the new nuclear power plants.

As for the codes and standards, the Nuclear Reactor Facilities Regulation Act, its Enforcement rules, General Design Criteria for Nuclear Reactor Facilities, and related domestic regulations are the basic regulations and criteria that the TPC has to follow for the design and construction of nuclear reactor facilities. In addition, codes and standards of the exporting country such as the USNRC regulations, ASME and IEEE standards are also the important references for domestic regulator and utility to follow. Then for the qualifications of manufacturers and constructors, the TPC usually set up very stringent criteria in the bid specifications for nuclear reactor procurement. The AEC will audit the performance of manufacturers and contractors through safety review and inspections.

18.3 Consideration of Human Factors and Man-Machine Interface

The Nuclear Reactor Facilities Regulation Act, its Enforcement Rules and Regulations stipulates that the main control room, the safety parameter display system, and the remote control room shall be designed so that the results of analyzing and evaluating the human factors are reflected therein in order to maximize the safety and efficiency of the nuclear power plants. According to this provision, the analysis for the feasibility and suitability of the human engineering design are included in the PSAR and FSAR. The major contents of the analysis are as follows:

- In the design of the main control room, human factors are considered so that the man-machine interface is suitable for the safe operation of the nuclear power plants. The major factors are: working space and the environment around it, alarm and control facility, visual indicating facility, auditory signal facility, nameplates and their positioning, and layout of distributing boards.

- In the design of the safety parameter display system, the human engineering principle is considered so that the system continuously provides important safety information and the reactor operators can easily recognize them from designed location.

- The remote control room is designed in consideration of man-machine interface so that the reactor can be safely shutdown.

From the TMI accident, it showed that the operator performance is crucial to safety. Human error is one of the factors that affect the human performance. Currently, human error mitigation is being considered in the design of the Human System Interface of the main control room for the nuclear power plant as follows:

- Eliminating affordability of errors in the design phase,

- Including the training program improvement in the intelligent decision support systems,

- Providing memory aids for the maintenance personnel, e.g., portable interactive maintenance assistant,

- Training for error management, and

- Using ecological interface design.

For example, according to the FSAR of the Lungmen Project where two ABWRs are being installed, the primary goal for the Human-System Interface (HSI) design is to facilitate safe, efficient, and reliable operator performance during all phases of normal plant operation, abnormal events, and accident conditions. To achieve this goal, information, display, controls, and other interface devices in the main control room and other plant areas are designed and implemented in a manner consistent with good Human Factor Engineering (HFE) practices. Detailed HFE design and implementation process are described in Section 12.1 of this report.

ARTICLE 19. OPERATION

Each Contracting Party shall take appropriate steps to ensure that:

(i) the initial authorization to operate a nuclear installation is based upon an appropriate safety analysis and a commissioning program demonstrating that the installation, as constructed, is consistent with design and safety requirements

(ii) operational limits and conditions derived from the safety analysis, test, 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) programs to collect and analyze 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

19.1 Initial Authorization to Operate a Nuclear Installation

According to the Nuclear Reactor Facilities Regulation Act of 2003 (Article 5), to construct a nuclear reactor installation, one must have a construction license in advance and satisfy the following conditions:

- Construction of the installation is consistent with the aim of peaceful utilization of the atomic energy,

- Equipment and facilities of the installation are adequate to protect the health and safety of the public,

- Effects on the protection of environment and ecology are in accordance with the regulatory requirements, and

- Applicant’s technical and management capability and financial resources are adequate to operate the installation.

However, in order to obtain a construction license of a nuclear reactor installation, an applicant must follow the regulatory requirements as listed in the regulation entitled “Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities of 2004”(Article 3) to submit the following documents to the regulatory bodies, including mainly the Atomic Energy Council (AEC) and the Environmental Protection Administration (EPA), for review and approval:

(1) Preliminary safety analysis report (PSAR),

(2) Environmental impact assessment (EIA),

(3) Technical, management and financial capability, and

(4) Others as required and published by the regulatory body.

After receiving the above documents, the regulatory body (i.e. AEC) will issue its review conclusions in a safety evaluation report (SER) normally within one year.

Before initially loading fuel into a newly constructed reactor, the holder of the construction license must submit the following documents in the required periods for review and approval as required by Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities of 2004 as amended in 2005 (Article 2):

(1) Final safety analysis report (FSAR) – 14 months before scheduled initial fuel loading,

(2) Summary report on the corrective actions based on the inspection findings during the construction stage – 3 months before scheduled initial fuel loading,

(3) List of operating procedures – 2 months before scheduled initial fuel loading,

(4) Fuel loading plan – 2 months before scheduled initial fuel loading,

(5) Startup test plan – 2 months before scheduled initial fuel loading, and

(6) Reports on the systems’ functional tests (or preoperational tests) – before scheduled initial fuel loading.

If an approval is granted, then the initial fuel loading can be performed. To comply with the regulatory requirements addressed in Articles 13 and 14 of the Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities, it is required that the application for an operating license has to be submitted within 18 months after the initial fuel loading was approved. The applicant needs to submit the approved EIA at least one year prior to the scheduled date of operation and the following after the completion of the power tests (or startup tests) for review and approval:

(1) Updated FSAR,

(2) Summary report on results of the power tests, and

(3) Technical, management and financial capability.

19.2 Operational Limits and Conditions

The Article 6 of the Nuclear Reactor Facilities Regulation Act of 2003 and Article 2 of the Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities (as amended in 2005) require that the licensee of the construction license of a nuclear facility shall submit an application, enclosed with the FSAR, the summary on the corrective actions following inspection findings during the construction stage, list of the operational procedures, the fuel loading plan, the startup test plan, and the systems' functional test reports to the AEC for review, in order to obtain the approval of initial fuel loading. After the initial fuel loading, the licensee can officially start the commercial operation of the nuclear power reactor only after the licensing authority approves its summary report on various power tests and issues an operating license.

The technical specifications (TS or tech specs), being part of the PSAR and FSAR as required by Article 4 of the Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities and Articles 3 and 16 of the Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities, respectively, are established by the licensee to ensure the safe operation of the nuclear power plant. Included in the TS are the safety limits, the limiting safety system settings, the limiting conditions for operation (LCO), the surveillance requirements (SR), the design features of the facility, and the administrative management. Technical and administrative requirements as well as restrictions to ensure the safe operation of a nuclear power station shall be made in the technical specifications.

The limiting conditions for operation, derived from the safety analyses and test results, constitute the boundaries for the normal operating procedures and instructions. The LCOs are included in the plant technical specifications and thus need to be approved by the regulatory body. The operation of a nuclear power plant has to be in accordance with a proper set of LCOs.

The technical specifications are the important bases for both the operational safety and the surveillance test of a nuclear power plant. Because the tech specs were first completed before the nuclear power plant began to operate, timely revisions of them are required along with the operation of the plant. According to the Article 13 of the Nuclear Reactor Facilities Regulation Act and Article 8 of its Enforcement Rules as well as the Administrative Regulations of the Design Amendment and Equipment Change of the Nuclear Reactor Facilities (as amended in 2004), without the AEC's prior approval, neither a design amendment nor an equipment change shall be made, if it involves the revision of the technical specifications.

The AEC encourages licensees to use the improved technical specifications (ITS) as the basis for the plant-specific technical specifications. All three operating nuclear power plants have completed their conversion of TS from the customer's technical specifications or standard technical specifications to the ITS. A more detailed description of the updates of the technical specifications and the implementation of the ITS is provided in Subsection 6.3.4 of this report.

19.3 Operation, Maintenance, Inspection, and Testing Conducted in Accordance with Approved Procedures

According to the Regulations, the licensee is required to submit a list of the plant operational procedures for review in order to obtain an approval of initial fuel loading for a newly constructed nuclear power plant.

Listed in the FSAR of each existing operating nuclear power plant are the plant procedures including the administrative procedures, the operating and maintenance procedures, and other procedures. The administrative procedures are further classified into the following categories:

- Procedures for the shift leaders and operators,
- Special procedures,
- Equipment control procedures,
- Control of the maintenance and modification procedures,
- Master surveillance testing schedule,
- Log book usage and control procedures, and
- Temporary procedures.

On the other hand, the operating and maintenance procedures consist of two parts: (1) the control room operating procedures and (2) the other procedures. The control room operating procedures are further divided into the following categories:

- General Plant Operating Procedures

  — These procedures describe the steps required in the following plant condition changes: (i) bringing the plant from cold shutdown or hot standby condition to power operations, (ii) changing load of the plant, and (iii) bringing the plant back to hot standby or cold shutdown condition.

- System Operating Procedures

  — These procedures describe the steps required to take the individual system into or out of service. They also include the manipulation processes of the system for several normal conditions as required.

- Instrumentation Procedures

  — These procedures include the instructions for the operators to put the instrument into service, to secure the instrument out of service, and to operate the instrument for different conditions.

- System's Abnormal Procedures

  — These procedures describe the instructions for the operators to respond for abnormal system conditions.

- Alarm Procedures

  — Generally the alarm procedures are named after their alarm window position indices (panel, line and row numbers). This allows operators to easily refer to the specific alarm procedure. As for the alarm system, it is designed to give a visual (light) and an audible (sound) alarms for each window. The visible alarms are classified into two categories: "Red" for trips and "White" for alerts. Each visual alarm is initiated by a unique protective system and accompanied by a high frequency buzz noise alarm to remind the operator for taking actions. When the alarm is cleared, the annunciator system acknowledges with a low frequency buzz.

- Emergency Procedures

  — These procedures provide the instructions for the operators to handle plant emergency situations such as:

    * Earthquake,
    * Typhoon,
    * Loss of all feedwater, or
    * Loss of coolant.

  A more detailed description of the emergency procedures will be given in the following Section 19.4.

- Temporary Procedures

  — These procedures are to provide detailed instructions for the specific tests or operations of the safety related systems.

In addition to the above-mentioned procedures, there are other procedures including:

- Plant radiation protection procedures,
- Emergency preparedness procedures,
- Instrument calibration and test procedures,
- Chemical-radiochemical control procedures,
- Radioactive waste management procedures,
- Maintenance and modification procedures,
- Material control procedures, and

- Plant security procedures.

The section managers of a nuclear power plant are responsible for initiating, preparing, and controlling their relevant plant operating procedures consistent with their responsibilities to ensure that the work is properly performed in accordance with the latest applicable documents. When newly prepared or revised, the procedures will be reviewed by the Station Operation Review Committee (SORC) of the nuclear power plant (NPP), and then approved by the Plant General Manager.

As required by the Article 9 of Regulations on Quality Assurance Criteria for Nuclear Reactor Facilities, any activities that may affect the quality of the plant must have appropriate procedures, instructions, or drawings. These activities, as specified in the Article 3 of this Regulation, include the design, installation, operation, maintenance, inspection, modification, testing, and so forth of the facility and its structures, systems, and components (SSCs).

The SR included in the technical specifications shall be met during the operational modes (i.e., the power operation mode and the startup mode) or other conditions specified for individual limiting conditions for operation, unless otherwise stated in an individual surveillance requirement. The schedule for the surveillance tests of the safety -related systems will be established in accordance with the surveillance requirements including the surveillance intervals requirements as specified in the technical specifications of the NPP.

Regulatory inspections of the NPP during the construction stage or operating period include the resident inspections, regular inspections, expert team inspections, and special inspections as well as the unannounced inspections. The licensee of an operating NPP is required to provide the following reports to the regulatory body (i.e., the AEC) in accordance with the associated time intervals specified in the Article 7 of the Enforcement Rules for the Implementation of the Nuclear Reactor Facilities Regulation Act of 2003:

- Operation report,
- Radiation safety and environmental radiation surveillance report,
- Reportable event report or emergency event report, and
- Records on the radioactive waste production.

In an operating NPP, the SORC is responsible for reviewing all safety-related affairs and making recommendations to the Plant General Manager. As an illustration, the responsibilities of the SORC include the review of:

- All operating procedures and their revisions,
- All proposed tests and examinations which may affect the nuclear safety,
- All proposed modifications of the tech specs,
- All proposed changes or corrections which may affect the nuclear safety systems or components,

- All tech specs violation events,
- All reportable events,
- Plant emergency plan,
- Etc.

In the headquarters of the licensee (the TPC), the Nuclear Safety Committee (NSC) is the highest advisory organization to give the advisory recommendations to the President of the TPC about the major nuclear safety problems. The NSC is responsible for reviewing and auditing the nuclear safety-related management affairs of all nuclear departments and nuclear power plants which belong to the licensee.

Besides, the Department of Nuclear Safety (DNS) and the Department of Nuclear Generation (DONG) of the TPC perform the reviewing jobs of nuclear safety, independent to each other. The DNS is also responsible for auditing the nuclear power plants periodically or non-periodically to ensure the safe and reliable operation of the plant. The inspections done by the DNS include the annual nuclear safety inspections, the project inspections, vendor auditing, administration auditing, and the component inspections in order to check the implementation of the overall quality assurance (QA) program.

According to the Article 9 of the Nuclear Reactor Facilities Regulation Act and the Article 6 of the Enforcement Rules for the Implementation of Nuclear Reactor Facilities Regulation Act, a nuclear power reactor facility shall perform a periodic safety assessment (PSA) at least every 10 years. In the PSA, a nuclear power plant is required to assess its own operating experience as well as the betterment programs to be implemented in a summarized version and to give a summary about the important things which should be noticed in addition to the corrective actions committed during the next 10-year operation.

19.4 Procedures for Responding to Anticipated Operational Occurrences and Accidents

In the FSAR of a nuclear power plant, transients and accidents are analyzed based on the single-failure criterion. However, when considering multiple failure events, the single-failure criterion is considered to be not appropriate for the emergency operating procedures (EOPs). Therefore, the licensee is required to develop procedures to cope with accidents and transients that are caused by initiating events with multiple system or component failures or operator errors. Examples of multiple failure events include:

(1) Multiple tube ruptures in a single steam generator (SG) and/or tube ruptures in more than one steam generator,

(2) Failure of both main and auxiliary feedwater systems,

(3) Failure of high pressure reactor coolant makeup system,

(4) An anticipated transient without scram (ATWS) event following a loss of offsite power (LOOP), a stuck-open power operated relief valve (PORV) or

safety valve (SV), or a loss of main feedwater, and

(5) Operator errors of negligence.

Symptom-oriented EOPs have been developed and implemented in all three operating nuclear power plants of this nation after the Three Mile Island Unit 2 (TMI-2) accidents in 1979. Based on the generic Emergency Procedure Guidelines (EPGs) provided by the reactor vendors, detailed EOPs for each nuclear power plant were developed by the TPC. Differences between the EOP and EPG have been properly documented and justified. The resultant emergency operating procedures shall comply with the requirements of the NUREG-0737, Item I.C.1. To ensure that the proper procedures had been developed, the TPC performed the verification and validation (V&V) of the EOPs. In addition, simulators have also been used to ensure that the EOPs can be properly simulated. Then, the V&V program of the EOP as well as the EOP itself has to be reviewed and approved by the AEC.

19.5 Engineering and Technical Support

The engineering and technical supports for the plant operations are available from various sources for all the time span from the initial testing program period throughout the lifetime of the plant. The TPC has retained various local as well as overseas consultants to provide the technical assistance on subjects related to the plant safety and operation. The principal backup supports for the plant operation are from various TPC Departments, other TPC nuclear power plants, and the Institute of Nuclear Energy Research (INER) in accordance with the following special technical areas:

(1) Nuclear, mechanical, structural, electrical, thermal hydraulic, metallurgy and materials, instrument, and controls engineering supports were provided by the DONG, DNS, Department of Nuclear Engineering (DNE), other TPC's nuclear power plants, and the INER.

(2) Plant chemistry and health physics supports were provided by the DONG, other TPC's nuclear power plants, and the INER.

(3) Fueling and refueling operation supports were provided by the DONG, other TPC's nuclear power plants, and the INER.

(4) Maintenance support was provided by the Department of Maintenance (DOM), DONG, other TPC's nuclear power plants, and the INER.

As an illustration, the DONG regularly supported the TPC's nuclear power plants in the following areas:

- Establishment and/or implementation of the projects for uprating the plant power or performance,
- Collection and provision of technical information, operating experiences, etc,
- Reloading core designs and safety analysis review,
- Long-term fuel management planning,

- Safety evaluation and review of unexpected and/or important events,
- Review of the modifications of TS and/or FSAR, and
- Review of the plant design change requests (DCRs).

On the other hand, in addition to dispatching a quality assurance (QA) team to stay in each NPP, areas regularly supported by the DNS to the NPPs are as follows:

- Projects for implementing the maintenance rule (MR), life extension, etc,
- Establishment of the plant-specific probabilistic risk assessment (PRA) models,
- Safety evaluation and review of unexpected and/or important events,
- Review of the modifications of TS and/or FSAR, and
- Review of the safety analyses of the reloading core operation transients and/or accidents.

The Institute of Nuclear Energy Research conducts the research and development (R&D) programs in the areas of nuclear safety such as the establishment of domestic nuclear safety and regulatory technologies in the fiscal year 2009, which includes the development of the independent verification technology for nuclear safety analysis, the development of the regulatory tools and guidelines for regulations on the nuclear and radiation safety, the establishment of the accreditation platform for the nuclear grade industrial technologies, and so forth. Besides, the INER can also form a technical team or establish a project with the purpose of solving a particular safety issue when requested. This Institute has been established over 40 years and is the sole domestic and specialized nuclear energy R&D institution.

19.6 Incidents Reporting

19.6.1 Regulatory Requirements for Reporting of Incidents

The requirements of reporting the abnormal or emergency events by the licensee timely are stipulated in the Nuclear Reactor Facilities Regulation Act (Article 10), the Enforcement Rules for the Implementation of this Act (Article 7) and the Regulations on Immediate Notification Requirements and Reportable Event Report for Nuclear Reactor Facilities.

According to the technical guidelines specified in “Regulations on Immediate Notification Requirements and Reportable Event Report for Nuclear Reactor Facilities”, the licensee of an operating nuclear power plant shall report to the AEC any abnormal events described in the following conditions within one hour after the discovery of the event:

(1) Violation of the safety limits in the TS,

(2) Any natural disaster or other external condition which poses an actual threat to the operation safety of a nuclear reactor facility (NRF) or significantly retards

the personnel of the NRF to perform their jobs necessary for the safe operation of the nuclear reactor, and

(3) Events having been published in the news media or notified to the relevant organizations and relating to the health and safety of the public or the persons on site.

Events which must be reported within 2 hours after the discovery of them are as follows:

(1) Events possibly resulted in serious degradation of the fission products protection barriers or a nuclear unit operating in a condition not analyzed in the safety analysis report which could degrade the safety of the unit,

(2) Initiation of the power reduction or shutdown as required by the LCO in the plant's technical specifications,

(3) Any event resulted in one of the following conditions that might significantly affect the ability of the facility to deal with the accident:

(a) loss of the capability to evaluate a nuclear accident including the loss of display of more than half of the safety systems parameters or the loss of alarms for more than 15 minutes,

(b) loss of the off-site emergency response capability including a traffic interruption caused by a natural disaster, and

(c) loss of the communication capability related to the nuclear accident emergency response,

(4) An event or condition which causes the automatic or manual actuation of the engineered safety features (ESFs) or the reactor protection systems (RPS),

(5) An event or condition which makes the structures or systems with the following safety functions unable to fulfill their designed functions:

(a) to shutdown the reactor and maintain in a safe shutdown condition,

(b) to remove the residual heat of the reactor,

(c) to control the release of the radioactive materials, and

(d) to mitigate the consequence of the accident,

(6) Conditions listed in Item 1 of the Article 13 of the Ionizing Radiation Protection Act (i.e. having persons exposed to a radiation dose exceeding the standards set in the regulation: “Safety Standards for Protection against Ionizing Radiation” as amended in December 2005), and

(7) Conditions which result in any one of the following events which are related to and may degrade the safety and health of the public or the employees on site:

(a) Fatalities on site or industrial accident resulting in the transfer of person or persons to off-site for medical care,

(b) Removal of radioactive materials or wastes to offsite that violates the regulations including the Ionizing Radiation Protection Act or the Nuclear Materials and Radioactive Waste Management Act,

(c) Person or persons contaminated by radioactive materials and needed to be transported to off-site for medical treatment,

(d) Occurrence of a gigantic noise, smoke, natural hazard or accident onsite or in the neighboring area which may cause the anxiety of the public,

(e) Incident occurred during the handling of the nuclear fuels, radioactive wastes, or components of the reactor internals on site, and

(f) Event which involves the loss, stealing, or damage of the nuclear fuel, radiation sources, or radioactive wastes,

(8) Intrusion or sabotage related to security,

(9) Forced outage or disconnection from the grid of the unit, and

(10) Conditions listed in Articles 19 and 21 of the Operating Regulations Governing Nuclear Safeguards of 2003.

19.6.2 Restart of a Nuclear Power Unit after Scram

As mentioned in the previous section, within two hours of the occurrence of a reactor scram, the TPC must report to the Nuclear Safety Duty Center, which is on behalf of the Department of Nuclear Regulation of the AEC, about the conditions of the plant after the scram and probable causes. If the cause of the scram is unclear or it is with possible safety concerns, the restart of the said nuclear unit will be under rigorous control. The unit will be allowed to restart only if the root cause is identified or a satisfactory safety assessment is completed. The guidelines for a reactor to restart after a scram are given in the Chapter 4 (Articles 17 to 19) of the Regulations on the Restart of Nuclear Reactor Facilities after Operating Outage (as amended in January 2008).

19.6.3 Evaluation of the Abnormal Occurrence and Equipment Malfunctions of the Nuclear Power Plant

If there is an abnormal event occurring in a nuclear power unit that is required to report as specified in the technical specifications, a detailed report of the situation, the corrective actions and the measures to prevent recurrence must be submitted to the AEC within 30 days. The detailed requirements for this report are given in the regulation "Immediate Notification of Abnormal Events for Operating Nuclear Power Reactors (2004)." The AEC will review this report, evaluate the remediation measures of the plant, dispatch inspectors to the plant if necessary, and monitor the corrective actions if taken.

The number of reportable events for each of the TPC's three operating nuclear power plants during the period from 1988 till December 2009 is shown in Table 19.1.

Similar process is applied to the malfunctions of the major equipment. To assure the safe operation of a nuclear power plant, whenever there is a malfunction of any major equipment, the AEC will immediately dispatch its inspectors to the site for on-site inspection, detailed review of the TPC's analysis of the root cause, and enquiring about the further improvements if necessary.

19.7 Programs to Collect and Analyze Operating Experience

19.7.1 Regulatory Information Study and International Operating Experience Collections

With the assistance of the INER, a program has been established by the AEC to regularly collect and analyze foreign countries' plant operating experiences, especially those of the USA, Japan and France, since 1993. This includes the collection of the generic communications from the USNRC, such as regulatory issue summaries, generic letters, bulletins, and information notices as well as the abnormal events from both Japan and France.

On the other hand, the TPC obtains operating experiences from the General Electric Service Information Letter, Westinghouse Technical Bulletins, BWR Owners' Group (BWROG), Westinghouse Owners' Group (WOG), INPO/WANO Networks and NRC bulletin. Feedbacks of the foreign operating experiences by the TPC to learn the lessons consist of critical review of the relevant circumstances, collecting additional relevant information and carrying out the recommendations identified in the outside reports. Additional surveillance, testing and periodic inspections may be enforced by the AEC as a result of the experience feedback.

19.7.2 Establishment of a System for the Feedback of Operating and Maintenance Experiences

To share the important operating and maintenance experiences among different NPPs, the TPC worked out a program, called the Operation Experience (OE) program, which can be applied to all the TPC's nuclear installations.

The standard operating procedures (SOP) of a plant have been developed to ensure that the plant operating personnel is kept informed of the pertinent improvement information on the plant operation. In addition, steps have been taken to ensure that this information is continually factored into the training programs. For example, the Maanshan's standard operating procedures (SOP 108) have been developed to comply with the requirements of the operating experience feedback to the plant staff.

19.8 Radioactive Waste

The Nuclear Materials and Radioactive Waste Management Act was enacted on December 25, 2002, which replaced all administrative orders for the radioactive waste management enforced upon licensees in the past decades. This Act sets the regulatory requirements for all licensing and enforcement activities on the treatment and storage of the nuclear materials, nuclear fuels and the radioactive wastes as well as the construction, operation, closure, decommissioning and institutional control of the repository of the radioactive wastes including the spent nuclear fuels. The AEC with its subsidiary agency, the Fuel Cycle and Materials Administration (FCMA), is the regulatory authority for the radioactive waste management in this nation.

19.8.1 Low Level Radioactive Waste

The AEC's low level radioactive waste (LLW) management strategies are to do the best to reduce the waste volume, renovate the waste treatment technology, ensure the safety of the storage and actively promote the final disposal program. Until December 2009, a total of 195,147 drums (55-gallon each) of the LLW are stored in Taiwan. Among them, more than 90 percent of the LLW was generated by the three operating NPPs, while the hospitals, research institutes and the industry accounted for the rest. The Lanyu storage facility, located on an offshore islet Lanyu, provides an interim storage for the solidified LLW since 1982. This facility, designed to store 98,112 drums of the LLW in 23 semi-underground engineered trenches, reached its full capacity in 1996. New storage facilities have been constructed at each nuclear power plant site to accommodate the newly generated LLW.

In May 2006, the "Act for Establishment of Low Level Radioactive Waste Final Disposal Facility" (hereafter referred as the "Site Selection Act"), was enacted and became effective. This Act stipulates the disposal site selection procedures and the associated measures. It designates the Ministry of Economic Affairs (MOEA), which supervises the TPC, as the implementing authority and the TPC as the site selection operator. Field investigation and public acceptance activities are being carried out for the site selection. In August 2008, three potential sites were selected for further study, of which two were later recommended in March 2009 as the recommended candidate sites for local county referendum. Regretfully, in September 2009, one recommended candidate site was declared as a "natural landscape ptotection area", which renders the site not eligible for hosting a disposal facility. Since at least two recommended candidate sites are required, in accordance with the Site Selection Act, for holding local county referendum, the site selection process was restarted over again.

If a site passes the referendum, then it will need to get a further approval from the Executive Yuan to become the final disposal site.

19.8.2 Spent Nuclear Fuel

Till December 2009, a total of 14,770 spent fuel assemblies with the equivalent of 3,033 MTU (metric ton uranium) were discharged from the three operating NPPs in Taiwan. As for the spent fuel management, the on-site interim dry storage is considered as a favorable option in Taiwan before implementing the final disposal. A program to build an independent spent fuel storage installations (ISFSI) employing the dry storage technology at the Chinshan plant site is being implemented. The INER is the main contractor of the Chinshan program. Commissioning of the Chinshan ISFSI was originally planned to be held in 2010. However, due to the local protest, the progress of this ISFSI program has been significantly delayed.

On the other hand, a long-term investigation plan is being carried out by the TPC to select a site with suitable geological formation and characteristics for hosting a final repository of the spent nuclear fuel. Preliminary results submitted by the TPC to the AEC showed that there are some potential host rocks in certain regions of Taiwan worthy of further investigation.

In accordance with the “Nuclear Materials and Radioactive Waste Management Act” of 2002, the TPC submitted its “Spent Nuclear Fuel Final Disposal Program” in 2004, which was approved by the AEC in 2006. Currently, the research and development program on spent fuel disposal is at the stage of conducting a study on “Potential Host Rock Characterization and Evaluation”.

Considering that reprocessing of the spent nuclear fuel not only recovers valuable uranium resource and fissile material but also reduces the amount of high level waste, in parallel with the original policy of permanent storage of spent fuel, TPC is also exploring the feasibility of sending a small portion of spent fuel to an European country with the capability of spent fuel reprocessing. According to the Trilateral Nuclear Safeguards Agreement and AIT-TECRO Agreement of Civil Nuclear Cooperation, an MB-10 form (Material Balance) will be submitted to the American Institute in Taiwan once this pilot project is approved by the responsible agencies of the ROC government.

19.9 Transparency of Nuclear Information

In order to make the information of the nuclear power operation as transparent as possible, the TPC put a “Nuclear Safety Information System” on its public Web site: www.taipower.com.tw. The information shown in this System includes:

- Operation performances of the nuclear power units,
- Introduction to the NPPs,
- Real-time information about the operation status of the NPPs,
- Introduction to the safety of an NPP,
- Safety culture of the TPC,
- Environmental radiation monitoring of the NPPs,
- Radiation protection practices at the NPPs,
- Nuclear back-end management,
- Etc.

Furthermore, to make the public aware of the fact that why the nuclear power is an indispensable source of energy in Taiwan and to provide the public with the most updated information about the status of nuclear power operations, the NPPs and the TPC also tried their best to demonstrate their sincerity and openness by making the following efforts:

- Well informing the public all nuclear events at all times through the release of countrywide newsletters. The local residents neighboring the NPPs are given the first priority to access the most updated nuclear information through media or the special/weekly reports.
- Issuing newsletters and/or bulletins and distributing them to local residents, in which the plant status about the plant construction activities and/or operation,

radwaste disposal, environmental protection, etc. is addressed.

- Providing toll-free service telephones to the public for enquiries.
- Inviting the reporters to visit the plant site during plant outage to understand the practice of the maintenance work.
- Hosting two-way communication meetings with the residents committee or community groups around the sites.
- Sending delegates of the NPPs to participate in the community activities.
- Preparing and releasing informative materials at the requests of legislators, reporters, scholars and experts, environmental protection groups, students, the residents neighboring the NPP, and the general public.

Table 19.1 Number of Reportable Events of the TPC's Nuclear Power Plants

	Chinshan	Kuosheng	Maanshan	Total
1988	59	66	15	140
1989	52	93	40	185
1990	91	62	31	184
1991	56	92	62	210
1992	39	77	23	139
1993	49	59	25	133
1994	37	43	19	99
1995	33	34	12	79
1996	20	31	27	78
1997	11	26	9	46
1998	14	40	14	68
1999	23	22	17	62
2000	4	18	6	28
2001	14	15	17	46
2002	6	8	11	25
2003	7	5	12	24
2004	2	4	2	8
2005	4	1	4	9
2006	4	4	4	12
2007	4	9	1	14
2008	8	4	1	13
2009	4	4	0	8

APPENDIX A　GLOSSARY AND ABBREVIATIONS

ABWR	advanced boiling water reactor
AE	alert event
AE or A/E	architect-engineer
AEC	Atomic Energy Council
AFD	axial flux difference
AFI	area for improvement
AIT	American Institute in Taiwan
ALARA	as low as reasonably achievable
AMT	accident management team
ANI	authorized nuclear inspector
ANSI	American National Standards Institute
AO	assistant operator (a licensed RO)
AOO	anticipated operational occurrences
AOP	abnormal operating procedure
AOT	allowed outage time
ARO	assistant reactor operator (a licensed RO)
ASME	American Society of Mechanical Engineers
ASP	alternate shutdown panel
ASSE	automatic scram on strong earthquake
ASTM	American Society for Testing and Materials
ASTS	automatic seismic trip system
ATWS	anticipated transient without scram
AVR	automatic voltage regulator
BOC	beginning of cycle
BOP	balance of plant
BWR	boiling water reactor
BWROG	BWR Owners' Group
BWRVIP	BWR Vessel and Internals Project
CAMP	Code Applications and Maintenance Program
CAP	corrective action program
CFR	Code of Federal Regulations (of US)
CL	construction license
CNPP	Chinshan Nuclear Power Plant

CNS	Committee of Nuclear Safety, TPC
COL	combined construction and operating license
COLR	Core Operating Limits Report
COMPSIS	Computer-Based Systems Important to Safety
COOPRA	Cooperative PRA Research Program
CP	construction permit
CPD	Cooperative Program on Decommissioning
CRD	control rod drive
CRIEPI	Central Research Institute of Electric Power Industry
CSARP	Cooperative Severe Accident Research Program
CTS	customer technical specifications
DBA	design basis accident
DCR	design change request
DNBM/TPC	Department of Nuclear Backend Management, TPC
DNE/TPC	Department of Nuclear Engineering, TPC
DNR	Department of Nuclear Regulation, AEC
DNS/TPC	Department of Nuclear Safety, TPC
DNT	Department of Nuclear Technology, AEC
DONG/TPC	Department of Nuclear Generation, TPC
DPGM	deputy plant general manager
D/Q	deposition factor
DRPI	digital rod position indicator
DRP	Department of Radiation Protection, AEC
EA	exclusion area
ECCS	emergency core cooling system
ECW	emergency circulating water
ECW	emergency cooling water
ECW	essential chilled water
EDG	emergency diesel generator
EIA	environmental impact assessment
EIS	environmental impact statement
EO	equipment operator
EOC	end of cycle
EOP	emergency operating procedures

EPA	Environmental Protection Administration
EPG	emergency procedure guidelines
EPRI	Electric Power Research Institute (of US)
EPU	extended power uprates
EPZ	emergency planning zone
ERF	emergency response facility
ESF	engineered safety features
ETA	ethanolamine
EY	Executive Yuan
EZ	exclusion zone
FCMA	Fuel Cycle and Materials Administration
FMCRD	fine motion control rod drive
FME	foreign material exclusion
FPB	fission product barrier
FSAR	final safety analysis report
GCB	gas cooled breakers
GDC	general design criteria
GEA	general emergency accident
HEP	human error probabilities
HFE	human factor engineering
HLW	high level waste
HPCI	high pressure coolant injection
HPIC	high pressure ionization chamber
HRA	human reliability analysis
HSI	human-system interface
HWC	hydrogen water chemistry
HX	heat exchanger
I&C	instrumentation and control
IAEA	International Atomic Energy Agency
IASCC	irradiation-assisted stress corrosion cracking
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	inter-granular stress corrosion cracking
ILRT	integrated leak rate test

INER	Institute of Nuclear Energy Research
INPO	Institute of Nuclear Power Operations
INSAG	International Nuclear Safety Advisory Group
IPA	integrated plant assessment
IRA	integrated reliability analysis
IRM	intermediate range monitor
ISAR	integrated safety assessment report
ISFSI	independent spent fuel storage installation
ISI	in-service inspection
IST	in-service test
ITS	improved technical specifications
KNPP	Kuosheng Nuclear Power Plant
LCO	limiting conditions for operation
LCO/SR	limiting conditions for operation / surveillance requirements
LER	Licensee Event Report
LLW	low level waste or low level radioactive waste
LNPP	Lungmen Nuclear Power Plant
LOCA	loss of coolant accident
LOOP	loss of offsite power
LPZ	low population zone
LSSS	limiting safety system settings
LY	Legislative Yuan
MCR	main control room
MIRU	maintenance integrated risk utilities
MMCS	maintenance management computerization system
MMI	man-machine interface
MNPP	Maanshan Nuclear Power Plant
MOEA	Ministry of Economic Affairs
MOFA	Ministry of Foreign Affairs
MOV	motor-operated valve
MR	maintenance rule
MSIV	main steam isolation valves
MSL	main steam lines

MSR	moisture separator reheater
MTU	metric-ton uranium
MUR	measurement uncertainty recapture
NCD	non-conformance disposition
NEA	Nuclear Energy Agency of the Organization for Economic Cooperation and Development
NERU	Nuclear Emergency Response Unit, TPC
NESC	Nuclear Emergency Support Center
NEST	Nuclear Energy Society, Taipei
NM	near-miss
NNERC	National Nuclear Emergency Response Center
NORM	naturally-occurring radioactive material
NPP	nuclear power plant
NPS	nuclear power station
NSC	Nuclear Safety Committee, TPC
NSDC	Nuclear Safety Duty Center
NSSS	nuclear steam supply system
NTHU	National Tsing Hua University
NT$	new Taiwan dollar
NUPIC	Nuclear Procurement Issues Committee
NuSTA	Nuclear Science and Technology Association
O&M	operation and maintenance
OBE	operating basis earthquake
OE	operating experience
OECD	Organization for Economic Cooperation and Development
OEM	original equipment manufacturer
OJT	on-the-job training
PAG	protective action guides
PBNC	Pacific Basin Nuclear Conference
PC	personal computer
PCIS	primary containment isolation system
PCN	procedure change notice
PDA	personal digital assistant
PGM	plant general manager

PI	performance indicator
PM	preventive maintenance
PNC	Pacific Nuclear Council
PORV	power operated relief valve
PRA	probabilistic risk assessment
PSAR	preliminary safety analysis report
PSA	periodic safety assessment
PWR	pressurized water reactor
QA	quality assurance
QC	quality control
R&D	research and development
RCA	root cause analysis
RCCA	rod cluster control assembly
RCIC	reactor core isolation cooling (system)
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RER	reportable event reports
RHR	residual heat removal
RMC	Radiation Monitoring Center (of AEC)
RMDAC	Radiation Monitoring and Dose Assessment Center
RNERC	Regional Nuclear Emergency Response Center
RO	reactor operator
ROC	Republic of China
ROP	reactor oversight process
RPS	reactor protection system
RPV	reactor pressure vessel
RWCU	reactor water clean-up system
SAEA	site area emergency accident
SAMG	severe accident management guidelines
SAR	safety analysis report
SBO	station blackout
SC	safety culture
SCC	stress corrosion cracking
SDP	significance determination process

SER	safety evaluation report
SER	significant event reports (of WANO)
SH	section head
SL	safety limits
SL	shift leader
SM	section manager
SM	shift manager
SM	shutdown margin
SOER	significant operating experience reports (of WANO)
SOP	standard operating procedures
SORC	station operation review committee, TPC
SPDS	safety parameter display system
SPU	stretch power uprates
SR	surveillance requirements
SRM	source range monitor
SRO	senior reactor operator
SSC	structure, system and component
SSI	soil-structure interaction
ST	surveillance test
STS	standard technical specifications
SV	safety valve
TAF	Taiwan Accreditation Foundation
Taipower	Taiwan Power Company
TBM	tool box meeting
tech. spec.	technical specifications
TECRO	Taipei Economic and Cultural Representative Office in the US
TENORM	technologically enhanced naturally-occurring radioactive material
TEPCO	Tokyo Electric Power Company
TLAA	time-limited aging analysis
TLD	thermo-luminescent dosimeter
TPC	Taiwan Power Company
TPRI	Taiwan Power Research Institute, TPC
TRIM	TPC risk integrated monitor

TRM	technical requirement manual
TS	technical specifications
TSC	Technical Support Center
USNRC	United States Nuclear Regulatory Commission
V&V	verification and validation
W	Westinghouse Electric Corporate
WANO	World Association of Nuclear Operators
WANO-TC	World Association of Nuclear Operators – Tokyo Center
WEC	Westinghouse Electric Company
WOG	Westinghouse Owners' Group
WRNMS	wide range neutron monitor system
X/Q	relative atmospheric dispersion factor

APPENDIX B：CONTRIBUTORS TO THE ROC's NATIONAL REPORT

The Atomic Energy Council and the Institute of Nuclear Energy Research prepared this report in consultation with:

Chinshan Nuclear Power Plant,
Department of Nuclear Backend Management, TPC,
Department of Nuclear Generation, TPC,
Department of Nuclear Engineering, TPC,
Department of Nuclear Regulation, AEC,
Department of Nuclear Safety, TPC,
Department of Nuclear Technology, AEC,
Department of Planning, AEC,
Department of Radiation Protection, AEC,
Fuel Cycle and Materials Administration, AEC,
Kuosheng Nuclear Power Plant,
Lungmen Nuclear Power Plant,
Maanshan Nuclear Power Plant,
Nuclear Emergency Response Unit, TPC,
Radiation Monitoring Center, AEC, and
Taiwan Power Company.

Main contributors to this report were:

Chou, Yuan-Ching (周源卿),
Yang, Chao-Yie (楊昭義), and
Yin, Hsueh-Li (尹學禮).

Other contributors included mainly:

Chang, Chin-Jang (張欽章)
Chang, Shin (張欣),
Chow, Ting (周鼎),
Liao, Lih-Yih (廖俐毅),
and many other technical and regulatory experts from the AEC and INER.

The project manager was Chao-Yie Yang and the coordinator for the whole project was Ting Chow.

Assistance from Ms. Mei-Guet Chien (簡美桂) of the INER in the preparation of this report is acknowledged.

ANNEX 1

MAJOR TECHNICAL CHARACTERISTICS OF NPPS

IN

TAIWAN

I. Major Technical Characteristics of the Chinshan NPP

Items	Units 1 & 2
THERMAL-HYDRAULIC DESIGN	
Rated Thermal Power, MWt	1,804
Design Power (ECCS design basis = 105% rated), MWt	1,864
Rated Electrical Power, MWe	636
Reactor Coolant System: System Pressure, nominal in steam dome, psia Core Coolant Flow Rate, lb/hr	 1,020 53×10^6
Steam Flow Rate, lb/hr	7.693×10^6
Feedwater Flow Rate, lb/hr	7.670×10^6
Feedwater Temperature, °F	420
Heat Transfer: Maximum Thermal Output, kW/ft Average Thermal Output, kW/ft Maximum Heat Flux, Btu/hr-ft^2 Average Heat Flux, Btu/hr-ft^2	13.4 4.04 441,400 133,200
Minimum Critical Power Ratio (MCPR)	≥ 1.32
NUCLEAR DESIGN	
Average Feed Enrichment (First Core), w/o	1.90
H_2O/UO_2 Volume Ratio (cold)	2.59
CORE MECHANICAL DESIGN	
Equivalent Core Diameter, in.	136.8
Core Height (Active Fuel Length), in.	Full length rod: 149.45 Partial length fuel rod: 90
Fuel Assembly (FA)(Initial core): Number of FAs in the Core Fuel Rod Array Overall FA Length, in.	 408 10 x 10 176.39
Fuel Rod: Number of Fuel Rods per FA Outside Diameter, in. Diametrical Gap (Pellet to cladding), in. Cladding Thickness, in. Cladding Material	 91 0.3957 0.0067 0.02385 Zircaloy-2

Fuel Pellet:	
Material	Uranium Dioxide (UO_2)
Density, % of theoretical	95.85
Diameter, in.	0.3413
Length, in.	0.41
Fuel Channel:	
Material	Zircaloy-4
Overall Length, in.	166.91
Thickness, in.	0.08
Cross-sectional Dimension, inch x inch	5.438 x 5.438
Control Rod Assembly (CRA):	
Shape	Cruciform
Neutron Absorber Material	B_4C & Hf
Cladding Material	SS
Total Number of CRAs in the Core	97
CONTAINMENT	
Type	Mark I, Steel Drywell and Pressure Suppression Pool
Leakage Rate, % vol/day	0.5
Drywell:	
Construction	Light Bulb Shape, Steel Vessel
Internal Design Temperature, °F	340
Maximum Internal Pressure, psig	56
Total Free (air) Volume, ft^3	130,000
Suppression Pool:	
Construction	Torus, Steel Vessel
Internal Design Temperature, °F	340
Internal Design Pressure, psig	56
Water Volume, ft^3	78,000
Total Free (air) Volume, ft^3	87,200

II. Major Technical Characteristics of the Kuosheng NPP

Items	Units 1 & 2
THERMAL-HYDRAULIC DESIGN	
Rated Thermal Power, MWt	2,943
Design Power (ECCS design basis = 105% rated), MWt	3,039
Rated Electrical Power, MWe	1,029.7 (unit 1)* 997.5 (unit 2)
Reactor Coolant System: System Pressure, nominal in steam dome, psia Core Coolant Flow Rate, lb/hr	 1,040 84.5×10^6
Steam Flow Rate, lb/hr	12.734×10^6
Feedwater Flow Rate, lb/hr	12.831×10^6
Feedwater Temperature, °F	424.14
Heat Transfer: Maximum Thermal Output, kW/ft Average Thermal Output, kW/ft Maximum Heat Flux, Btu/hr-ft^2 Average Heat Flux, Btu/hr-ft^2	13.4 4.3 0.50×10^6 0.16×10^6
Minimum Critical Power Ratio (MCPR)	1.20
NUCLEAR DESIGN	
Average Feed Enrichment (first core), w/o	1.88
H_2O/UO_2 Volume Ratio (cold)	2.70
CORE MECHANICAL DESIGN	
Equivalent Core Diameter, in.	160.2
Core Height (Active Fuel Length), in.	150
Fuel Assembly (FA)(first core): Number of FAs in the Core Fuel Rod Array Overall FA Length, in.	 624 10 x 10 176
Fuel Rod: Number of Fuel Rods per FA Outside Diameter, in. Diametrical Gap (pellet to cladding), in. Cladding Thickness, in. Cladding Material	 91 0.396 0.0067 0.0238 Zircaloy-2

Fuel Pellet:	
Material	Uranium Dioxide (UO_2)
Density, % of theoretical	95.85
Diameter, in.	0.3413
Length, in.	0.413
Fuel Channel:	
Material	Zircaloy-4 or Zircaloy-2
Overall Length, in.	166.9
Thickness, in.	0.067/0.114
Cross-sectional Dimension, inch x inch	5.278 x 5.278
Control Rod Assembly (CRA):	
Shape	Cruciform
Neutron Absorber Material	B_4C and Hf
Cladding Material	SS
Total Number of CRAs in the Core	145
CONTAINMENT	
Type	Mark III, Reinforced Concrete Containment with Pressure Suppression and Reactor Building Enclosing Drywell and Suppression Pool
Leakage Rate, % vol/day	0.45
Reactor Building Construction	Reinforced Concrete Cylindrical Structure with Hemispherical Head and Steel Liner
Internal Design Temperature, °F	200
Design Pressure, psig	15
Total Free (air) Volume, ft^3	1.43×10^6
Drywell:	
Construction	Reinforced Concrete Unlined; Basically Cylindrical; Steel Head
Internal Design Temperature, °F	330
Design Pressure, psig	+27.5, -21.7
Total Free (air) Volume, ft^3	238,000

Suppression Pool:	
Construction	Reinforced Concrete, Steel Lined and Cylindrical
Internal Design Temperature, °F	200
Design Pressure, psig	15
Water Volume (at high water level), ft^3	113,950

*Unit 1 low pressure turbine rotor had been replaced in 2006.

III. Major Technical Characteristics of the Maanshan NPP

Items	Units 1 & 2
THERMAL-HYDRAULIC DESIGN	
Reactor Core Thermal Power, MWt	2,822
NSSS Thermal Power, MWt	2,834
Rated Electrical Power, MWe	960
Reactor Coolant System: System Pressure, nominal design, psia System Pressure, minimum steady state, psia System Pressure, nominal operating, psia Coolant Inlet Temperature, nominal, °F Reactor Pressure Vessel Inlet Temperature, °F Reactor Pressure Vessel Outlet Temperature, °F	 2,280 2,220 2,250 554.2 554.2 621.4
Total Reactor Coolant Flow Rate, gpm	277,800
Steam Generator: Feedwater Temperature, °F SG Steam Outlet Temperature, °F Steam Pressure, psia Total Steam Flow Rate, lb/hr	 442.6 537.2 979 12.55×10^6
Heat Transfer: Maximum Thermal Output, kW/ft Average Thermal Output, kW/ft Maximum Heat Flux, Btu/hr-ft^2 Average Heat Flux, Btu/hr-ft^2	13.4 5.53 505,089 201,130
Minimum DNBR (for design): Typical Flow Channel Transients Thimble Flow Channel	 ≥ 1.23 ≥ 1.22
NUCLEAR DESIGN	
Feed Enrichment (First Core), w/o: Region 1 Region 2 Region 3 Reload	 1.6 2.4 3.1 4.68 & 4.95
H_2O/U Molecular Ratio (cold)	2.73
CORE MECHANICAL DESIGN	
Equivalent Core Diameter, in.	119.7
Core Height (Active Fuel Length), in.	144

Core Barrel: Inside Diameter, in. Outside Diameter, in.	133.85 137.875
Thermal Shield	Neutron Pad Design
Fuel Assembly (FA): Number of FAs in the Core Fuel Rod Array Number of Fuel Rods per FA	 157 17 x 17 264
Fuel Rod: Outside Diameter, in. Diametrical Gap (Pellet to cladding), in. Cladding Thickness, in. Cladding Material	 0.360 0.0062 0.0225 zirlo
Fuel Pellet: Material Density, % of theoretical Diameter, in. Length, in.	 Uranium Dioxide (UO_2) 95 0.3088 0.507
Control Rod Assembly (CRA): Shape Neutron Absorber Material Cladding Material Cladding Thickness, in. Total Number of CRAs in the Core Number of Absorber Rods per CRA	 Cluster-type Ag-In-Cd Type 304 SS 0.0185 52 24
CONTAINMENT	
Type	Steel-Lined Pre-Stressed Post-Tensioned Concrete Cylinder, Hemispherical Dome Roof
Leakage Rate, % vol/day	0.1 (24 hr), 0.05 (after)
Internal Design Pressure, psig	60
Total Free (air) Volume, ft^3	2.0×10^6
Diameter, ft	130
Height, ft	195

IV. Major Technical Characteristics of the Lungmen NPP

Items	Units 1 & 2
THERMAL-HYDRAULIC DESIGN	
Rated Thermal Power, MWt	3,926
Design Power (ECCS design basis = 105% rated), MWt	4,005
Rated Electrical Power, MWe	1,350
Reactor Coolant System:	
System Pressure, nominal in steam dome, psia	1,040
Core Coolant Flow Rate, lb/hr	115.1×10^6
Steam Flow Rate (at 420 °F, FW temp.), lb/hr	16.843×10^6
Feedwater Flow Rate, lb/hr	16.807×10^6
Feedwater Temperature, °F	420
Heat Transfer: Maximum Thermal Output, kW/ft	13.4
Average Thermal Output, kW/ft	4.2
Maximum Heat Flux, Btu/hr-ft^2	432,296
Average Heat Flux, Btu/hr-ft^2	135,496
Minimum Critical Power Ratio (MCPR)	1.35
NUCLEAR DESIGN	
Average Feed Enrichment (first core), w/o	1.79
H_2O/UO_2 Volume Ratio (cold)	3.04
CORE MECHANICAL DESIGN	
Equivalent Core Diameter, in.	203.3
Core Height (Active Fuel Length), in.	150
Fuel Assembly (FA)(Initial core):	
Number of FAs in the Core	872
Fuel Rod Array	10 x 10
Overall FA Length, in.	176
Fuel Rod:	
Number of Fuel Rods per FA	92
Outside Diameter, in.	0.404
Diametrical Gap (pellet to cladding), in.	0.007
Cladding Thickness, in.	0.026
Cladding Material	Zircaloy-2

Fuel Pellet: Material Density, % of theoretical Diameter, in. Length, in.	 Uranium Dioxide (UO_2) 97 0.345 0.35
Fuel Channel: Material Overall Length, in. Thickness: Corner, in. Wall, in. Cross-Sectional Dimension, inch x inch	 Zircaloy-2 176 0.120 0.075 5.278 x 5.278
Control Rod Assembly (CRA): Shape Neutron Absorber Material Cladding Material Total Number of CRAs in the Core	 Cruciform B_4C and Hafnium SS 205
CONTAINMENT	
Primary Containment:	
Type	Over-and-Under Pressure Suppression
Construction	Reinforced Concrete with Steel Liner; Steel Structure
Drywell	Concrete Cylinder
Pressure Suppression Chamber	Concrete Cylinder
Containment Internal Design Pressure, psig	45
Drywell Internal Design Pressure, psig	45
Drywell Free (air) Volume, ft^3	259,600
Pressure Suppression Chamber Free (air) Volume (at high water level), ft^3	210,000
Pressure Suppression Chamber Water Volume (at low water level), ft^3	126,400
Drywell Design Temperature, °F	340
Pressure Suppression Chamber Design Pressure, psig	30.5
Leakage Rate, % free volume/day	0.5
Secondary Containment:	
Type	Controlled Leakage

Construction: Lower Levels	Reinforced Concrete
Upper Levels	Reinforced Concrete
Roof	Reinforced Concrete

ANNEX 2

BILATERIAL PEER REVIEW BETWEEN US NRC AND ROC AEC

Review Questions and Comments for
「The United States of America Fifth National Report for the Convention on Nuclear Safety」

INTRODUCTION

1. **On page 22, it is mentioned in the Cyber Security that the reactor system is protected from cyber attacks up to and including the design-basis threat. Is there a possibility that a cyber attack could go beyond design-basis? If it is possible, how will NRC deal with this issue?**

   Answer: The update of 10CFR 73.54 extends the design-basis threat concept such that licensees are required to provide risk-based protection of industrial and digital control systems against likely cyber threats and industrial control system vulnerabilities that might be leveraged by an attacker to compromise the safe operation and control of the plant environment. The NRC has charged its licensee community to align or map the National Institute of Standards and Technologies, Special Publication 800-53, Revision 3 controls, which comprise a strong, risk-based approach to component acquisition, security control selection, security system monitoring and incident response and contingency planning as necessary to ensure that our plants possess and operate, a safe and robust digital control system within the operating environment. We do not prescribe a set approach, but work with the licensee community to understand their approach and to objectively inspect and assess the strength of the approach against the requirements in 10CFR 73.54.

2. **On page 20, it is mentioned that "Survey of Safety and Regulatory Issues". What's your schedule in resolving specific issues such as degradation of buried piping, degradation of neutron-absorber materials in spent fuel pool and containment pressure credit for ECCS pump net positive suction head?**

   Answer: (DCI) The NRC's response to degradation of buried piping is concentrated in two areas. The first of these areas concerns the management of buried piping at all operating U.S. reactors during their currently licensed period of operation. The NRC has determined that, at this time, it is unnecessary to change its regulatory approach to buried piping. However, the nuclear industry has developed a voluntary initiative to address the degradation of buried (in contact with soil) and underground (in trenches or vaults) piping. The industry has committed to fully implement this program by December 31, 2014. The NRC has developed a "Buried Piping Action Plan" to provide for long term monitoring of the status of buried piping. As part of the Buried Piping Action Plan, the NRC has developed guidance for NRC inspectors to monitor the effectiveness of the industry's implementation of their voluntary initiative. Inspections conducted under this guidance begin in September 2011 and continue into 2015.

The second area is in regards the management of buried piping during periods of extended operation. The aging management program (AMP) related to buried and underground piping (NUREG-1801, "Generic Aging Lessons Learned Report," Revision 2, December 2010, AMP XI.M41) has recently been revised to better address buried piping degradation. This AMP is being applied to plants which are currently seeking renewed licenses. This AMP will be applied to plants which seek renewed licenses in the future. This AMP will not be directly applied to plants which already hold renewed licenses.

Regarding neutron absorber materials, the NRC has a research initiative underway to better understand degradation of neutron absorbing materials in the spent fuel pool by compiling a literature knowledge base, performing confirmatory research on the surveillance methodologies, and performing confirmatory research on appropriate surveillance intervals. The NRC's goal is to have all of these activities completed by the end of 2012. After completion of this research, the NRC will evaluate whether other regulatory or generic actions need to be taken. If interim steps identify safety issues prior to completion of this effort, appropriate regulatory actions will be taken.

(DSS) Regarding containment pressure credit for ECCS pump net positive suction head, the staff proposed options to the NRC Commission in a paper (SECY 11-0014, "Use of Containment Accident Pressure in Analyzing Emergency Core Cooling System and Containment Heat Removal System Pump Performance in Postulated Accidents," dated January 31, 2011; ML102590196). The Commission provided direction to the staff in SRM-SECY-11-0014, dated March 15, 2011 (ML110740254). This effectively resolves the major issues on the use of containment accident pressure (CAP). The staff also provided guidance to the industry in a letter to the Boiling Water Reactor Owners' Group dated March 1, 2010 (ML100740579). The staff is in the process of revising this guidance. However, after direction from the Commission, the staff has commenced all reviews that include the use of CAP.

ARTICLE 6：EXISTING NUCLEAR INSTALLATIONS

1. **In Section 6.3.10, please elaborate on the Reactor Safety Research program: what are the recent research topics chosen and what is the budget associated with them?**

   Answer: The NRC's Reactor Safety Research program is run by the Office of Nuclear Regulatory Research (RES). Details on the structure of RES, the general processes followed to identify and institute research programs and the funding of research is included in a document known as NUREG-1925 (rev 1) – "Research Activities FY 2010-FY 2011," dated December 2010. It is available on the NRC's website at: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1925/r1/.

2. **In Section 6.3.5, the centralized clearinghouse of NRC seems to integrate all the works relating to OE and also establish internal operating program. Is there a document that is available to foreign members for understanding operating mechanism within the clearinghouse and its manpower deployment? Is there any interaction between the clearinghouse and INPO?**

   Answer: Management Directive (MD) 8.7, *Reactor Operating Experience Program*, lists the duties and responsibilities of various NRC offices with regards to their interaction with NRC's Operating Experience program. This document is publicly available under ADAMS Accession Number ML062970023. More detailed instructions on the screening and evaluation process are not publicly available. The clearinghouse group and INPO communicate on a bi-weekly basis to exchange information on issues of interest and to discuss any items of concern to either organization. INPO shares its Event Reports with NRC for their internal use as outlined in the INPO/NRC Memorandum of Agreement. In addition an annual meeting between members of the operating experience groups at NRC and at INPO is held to discuss program updates and ongoing work.

3. **In Section 6.3.3, it is stated that no statistically significant adverse industry**

trends in the Industry Trend Program FY 2009. Is there any example which did show the statistically significant adverse industry trends in the past? Is there a specific regulatory activity for the statistically significant adverse industry trends in the NRC?

Answer: Based on the results of the Industry Trends Program, the NRC staff has not identified any statistically significant adverse trends in industry safety performance since initial implementation through the end of FY 2010.

The planned NRC response to a statistically significant adverse trend is described in NRC Inspection Manual Chapter 0313, "Industry Trends Program," Sections 06.04 and 06.05, as follows:

<u>06.04</u> <u>Analyses of Issues.</u>

Once an adverse trend is identified, the staff conducts an initial analysis of information readily available in the databases used to compile the indicator data to determine whether the trend is unduly influenced by a small number of outliers and to identify any contributing factors. If the trend is the result of outliers, then it is not considered a trend requiring generic actions, and the agency will consider any appropriate plant-specific actions using the ROP. For example, the affected plants unduly influencing the adverse trend may have already exceeded plant-level thresholds under the ROP, and the NRC regional offices would conduct supplemental inspections at these plants to ensure the appropriate corrective actions have been taken. If the plants did not exceed any thresholds, the NRC would not take regulatory actions beyond the ROP, however, the NRC may gather additional information regarding the issue within the scope of the ROP using risk-informed baseline inspections. The results of these inspections would be examined to determine if a generic issue exists, requiring additional NRC review or generic inspections.

If no outliers are identified, the staff conducts a broader review to assess whether larger groups of facilities are contributing to the decline and to assess any contributing factors and causes. For example, the data review is expanded to include a review of various plant comparison groups, contributing factors such as the operational cycle stage of the facilities (shutdown, at-power, startup from refueling, etc.), and the apparent causes for the data (equipment failures, procedure problems, etc.). The staff will also conduct a more detailed review of applicable Licensee Event Reports. Should a group of plants be identified, the staff will examine the results of previously conducted inspections at these plants, including any root causes and the extent of the conditions.

Once this information is reviewed, the staff assesses the safety significance of the underlying issues. The staff is mindful that trends in individual indicators must be considered in the larger context of their overall risk significance. For example, a hypothetical increase in automatic scrams from 0.4 to 0.7 per plant per year over several years may be a statistically significant trend in an adverse direction. However, it may not represent a significant increase in overall risk since the contribution of a small number of scrams is relatively low, and it is possible that overall risk may actually have declined if there were reductions in the frequency of more risk-significant initiating events or the reliability and availability of safety systems had improved. Depending on the issues, the staff may perform an additional evaluation using the most current risk analysis tools or an evaluation by the ASP Program.

06.05 Agency Response.

Should a statistically significant adverse trend in safety performance be identified or an indicator exceed a prediction limit, the staff will determine the appropriate response using the processes described above and the NRC's established processes for addressing and communicating generic issues. The generic issue process is described in SECY-99-143, “Revisions to Generic Communications Program.”

In general, the issues will be assigned to the appropriate branch of NRR for initial review. The branch will engage NRC senior management and initiate early interaction with the nuclear power industry. Depending on the issue, the process could include requesting industry groups such as the Nuclear Energy Institute (NEI) or various owners groups to provide utility information. As discussed in SECY-00-0116, “Industry Initiatives in the Regulatory Process,” dated June 28, 2000, industry initiatives, such as the formation of specialized working groups to address technical issues, may be used instead of, or to complement, regulatory actions. This can benefit both the NRC and the industry by identifying mutually satisfactory resolution approaches and reducing resource burdens.

Depending on the issues, the NRC may consider generic safety inspections at plants. In addition, the issues underlying the adverse trend may also be addressed as part of the generic safety issue process by RES. The NRC may consider additional regulatory actions as appropriate, such as issuing generic correspondence to disseminate or gather information, or conducting special inspections for generic issues. The process also includes consideration of whether any actions proposed by the NRC to address the issues constitute a backfit.

4. **In Section 6.3.3, it is mentioned that "implemented the Baseline Risk Index for Initiating Events (BRIIE), a new indicator". Does it mean that only the indicators of internal events will be evaluated for the BRIIE? What is your position regarding the external initiating events?**

   Answer: The risk-significant initiating event types included in BRIIE consists of 10 initiating event categories applicable to pressurized-water reactors (PWRs) and 9 applicable to boiling-water reactors (BWRs) as listed below.

   Risk-significant initiating event categories covered by the BRIIE

Pressurized Water Reactors (PWRs)	Boiling Water Reactors (BWRs)
1. Loss of offsite power (LOOP)	1. Loss of offsite power (LOOP)
2. Loss of vital AC bus (LOAC)	2. Loss of vital AC bus (LOAC)
3. Loss of vital DC bus (LODC)	3. Loss of vital DC bus (LODC)
4. Loss of main feedwater (LOMFW)	4. Loss of main feedwater (LOMFW)
5. Very small loss of coolant accident (VSLOCA)	5. Very small loss of coolant accident (VSLOCA)
6. PWR general transient (TRAN)	6. BWR general transient (TRAN)
7. PWR loss of condenser heat sink (LOCHS)	7. BWR loss of condenser heat sink (LOCHS)
8. PWR stuck open safety/relief valve (SORV)	8. BWR stuck open safety/relief valve (SORV)
9. PWR loss of instrument air (LOIA)	9. BWR loss of instrument air (LOIA)
10. Steam generator tube rupture (SGTR)	

   In general, these risk-significant initiating event types cover a majority of the internal event core damage risk (excluding internal flooding) from the operating commercial nuclear power plants in the United States. Also, these initiating events do not overlap.

The technical basis for the current BRIIE indicators is detailed in NUREG/CR-6932, "Baseline Risk Index for Initiating Events (BRIIE)," dated June 2007. As stated in that document, "A review of SPAR model CDF results for the 103 operating U.S. commercial nuclear power plants indicates that the BRIIE initiating events cover approximately 60% of the total internal event CDF from these models. Other initiating events within these models (covering the remaining 40% of CDF) that are not included within the BRIIE include such events as loss of service water, loss of component cooling water, medium and large LOCAs, and interfacing system LOCAs. These events are rare and generally would not be expected over the lifetimes of the plants. Therefore, such events are not monitorable on a yearly basis."

At the present time we have no plans to extend the BRIIE concept to other events.

5. **In the paragraph of Section 6.3.4, it is said that the last significant precursor was identified in FY 2002 (i.e., multiple degraded conditions at Davis-Besse). Was there a specific regulatory activity after the finding of significant precursor in the NRC?**

   Answer: The regulatory activities related to Davis-Besse were carried out long before the Accident Sequence Precursor (ASP) analysis was completed, and there was no additional regulatory activity after the ASP analysis. Since the problem was obviously serious, the regulatory activity started as soon as the problem was discovered in February 2002. As soon as the significant degradation of the reactor vessel head was identified, the NRC issued a Confirmatory Action Letter to the licensee. In April 2002, the NRC performed an Augmented Inspection Team (AIT) inspection, and on April 29, 2002, in response to the findings of that inspection, the NRC implemented Inspection Manual Chapter (IMC) 0350 at Davis-Besse. The vessel head was replaced during this extended outage, along with numerous other regulatory and technical activities. In March 2004, the NRC lifted its restart restriction and the plant was restarted.

   The ASP analysis was substantially delayed since the parameters were derived from probabilistic structural mechanical analyses, which required the results of numerous laboratory tests and an extensive modeling effort. The ASP analysis process includes internal reviews and an opportunity for the licensee to review and provide comments. The final ASP analysis was not issued until March 2005. While this chain of events is not typical for an event at a US nuclear power plant, it is also not typical for an ASP analysis to lead to immediate regulatory activities. The NRC normally waits until after the post-event inspection activities are complete to do the ASP analysis, and most regulatory activities are driven by the inspections.

6. **In the paragraph of Section 6.3.8, it is stated that NRC is reviewing two volunteered pilot plants to transitioning to 10 CFR 50.48(c). However, the current Fire PRA models were not robust enough to support the type of risk-informed decision-making as mentioned in the ML110210990 titled "Roadmap for attaining Realism in Fire PRAs" (NEI, December 2010). Will NRC require the pilot plants to update their Fire PRA models after the improvement of the model completed two or three years later?**

   Answer: There is no requirement to upgrade fire PRA methods to reflect new methods; this is voluntary on the part of the licensee. Guidance on updating and upgrading licensee PRAs is provided in the NRC-endorsed PRA standard ASME/ANS-RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications." It should be noted that for the second pilot plant application, issues were identified with the licensee's PRA related to their internal events PRA, as well as their fire PRA, that resulted in the NRC restricting its use by the licensee until an industry peer review was conducted and identified issues were resolved, which will require another application from the licensee to request approval for use of the revised, peer-reviewed, PRA.

   In addition, the cited paper was developed by industry representatives and the NRC disagrees with the conclusions made in this paper. The review of the referenced two volunteered pilot plants was completed in 2010. The completion of these plant reviews, in addition to the current submittals requesting to implement the performance-based, risk-informed fire protection program (10 CFR 50.48(c), also referred to as NFPA 805 rule), indicates that there is the capability to use current Fire PRA models for this type application.

It has also been recognized that fire PRA methods are being further refined and will evolve as they are used. The NRC Office of Nuclear Regulatory Research continues to work with the Electric Power Research Institute (EPRI) to develop essential documents that support the successful implementation of the NFPA 805 rule, including NUREG/CR-6850, "*Fire PRA Methodology for Nuclear Power Facilities*," September 2005 and its supplement, which was issued in 2010. This document provides credible Fire PRA methods to help ensure the application of risk is appropriate for the NRC's fire protection regulation. Collaboration with EPRI brought together nationally recognized technical experts to document the state-of-the-art in Fire PRA methodology. To ensure new or refined methods are acceptable, the industry has established a Fire PRA Methods Task Force and the NRC interacts with this industry task force.

7. **In the paragraph of Section 6.3.8, it is stated that the NRC staff is reviewing 11 such requests with related to exemptions to the regulations or changes to their approved fire protection program for the use of operator manual actions. Please provide some information which requested the exemptions or changes.**
Answer: NRC generic communication Regulatory Issue Summary (RIS) 06-010, "Regulatory Expectations with Appendix R Paragraph III.G.2 Operator Manual Actions," dated June 30, 2006, reiterated regulatory expectations for the use of operator manual actions to assure safe shutdown capability. Operation of redundant trains of equipment may not rely on operator manual actions from outside of the control room. Enforcement Guidance Memorandum (EGM) 07-004, "Enforcement Discretion For Post-Fire Manual Actions Used as Compensatory Measures for Fire Induced Circuit Failures," dated June 30, 2007, provided enforcement discretion for plants to submit exemptions or licensing actions to request permission to perform operator manual actions in lieu of the protection described in 10 CFR 50, Appendix R, Section III.G.2a, b, and c.
Eleven (11) licensing actions were submitted to the NRC for review. Below are the plant names and accession numbers for the NRC's approval of the submittals. The NRC approval document provides references to the submittals and the related correspondence.
Pilgrim (withdrawn) ML101170118 (1 action)

Three Mile Island ML101310113 (1 action)

Oyster Creek ML110700451 and ML110700267 (2 actions)

FitzPatrick ML100340670 (1 action)

Wolf Creek ML103090262 (1 action)

Peach Bottom ML102430566 (1 action)

South Texas ML100780075 (2 actions in one document)

Indian Point Units 2 and 3 have manual actions exemptions currently under review. (2 actions)

8. **Section 6.1 describes five strategic outcomes. How is "significant radiation exposures" defined? How is "significant adverse environment impacts" defined? Is there a definite value to become significant?**

How is "significant radiation exposures" defined?

Answer: Licensees report overexposures through the Sequence Coding and Search System (SCSS) Licensing Event Report (LER) database, maintained at the Oak Ridge National Laboratory, which receives all LERs and codes them into a searchable database. The SCSS database is used to identify those LERs that report overexposures. NRC resident inspectors stationed at each nuclear power plant provide a high degree of assurance that all events meeting reporting criteria are reported to the NRC. In addition, the NRC conducts inspections if there is any indication that an exposure exceeded, or could have exceeded, a regulatory limit. Finally, areas of the facility that may be subject to radiation contamination have monitors that record radiation levels. These monitors would immediately reveal any instances in which high levels of radiation exposure occurred.

How is "significant adverse environment impacts" defined?

Answer: As with worker overexposures, licensees report environmental releases of radioactive materials that are in excess of regulations or license conditions through the SCSS LER database maintained at the Oak Ridge National Laboratory. The SCSS database will be utilized to identify those LERs reporting releases and the number of reported releases is then applied to this measure. The NRC also conducts periodic inspections of licensees to ensure that they properly monitor and control releases to the environment through effluent pathways. In addition, onsite monitors would record any instances in which the plant releases radiation into the environment. If the inspections or the monitors reveal any indication that an accident or inadvertent release has occurred, the NRC conducts follow-up inspections.

Is there a definite value to become significant?

Answer: No. There is no definitive value to become significant. The NRC will evaluate significance if/when a regulatory limit has been exceeded.

9. **In Section 6.2, "NRC has full authority to take whatever action is necessary to protect public health and safety and may demand immediate licensee actions, up to and including a plant shutdown". How about an unusual action which may eventually terminate operation of an installation?**

   Answer: NRC has full authority to take whatever action is necessary to protect public health and safety. This includes the authority to revoke a license.

   § 2.202 Orders.

   (a) The Commission may institute a proceeding to modify, suspend, or revoke a license or to take such other action as may be proper by serving on the licensee or other person subject to the jurisdiction of the Commission an order that will:

   (1) Allege the violations with which the licensee or other person subject to the Commission's jurisdiction is charged, or the potentially hazardous conditions or other facts deemed to be sufficient ground for the proposed action, and specify the action proposed;

10. **In Section 6.3.10, does the research conducted by NRC for the reviews of advanced reactor designs include Generation-IV reactors?**

    Answer: The research conducted by the NRC does include Generation IV reactor types. For more details on the Advanced Reactor Research program, please refer to NUREG-1925:
    http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1925/r1/sr1925r1.pdf#page=201.

ARTICLE 8: REGULATORY BODY

1. **Section 8.1.5 and section 11.2 described the human resource of regulatory body**

and licensees. Aging of experienced personnel is a common problem faced by the nuclear industry around the world. How to recruit qualified young nuclear engineers and how to encourage college student major in nuclear engineering are important for the safe operation of nuclear power plant. It seems that a section of the description of the national nuclear engineering education program is required.

Answer: The *NRC Education Grant Program for Curriculum Development* and the *Nuclear Education Program,* established four and three years ago, respectively, were created to support the nuclear workforce infrastructure, and to ensure that the nuclear sector in the U.S. would have the human resources necessary to continue the safe operation of the existing fleet of 104 plants and any new builds. These programs rely on continued funding and support from the United States Congress:

1. <u>NRC Education Grant Program for Curriculum Development</u> – Funding under this program supports courses, studies, training, curricula, and disciplines pertaining to nuclear safety, nuclear security, nuclear environmental protection, and other fields that the Commission determines to be critical to the NRC's regulatory mission. Its primary purpose is supporting and developing the educational infrastructure necessary to allow the Nation to safely move its nuclear energy initiatives forward.

2. <u>Nuclear Education Program</u> – Funding under this program includes support for education in nuclear science and engineering, to develop a workforce capable of supporting the design, construction, operation, and regulation of nuclear facilities and the safe handling of nuclear materials. Included in this program are the following:

<u>Scholarship and Fellowship</u>

The Scholarships and Fellowship Program award is granted directly to accredited U.S. institutions of higher education. Individual students cannot apply directly to NRC for scholarships or fellowships. As a condition for receiving scholarships or fellowships, recipients must demonstrate satisfactory academic progress in their fields of study, as determined by criteria contained in this announcement and as established by the NRC. The nuclear education supported by this funding is intended to benefit the nuclear sector broadly. Consequently, NRC requires scholarship and fellowship recipients to serve 6 months in nuclear-related employment for each full or partial year of academic support. The employment may be with NRC, other Federal agencies, State agencies, Department of Energy laboratories, nuclear-related industry, or academia in the recipients' sponsored fields of study. A waiver of this requirement may be granted in appropriate circumstances.

Faculty Development

The Faculty Development Grants Program recognizes the need to attract and retain highly-qualified junior faculty in academic teaching careers. Funding under this announcement is intended to support new faculty in the nuclear-related fields of Nuclear Engineering, Health Physics, and Radiochemistry. The grants specifically target probationary, tenure-track faculty in these academic areas during the first 6 years of their career. Grants could include support for developing proposals for research and small amounts for initiating or continuing research projects in their areas of expertise.

Trade School and Community College Scholarship

The Trade School and Community College Scholarship supports trade school scholarships. As a condition for receiving trade school scholarships, recipients must demonstrate satisfactory academic progress in their fields of study, as determined by criteria contained in this announcement and as established by the NRC. Trade schools must be postsecondary educational institutions or programs accredited by an accrediting agency or state approval agency recognized by the U.S. Secretary of Education or be registered apprenticeship programs. The nuclear education supported by this funding is intended to benefit the nuclear sector broadly. Requirements for this scholarship are the same as under the Scholarship and Fellowship program.

ARTICLE 9: RESPONSIBILITY OF THE LICENSE HOLDER

1. **In Section 9.3, how NRC collaborate with the site management during a severe accident?**

   Answer: In accordance with NRC incident response protocols and procedures, the NRC will collaborate with site management on a number of levels: (1) the NRC resident inspectors will integrate with site personnel in the control room and the Technical Support Center (TSC) to provide hands-on support, information collection, and assessment; (2) the NRC Site Team will be dispatched to integrate with site personnel and management at the licensee's TSC and Emergency Offsite Facility (EOF) to act as the NRC's decision-making authority once it enters its Expanded Activation Mode; and (3) the Executive Team Director will call the Licensee's Emergency Director to offer him/ her any assistance that is available from the NRC or Federal family.

ARTICLE 10：PRIORITY TO SAFETY

1. **In Section 10.3.3, for the risk-informed technical specifications and licensing basis, to what extend have these programs been accomplished? Do most of the plants modify their TS and LB accordingly? What are the major improvements from these two programs as far as the safety enhancement of the power plant is concerned?**

   Answer: Consistent with the Commission=s policy statements on technical specifications and the use of PRA, the NRC and industry continue to develop risk-informed improvements to the current standard technical specifications (STS). These improvements are intended to maintain or improve safety while reducing unnecessary burden and to make technical specification requirements consistent with the Commission=s other risk-informed regulatory activities. The Electric Power Research Institute (EPRI) has developed a white paper entitled, "Safety and Operational Benefits of Risk-Informed Initiatives," which was issued in February of 2008 (publically available on the EPRI website, product identification number 1016308). This paper states tangible benefits, such as risk reductions, and intangible benefits, such as improved safety focus (by the licensee and regulator). The paper also identifies operational benefits of higher quality, greater plant flexibility, and reduced complexity. Risk-informed programs also improve the focus and resources of the licensee and regulator on issues that are truly important to safety.

   Proposals for risk-informed improvements to the STS are judged based on their ability to maintain or improve safety, the amount of unnecessary burden reduction they will likely produce, their ability to make NRC=s regulation of plant operations more efficient and effective, the amount of industry interest in the proposal, and the complexity of the proposed change.

   The current status on each of the identified eight initiatives is reported in the Office of Nuclear Regulatory Research on an annual basis in the Risk-Informed and

Performance-Based Plan (RPP) (available at:
http://www.nrc.gov/about-nrc/regulatory/risk-informed/rpp/reactor-arena/operating/op-licensing.html)

Many licensees have improved their technical specifications by implementing these voluntary risk-informed initiatives. Specifically, in the last year, two of the major initiatives are being actively pursued by licensees. Numerous licensees have made applications to implement Initiative 5B, which optimizes surveillance frequencies, and there is a pilot application planned for the coming year to implement Initiative 4B, which risk-informs completion times. Following the pilot application review on initiative 4B it is expected that a large segment of the industry will pursue this initiative over the subsequent years.

2. **In the paragraph of Section 10.3.2, it is mentioned that 12 plant-specific RI-ISI Programs have been reviewed and approved by NRC. Please provide your recommendations and important findings with related to the RI-ISI programs.**

   Answer: The reference to 12 plant-specific RI-ISI Programs being reviewed and approved by the NRC is related to a specific RI-ISI methodology (Code Case N-716). In fact, RI-ISI is one of the most widely adopted risk-informed applications, with 101 out of 104 plants having implemented an approved RI-ISI Program. RI-ISI Programs use operating experience and risk insights to target the pipe segments that present the greatest risk, considering both likelihood and consequence of failure. Due to its systematic risk-informed approach, the RI-ISI process generally identifies fewer welds for inspection than the traditional ISI program, but by focusing on the segments of greatest risk can also maintain or even reduce the risk associated with pipe failures.

3. **In Section 10.3.4, NRC had endorsed the PRA Standard of ASME/ANS-RA-Sa-2009. Please clarify the schedule to issue/ endorse the standards of LP/SD, L-2, and L-3. Will NRC consider to request the utilities to include all events and all operation modes according to the new standards? If not, what will be the major consideration?**

   Answer: The current standards development organizations have not established a firm date for the issuance of standards for low power and shutdown PRA, Level 2 PRA, or Level 3 PRA. Typically the NRC endorsement of standards, via a revision of Regulatory Guide 1.200, takes about a year and there is typically a one-year implementation period allowed for licensees to address the endorsed standard as needed for their applications. It should be noted, that current operating plant licensees are not required to upgrade their PRA to address the latest endorsed PRA standard unless the subject hazard, operating mode, or PRA Level is determined to be needed to address a specific risk-informed application. The guidance provided in Regulatory Guide 1.200, which endorses with qualifications and clarifications the latest PRA Standards, and application Regulatory Guides, such as RG 1.174 and 1.177, state that the scope, level of detail, and level of technical acceptability of a licensee's PRA needs to be commensurate with its reliance in the application. This guiding principle directs the staff reviews of the licensee's PRA for specific risk-informed applications.

ARTICLE 16：EMERGENCY PREPAREDNESS

1. **In Section 16.1, news briefing about the accident is also an important part of emergency preparedness, as it relieves the public anxiety. However, it is not stated or discussed about the necessity and importance of news briefing.**

   Answer: The NRC's Crisis Communication Plan outlines our process for dealing with the media and public communications during an incident. As part of our crisis communication process:

   1. We have pre-written press releases, preliminary statements and press briefings that can be quickly updated with relevant information and disseminated via the agency's usual distribution methodologies i.e. listserve, posting on the web site, etc.;
   2. We have a pre-arranged media bridge line that would be activated as necessary to drive all media representatives to one phone number; this line would be used for technical briefings and announcements;
   3. We have recently instituted several social media sites that would support media/public communications, including an external blog, Twitter and YouTube
   4. We have a pre-populated Emergency Event Web page that is ready to "go live" if an incident turns into a General Emergency. The web page would not just centralized relevant information about the incident, but has links to important

emergency sheltering information and links to information on other federal websites.

5. We have an arrangement with the Department of Homeland Security (DHS), which arranges federal-government-wide conference calls to coordinate federal messaging and media relations, and would participate as lead technical agency in any press briefings arranged by DHS or the White House.
6. We have a mechanism to turn out auditorium into a News Center to accommodate media who travel to NRC HQ, and to have regular media briefings at that location.

2. **In Section 16.4, NRC is currently evaluating the position of prompt evaluation for the population near a plant in a severe reactor accident, as under some circumstances, it may be better to shelter in place. Can you describe these circumstances?**

   Answer: The recommendation for "sheltering in place" is based on citizens' circumstances and the nature of the attack or accident. Citizens should understand their possible options and use common sense and available information to determine if there is immediate danger. In any emergency, local authorities may or may not be able to provide information immediately on what is happening and what citizens should do. However, citizens should monitor TV, radio news reports, and the plant's Alert and Notification System (ANS) for information or official instructions as they become available. If citizens are specifically told to evacuate or seek medical treatment, they should do so immediately.

3. **As stated in Section 16.3, there are four classes of emergencies: (1) unusual event, (2) alert, (3) site area emergency, and (4) general emergency. The specific class of emergency is declared on the basis of plant conditions that trigger the emergency action levels. Licensees have established specific procedures for carrying out emergency plans for each class of emergency. Licensees also set up a standard in determining the class of emergency during a drill or an actual accident. The standard is developed based on the emergency action levels suggested in NUREG-0654, NUMARC/NESP-007 Revision 2, or NEI-9901, Revision 4. The standard has to be approved by NRC. The emergency action**

levels specified in these documents might be different. Therefore, the class of emergency may be different when two plants under the same accident conditions. Is this inconsistency covered in the Reactor Oversight Process of emergency preparedness?

Answer: The Emergency Action Levels (EALs) specified in plant-specific documents are the same for each licensee. The NRC and industry have aligned to reach a common understanding of the classes of emergency and how the plants will respond to them. The review by the staff is to ensure consistency of the EALs irrespective of the guidance used to develop the EAL scheme. The Reactor Oversight Process and the licensee's demonstration of bi-yearly emergency preparedness drills and exercises demonstrate the ability to implement the scheme.

*Note, NUREG-0654, NURARC/NESP-007, and subsequently NEI-9901 reflect an evolution in EAL methodology to include enhancements and lessons-learned, rather than three radically different schemes.

4. **In Section 16.5, the areas for inspection in the emergency preparedness of the Reactor Oversight Process are listed. The implementation the plant specific Severe Accident Management Guidelines (SAMG) is not on the list. The development of plant specific SAMG is required by NRC and SAMG play an important role in the mitigation of the class nine accidents. The implementation of Emergency Operation Procedure by licensees is stated in section 19.4 of Article 19. Please elaborate the NRC's effort on the implementation of SAMGs of licensee.**

Answer: SAMGs are intended to be a supplement to, but not necessarily part of, a licensee's Emergency Plan. SAMGs are documents that provide information and options to consider when a licensee undergoes an event that is beyond the scope of their EOPs. In NEI 91-04, Rev 1, "Severe Accident Issue Closure Guidelines", dated December 1994 (ML072850981), the industry provided a regulatory commitment that all plants would implement SAMGs, but currently, there are no NRC regulations requiring them.

The guidance, SECYs (i.e., NRC Commission Papers), etc., that led to their development are numerous. Some key documents to review and reference are listed below, and each one lists other references for additional information if necessary:

- NEI 91-04, "Severe Accident Issue Closure Guidelines,"

- SECY-97-132, “Status of the Integration Plan for Closure of Severe Accident Issues and the Status of Severe Accident Research,”
- SECY-98-131, “Status of the Integration Plan for Closure of Severe Accident Issues and the Status of Severe Accident Research.”

The strengthening and integration of the requirements for SAMGs and Extensive Damage Mitigation Guidelines (EDMGs) is an area the NRC staff will address “without unnecessary delay,” through its 21-day action paper as tasked by the Commission in SRM-11-0093, on the “Near-Term Report and recommendation for Agency Actions Following the Events in Japan.”

ARTICLE 17：SITING

1. **In Section 17.2.2, from the recent Japanese Fukushima accident and the other seismic events in the world, it may conclude that tectonic movements are more frequent and the scale is getting higher. Will NRC consider asking licensees to modify their forcing functions for seismic and tsunami evaluation following these events? License renewal will be a good time to ask licensees to make this change and timely review of this could mean a lot if the trends go on.**

   Answer: In the Near Term Task Force Review of Insights from the Fukushima Daiichi Accident, the Task Force recommended that the NRC require licensees to reevaluate and upgrade as necessary, the design-basis seismic and flooding protection for each operating reactor. Recommendations include the following:

   - Draft a Commission policy statement that articulates a risk-informed defense-in-depth framework that includes extended design-basis requirements in the NRC’s regulations as essential elements for ensuring adequate protection.
   - Initiate rulemaking to implement a risk-informed, defense-in-depth framework consistent with the above recommended Commission policy statement.
   - Modify the Regulatory Analysis Guidelines to more effectively implement the defense-in-depth philosophy in balance with the current emphasis on risk-based guidelines.

- The Task Force believes that the Regulatory Analysis Guidelines could be modified by implementing some of the concepts presented in the technology-neutral framework (NUREG-1860) to better integrate safety goals and defense-in-depth.
- Evaluate the insights from the IPE and IPEEE efforts as summarized in NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," issued December 1997, and NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program," issued April 2002, to identify potential generic regulations or plant-specific regulatory requirements.
- Order licensees to reevaluate the seismic and flooding hazards at their sites against current NRC requirements and guidance, and, if necessary, update the design basis and SSCs important to safety to protect against the updated hazards.
- Order licensees to perform seismic and flood protection walkdowns to identify and address plant-specific vulnerabilities and verify the adequacy of monitoring and maintenance for protection features such as watertight barriers and seals in the interim period until longer term actions are completed to update the design basis for external events.

2. **In Section 17.2.2, tsunami is certainly a site-related factor that is likely to affect the safety of a nuclear installation near a coast for its projected lifetime. How is tsunami considered for your NPPs near coastal area? Have the criteria ever been revised/updated? Has any enhancement for prevention of tsunami of those NPPs been implemented during the operating period?**

   Answer: The Near Term Task Force provided the regulatory background for protection from natural phenomena as follows:

   The NRC has long recognized the importance of protection from natural phenomena as a means to prevent core damage and to ensure containment and spent fuel pool integrity. The NRC established several requirements addressing natural phenomena in 1971 with GDC 2, "Design Bases for Protection Against Natural Phenomena," of Appendix A to 10 CFR Part 50. GDC 2 requires, in part, that SSCs important to safety be designed to withstand the effects of natural phenomena such as floods, tsunami, and seiches without loss of capability to perform their safety functions. GDC 2 also requires that design bases for these SSCs reflect (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding region, with sufficient margin for the limited accuracy and quantity of the historical data and the period of time in which the data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

Since the establishment of GDC 2, the NRC's requirements and guidance for protection from seismic events, floods, and other natural phenomena have continued to evolve. The agency has developed new regulations, new and updated regulatory guidance, and several regulatory programs aimed at enhancements for previously licensed reactors.

In 1973, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100 was established to provide detailed criteria to evaluate the suitability of proposed sites and the suitability of the plant design basis established in consideration of the seismic and geologic characteristics of the proposed sites.

In 1977, the NRC initiated the Systematic Evaluation Program (SEP) to review the designs of older operating nuclear reactor plants in order to reconfirm and document their safety. The purpose of the review was to provide (1) an assessment of the significance of differences between then-current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety. The plants selected for SEP review included several that were licensed before a comprehensive set of licensing criteria (i.e., the GDC) had been developed or finalized. The SEP covered topics including seismic events, floods, high winds, and tornadoes.

In 1980, the NRC was concerned that licensees had not conducted the seismic qualification of electrical and mechanical equipment in some older nuclear reactor plants in accordance with the licensing criteria for the seismic qualification of equipment acceptable at that time. As a result, the NRC established the Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Mechanical and Electrical Equipment in Operating Nuclear Power Plants," program in December 1980. In

February 1987, the agency issued Generic Letter (GL) 87- 02, “Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46,” to address this concern. The objective of USI A-46 was to develop alternative seismic qualification methods and acceptance criteria that could be used to assess the capability of mechanical and electrical equipment in operating nuclear power plants to perform their intended safety functions. The scope of the review was limited to equipment required to bring each affected plant to hot shutdown and maintain it for a minimum of 72 hours.

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

In 1996, the NRC established two new seismic regulations for applications submitted on or after January 10, 1997. These regulations were not applied to existing reactors. The first regulation, 10 CFR 100.23, “Geologic and Seismic Siting Criteria,” sets forth the principal geologic and seismic considerations that guide the Commission in its evaluation of the suitability of a proposed site and adequacy of the design bases established in consideration of the geologic and seismic characteristics of the proposed site. The second regulation, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” to 10 CFR Part 50, requires that nuclear power plants be designed so that certain SSCs remain functional if the safe shutdown earthquake (SSE) ground motion occurs. These plant features are those necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1) or 10 CFR 100.11, “Determination of Exclusion Area, Low Population Zone, and Population Center Distance.”

In 1996, the staff also established a new requirement in 10 CFR 100.20, “Factors To Be Considered When Evaluating Sites,” for the evaluation of the nature and proximity of man-related hazards, such as dams, for applications submitted on or after January 10, 1997. This regulation was not applied to existing reactors.

In 1975, the NRC published the Standard Review Plan (SRP) (NUREG/75-087, later published as NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition”), which provides standardized review criteria to assist the staff in evaluating safety analysis reports submitted by license applicants. Since its first publication, the SRP has undergone several revisions to incorporate new developments in design and analysis technology. Since the last SRP update in 2007, the staff has established interim staff guidance (ISG) in three areas related to protection from natural phenomena: (1) DC/COL-ISG-1, “Interim Staff Guidance on Seismic Issues of High Frequency Ground Motion,” (2) DC/COLISG7, “Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures,” and (3) DC/COL-ISG-20, “Seismic Margin Analysis for New Reactors Based on Probabilistic Risk Assessment.” This interim guidance has been applied only to new reactor reviews.

The staff has also published several regulatory guides (RGs) that address specific technical issues related to protection from natural phenomena. These documents provide guidance to licensees and applicants on implementing specific parts of the NRC’s regulations, techniques used by the NRC staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits or licenses. These guides include the following:

- RG 1.29, “Seismic Design Classification,” issued in 1972 and updated in 1973, 1976, 1978, and 2007
- RG 1.59, “Design Basis Floods for Nuclear Power Plants,” issued in 1973 and updated in 1976 and 1977
- RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” issued in 1973
- RG 1.102, “Flood Protection for Nuclear Power Plants,” issued in 1975 and updated in 1976
- RG 1.125, “Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants,” issued in 1977 and updated in 1978 and 2009

• RG 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," issued in 2007

The NRC staff continually evaluates new information regarding natural phenomena, including operational experience, and its potential impact on risk and overall plant safety. These evaluations have led to new requirements or guidance as discussed above, updated regulatory guidance, generic communications, and plant-specific actions to address identified issues. Several examples are presented below.

Following the Sumatra earthquake and its accompanying tsunami in December 2004, the NRC staff initiated a study to examine tsunami hazards at nuclear power plant sites, to review offshore and onshore modeling of tsunami waves, to describe the effects of tsunami waves on nuclear power plant SSCs, to develop potential approaches for screening sites for tsunami effects, to identify the repository of historic tsunami data, and to examine ways for an NRC reviewer to approach site safety assessment for a tsunami. The study, NUREG/CR-6966, "Tsunami Hazard Assessment at Nuclear Power Plant Sites in the United States of America," was published March 2009. The results of this study were incorporated in the 2007 update of SRP Section 2.4.6, "Probable Maximum Tsunami Hazards." As discussed in NUREG/CR-6966, the 1977 revision to RG 1.59 (Revision 2) was expected to include guidance for assessment of tsunamis as a flooding hazard, but that effort was not completed. The staff is in the process of updating RG 1.59 to address tsunamis and other advances in flooding analysis. Since 1977, flood estimation techniques have significantly improved with the availability of more accurate datasets and newer hydrologic, hydraulic, and hydrodynamic models. It should be noted that the current fleet of reactors was sited before RG 1.59, Revision 2, was issued.

In August 2010, the NRC initiated a proposed generic issue (GI) regarding flooding of nuclear power plant sites following upstream dam failures. The staff evaluation of this issue is ongoing.

Lastly, the NRC is evaluating seismic hazards based on new Electric Power Research Institute models used to estimate earthquake ground motion and updated models for earthquake sources in the Central and Eastern United States. The NRC is addressing this issue through the ongoing evaluation of GI-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," initiated June 9, 2005. The results of the GI-199 safety/risk assessment stage were summarized in Information Notice 2010-018, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," dated September 2, 2010. As discussed in Information Notice 2010-018, currently available seismic data and models show increased seismic hazard estimates for some operating nuclear power plant sites in the Central and Eastern United States. Determination of site-specific seismic hazards and associated plant risk are planned for the next phase of GI-199.

ARTICLE 18：DESIGN AND CONSTRUCTION

1. **In Section 18.4, since there hasn't been any new nuclear power plant construction work for quite a long time in US, how will NRC look at the problem of vendor technology capacity gap which might exist due to the large retirement of design and construction engineers? Will NRC initiate any program to identify this problem in the vendor's conduct?**

   Answer: There is no specific guidance or initiatives for assessing aging workforce impacts on vendor activities. Vendor QA inspections routinely look at the training and qualification packages of individuals within vendor QA programs. Lack of experience and inadequate training are intended to be assessed at that point with respect to the vendor's compliance with their QA program and its ability to meet the 10 CFR 50, Appendix B criteria. Repeated adverse quality issues would cause the ordering entity to restrict or remove a non-performing vendor from their approved vendors list.

ARTICLE 19：OPERATION

1. **In Section 19.4, we suspect that in the Japanese Fukushima accident, the compound external events (higher than expected earthquake plus tsunami) caused multiple failure of equipment. Will NRC plan to modify the initiating events so that the multiple failure of equipment in Fukushima type accident or other compound external events could be envisaged?**

   Answer: The Near Term Task Force provided the background for concurrent related events as follows:

   The staff initiated Generic Safety Issue (GSI)-172, "Multiple System Responses

Program (MSRP)," to address 21 potential safety concerns that were raised by the Advisory Committee on Reactor Safeguards (ACRS) during the resolution of USI A-17, "Systems Interactions in Nuclear Power Plants"; USI A-46, "Seismic Qualification of Equipment in Operating Plants"; and USI A-47, "Safety Implications of Control Systems." GSI-172 included the ACRS concern that the resolution of USI A-46, other seismic requirements, or fire protection regulations did not adequately address seismically induced fires. This concern was identified as Item 7.4.16 in NUREG/CR-5420, "Multiple System Responses Program—Identification of Concerns Related to a Number of Specific Regulatory Issues," published October 1989. ACRS was also concerned that previous internal flooding studies had examined events such as pipe ruptures (and subsequent flooding) as single events and that the nature of a seismic event could cause such problems in multiple locations simultaneously. This concern was identified as Item 7.4.18 in NUREG/CR-5420.

The staff developed guidance for the review of the safety concerns of GSI-172 in the Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) programs. As a result, the IPEEE program subsumed the issues related to seismically induced fires and floods.

With regard to seismically induced fires, NUREG-1742 states the following:

> All of the IPEEE submittals reported that the licensees qualitatively examined seismically induced fire interaction issues as part of the treatment of Sandia fire risk scoping study issues. A few licensees performed a PRA study for seismically induced fire-initiating events; albeit the level of detail varied from a simplistic probabilistic analysis to inclusion in their plant's seismic or fire PRA.

In most of the submittals, licensees included seismically induced fire considerations within the scope of their overall seismic walkdown. The level of effort, scope, and detail directed toward addressing seismically induced fire issues varied significantly among the IPEEE submittals. One licensee did not discuss seismically induced fire evaluations in their IPEEE submittal. In most other cases, licensees limited their seismically induced fire evaluations exclusively to assessing direct impacts on safe shutdown equipment.

Seismically induced flooding events can potentially cause multiple failures of safety-related systems. The rupture of small piping could provide flood sources with the potential to affect multiple safety-related components simultaneously. Similarly, nonseismically qualified tanks are a potential flood source of concern. While some licensees proposed plant improvements to address related issues, NUREG-1742 states that the level of effort, scope, and detail directed toward addressing seismically induced flooding issues varied significantly among the IPEEE submittals. Some plants did not provide any information in their IPEEE submittals to verify this issue.

The GSI-172 issue regarding seismically induced fires and floods was closed based on the IPEEE results, and the NRC established no new requirements to prevent or mitigate seismically induced fires or floods. The Task Force concludes that the agency should reevaluate the closure of GSI-172 in light of the plant experience at the Kashiwazaki-Kariwa nuclear plant and the potential for common-mode failures of plant safety equipment as the result of seismically induced fires and floods.

The Task Force recommended, as part of a longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.

2. **In Section 19.5, for the availability of engineering and technical support under accident or emergency condition, which agency or institution in US will be responsible to provide proper support for, say, evacuation, massive decontamination, radioactive medical care, etc.?**

   Answer: In accordance with the National Incident Management System, States and local responders are responsible for the initial response actions to protect public health and safety; including evacuation, decontamination, and primary care of their constituents. If the event exceeds State and local capabilities and resources, Federal assistance may be requested to provide long-term event response and recovery support. The National Response Framework and its Nuclear/ Radiological Incident Annex, specifies that the Federal Emergency Management Agency (FEMA), in conjunction with the Department of Energy, the Environmental Protection Agency, and the Department of Health and Human Services, will provide emergency assistance, as needed.

INPO

1. **On page 182, the INPO evaluation team "provides the utility with reports of strengths and areas for improvement, along with a numerical rating of overall plant performance." (1) Please comment on the mean and standard deviation of the rating for the recent 5 years.**
2. **(2) On page 183, what is percentage for plants in category 3 and 4? (3) Do they move to category 1 or 2 after evaluation often? (4) What is the correlation between category rating and plant performance?**

Questions 1, 2, and 4 were addressed using the assessment results from January 1, 2006 to September 21, 2011. Question 2 required a look at more data (20 years) to get reliable probabilities.

(1) What is the mean and standard deviation of INPO assessments over the past five years?

Answer: Mean = 1.81, standard deviation = 0.82

(2) What is the percentage of 3 and 4 assessments in that five-year population?

Answer: 17.9 percent

(3) What is the probability of stations assessed a 3 or 4 moving out of these categories?

Note: a 20-year population was used in this analysis to provide a more-reliable probability of movement between assessments.

Answer:

- Probability of going from a 3 to a 1 in the subsequent evaluation is 0.008.
- Probability of going from a 3 to a 2 in the subsequent evaluation is 0.474
- Probability of going from a 4 to a 1 in the subsequent evaluation is 0.000
- Probability of going from a 4 to a 2 in the subsequent evaluation is 0.095

(4) What is the correlation of assessments to station performance over the past five years?

Answer: the correlation (r) is -0.6333

Note: This correlation was performed by measuring the assessments to the INPO Performance Indicator Index (PII). The PII is a measure of station performance and is used when determining the INPO assessment; however, it is not the only input and it plays a relatively minor role in determining the assessment.

Review Questions and Comments for 「The Republic of China National Report for the Convention on Nuclear Safety」

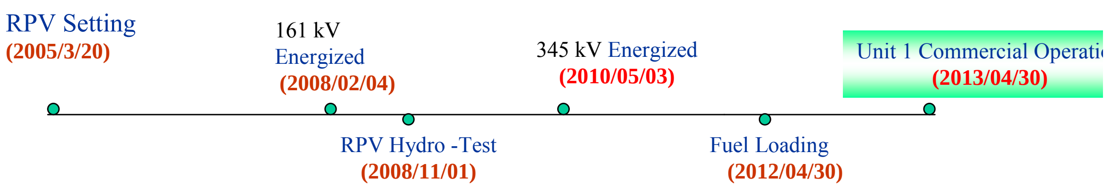
ARTICLE 6：EXISTING NUCLEAR INSTALLATIONS

1. **The report states that the construction of the Lungmen Nuclear Power Plant is still under way. Can you provide more details of the milestones for the construction of this Nuclear Power Plant?（Section 6.1, page 2）**

   Answer: The Key Milestones for the Project are outlined in the attached Table.

   Unit No. 1 Milestones (Unit No. 2 is 1 year late)

Key Milestones	Planned Schedule
RPV Setting	2005/03/20 (completed)
RPV Hydro -Test	2008/11/01 (completed)
161 kV Energized	2008/02/04 (completed)
345 kV Energized	2010/05/03 (completed)
Fuel Loading	2012/04/30
Commercial Operation	2013/04/30



   *TPC will issue the revised schedule in February 2012.

2. **The report states that the licensee is required by Regulations to conduct a comprehensive safety assessment of the operating NPP every ten years, similar to the international practice of the periodic safety review (PSR). Can you describe one or two improvements to plants that have been achieved as a result of this comprehensive safety assessment process?（Section 6.2.3.2, page 5）**

   Answer: The purpose of 10-year periodic safety review is: to conduct re-assessment consistent with present day state of knowledge, analytical methods, and equipment, e.g. new seismic methodology and digital seismometer and to identify potential aging problems. Examples of some important improvements include the following:

- Each Taipower unit installed an Automatic Seismic Trip System (ASTS) in order to strengthen the ability to safely shut down the reactor in the event of strong earthquake and enhance public confidence in the operation of nuclear power plants. The system is designed to automatically trip reactor under an earthquake stronger than OBE. To comply with the existing Reactor Protection System control logic of each unit, there are four or three groups of independent channels of seismic sensors in ASTS.
- System identification program of Seismic safety reevaluation has been installed in Kuosheng and Mannshan Nuclear Power Plants, and will be installed in Chinshan Nuclear Power Plant . Chinshan Nuclear Power Plant upgraded seismometers in 2005 and recently add two sets in an independent building. Kuosheng and Mannshan Nuclear Power Plants upgraded seismometers in 2006 and 2008 respectively.
- Most of cables within drywell in Chinshan Nuclear Power Plant , suffer for high temperature, have been replaced with cables to endure higher temperature after 1994. The cables of motor operated valves in upper area of drywell in Kuosheng Nuclear Power Plant have been replaced with cables to endure higher temperature.
- Each Taipower site has two offsite power sources (345kV and 69/161kV) for startup. Originally there was only one 345kV startup transformer in each site. Due to one aging failure incident in 2007, Taipower had added one more 345kV backup startup transformer and related equipment in each site.

3. The report states that AEC has implemented a compact reactor oversight process (ROP) system and that it now consists of 15 preformance indicators and 4 inspection indicators.（Section 6.2.3.4, page 7）

(1) Can you provide some examples of how this system has helped you in monitoring plant performance?

(2) Could you describe or list all performance and inspection indicators?

Answer:

(1) Two examples are delineated as follows.

- The performance indicator for the unavailability rate of the reactor core isolation cooling（RCIC）system of the Chinshan Unit 1 was assigned white in color from the fourth quarter of 2005 to the fourth quarter of 2007. AEC requested Taipower to take actions to troubleshoot the equipment, including root cause, maintenance, surveillance, monitoring, etc. Through the implementation of a series of actions, Taipower restores the intended function of RCIC. Since then, the indicator shows green in color.
- On March 23, 2011, the AEC' resident inspector conducted a flood protection inspection and found NSCW pump house of Maanshan Nuclear Power Station (MNPS) had two holes. Flooding through the holes could have impacted the ability of all NSCW pumps to perform their design accident mitigation functions. Based on this finding, a Level IV violation is identified. The reason for the violation has been determined not to meet the requirement of the updated final safety analysis report. Moreover, AEC held meetings with the licensee and requested MNPS to take corrective measures. Now the two holes are sealed according to their design requirements.

(2) The Atomic Energy Council (AEC) adapted the reactor oversight process developed by NRC to evaluate safety performance of operating nuclear power plants since 2006. The system combines the performance indicators (PIs) provided by nuclear power plants with inspection findings of AEC. The quarterly results are transformed into "color" similar to the traffic light. The green, white, yellow, and red colors indicate different safety status of each nuclear power unit with the red shows the strongest safety concerns. AEC will take regulatory actions based on the "color", and a tool called "PRiSE" has been developed by AEC to determine the risk associated with each inspection finding. AEC has also prepared the inspection procedures since 2006, and the results are released on AEC's English web site at: http://www.aec.gov.tw under the directory of Nuclear Reactor Safety/Reactor Oversight Program.

ARTICLE 10：PRIORITY TO SAFETY

1. **The report states that in 2009, TPC developed a unified Corrective Action Program (CAP) for all existing NPPs with the purpose of integrating problem solving across plants. Can you describe some improvements you have seen in**

the resolution of corrective actions as a result of the integrated program?（Section 10.5, page 76）

Answer: All condition/event reports processed in CAP system, in addition to the excellent examples and strengthening measures, are to be classified and encoded for trend analyses. The majority of site condition/event reports are on the grade 3 and 4, the lower severity levels, as expected. The purpose of a trend analysis is to collate low-impact problems of NPPs and make early detections of poor performance as a weakness detection tool. And the results would also be used for references on the importance of the annual works.

ARTICLE 14：ASSESSMENT AND VERIFICATION OF SAFETY

1. **The report states that an aging management system will be developed for each NPP regarding the planning, organizing, execution and control of the aging management process. Has the development of this aging management system started? Can you provide more details of the milestones for completion?（Section 14.1.4.2, page 116）**

   Answer: Due to Japan Fukushima Daiichi Nuclear Power Plant Accident in March 2011, Taiwan energy policy has been revised in November 2011. There will be no life extension beyond 40 years for all nuclear units. The aging management programs for each plant will continue to ensure the safety of operation. The AMPs for Chinshan and Kuosheng have the preliminary evaluation results. The implementation will assign to responsible person in each plant. The preliminary evaluation results for Maanshan will come out in 2014. In addition to the ASME section XI requirements, reactor vessel internal inspection, water chemistry, structure monitoring, piping and equipment inspection are among the most important AMPs. These AMPs will be revised to include the operational experience.