SEPTEMBER 2016

THE REPUBLIC OF CHINA NATIONAL REPORT FOR THE CONVENTION ON NUCLEAR SAFETY

ATOMIC ENERGY COUNCIL, EXECUTIVE YUAN

TAIWAN, REPUBLIC OF CHINA

EXECUTIVE SUMMARY

The Republic of China (hereafter referred to as ROC or Taiwan) has not yet signed the Convention on Nuclear Safety (hereafter referred to as the CNS or the Convention) 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. As long as an international activity is helpful to the promotion of this nation's nuclear safety, Taiwan always likes to participate in it and fulfill the relevant obligations. Thus, despite being a non-contracting party of the CNS, Taiwan is willing to act as a contracting party to meet all the requirements addressed in the applicable articles established by the Convention. This report is the updated version of the ROC National Report for the CNS for peer review in the Bilateral Technical Meeting between the USA and the ROC scheduled to be held in September 2017. It evaluated the implementation of the safety requirements of the Convention by the nuclear power plants (NPPs) in Taiwan and demonstrated how Taiwan fulfilled the obligations addressed in Chapter 2 (including Article 4 and Articles 6 to 19) of the Convention.

In Taiwan, there are currently a total of four land-based civil NPPs. Among them, three are operating, namely Chinshan, Kuosheng and Maanshan NPPs (abbreviated as CSNPP, KSNPP and MSNPP, respectively). The fourth one, Lungmen NPP (abbreviated as LMNPP), is currently put into mothballs.

According to the sequence of their corresponding project starting dates, Chinshan, Kuosheng, Maanshan and Lungmen NPPs are also called the First, Second, Third and Fourth NPPs, respectively in this nation. Each of these four NPPs has two identical nuclear power reactor units.

Each unit of the Chinshan NPP has one boiling water reactor of type 4 (BWR-4) with Mark I containment, while the reactors of the Kuosheng NPP are BWR-6 with Mark III containment. On the other hand, each unit of the Maanshan NPP is equipped with one 3-loop pressurized water reactor (PWR) with a steel-lined pre-stressed post-tensioned large dry reinforced concrete containment. For the Lungmen NPP, the reactor is an advanced boiling water reactor (ABWR) with a concrete of the cylindrical steel-lined reinforced concrete structure integrated with the reactor building, i.e., a so-called reinforced concrete containment vessel (RCCV). All four NPPs are owned and operated or will be operated by the Taiwan Power Company (hereafter referred to as TPC or Taipower), which can be viewed as a state-owned utility (stocks: 96.92% government owned, 3.08% private owned).

The regulatory body for all nuclear and radiation-related affairs in Taiwan is the Atomic Energy Council (hereafter referred to as the AEC), which is currently at the second-tier ministry level in the governmental organization and reports directly to the Executive Yuan (EY) (i.e. the Cabinet) or the Premier.

In this report, the specific improvements made in the regulatory requirements and activities of the regulatory body (AEC) as well as in the nuclear power operational safety of the license holder (TPC) were described. A number of important nuclear regulatory laws and acts have been strengthened and legislated or re-legislated during the years of 2000s. The enforcement rules and regulations related to these laws and acts were strengthened since then.

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 safe operation of the plant, and for an easily understood indicator of the safety status of an operating NPP to the public.

The surveillance of the NPP operation and/or construction by the regulatory body is strictly implemented by inspections and documents 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 other inspections when needed.

Ever since its installation, each of the three operating NPPs in Taiwan adopted the customer technical specifications (CTS) (Chinshan NPP) or standard technical specifications (STS) (Kuosheng and Maanshan NPPs). Later, on Feb. 26, 2002, the Chinshan NPP converted its CTS into the improved technical specifications (ITS). Because of the fruitful implementation of ITS in Chinshan, the STSs for both Maanshan and Kuosheng NPPs were also converted into ITS in September 2004 and January 2008, respectively. Therefore, all three operating NPPs currently adopt the ITS.

In order to strengthen the robustness of a nuclear unit against an accident of loss of power sources, the power supply systems design of the NPPs in Taiwan is based on the defense-in-depth concept. Every nuclear power unit has several lines of defense in preventing the occurrence of loss of power.

Besides, to improve the public's knowledge about radiation and nuclear power, the AEC directly communicates with the public through its facebook and thus transports the fundamental knowledge about nuclear safety and radiation protection. AEC also tried to visit and have dialogue with the environmental protection groups and nuclear-concerned organizations as frequently as possible. Furthermore, AEC invited members of non-governmental organizations (NGOs) to be its Council Members in order to have voice from broader aspects.

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 to the public. An on-site nuclear emergency response drill is required for each NPP to be conducted every year. Meanwhile, a national nuclear emergency exercise (NEE) is conducted annually with one of the operating NPPs as the reference plant. Through the experiences obtained from these drills and exercises, the contingency plans for the emergency response and emergency preparedness (EP) are in place and will be continually updated and improved.

After the Japanese Fukushima Daiichi NPP accident in March 2011, a national "Programs for Safety Re-assessment" for all NPPs in Taiwan was immediately initiated in April 2011. Furthermore, a nuclear stress tests (ST) program based on the European Union (EU) "stress tests" specifications was conducted for each NPP in August 2011, in order to well utilize the EU stress tests experiences. A final national report for the Safety Re-assessment Program, entitled "Comprehensive Safety Reassessment Report for NPPs in Taiwan in Response to the Lessons Learned from Fukushima Daiichi Accident" (written in Chinese), was published by AEC in August 2012, while the formal "Taiwan Stress Test National Report for Nuclear Power Plants" by AEC was issued on May 28, 2013. International peer reviews of the ST National Report as well as the licensee's ST reports by the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD/NEA) and the European Nuclear Safety Regulators Group of the European Commission (EC/ENSREG) expert teams, respectively, were also conducted. Based on results of these safety re-assessment and stress tests, a variety of lesson-learned requirements and orders to strengthen the nuclear safety of NPPs in Taiwan were issued by AEC to TPC. The TPC may propose alternatives subject to AEC approval.

Current abnormal operating procedures (AOPs) and emergency operating procedures (EOP) are symptom-based. They are suitable for handling internal events. When there is a large-scale severe compound external event, like the Fukushima accident, of which the effects are on the entire plant site, an urgent response will be required and thus the symptom-basis procedures may be required to enhance. Therefore, after the Fukushima Daiichi nuclear accident, each NPP of the TPC has developed an alternate rescue strategy, the ultimate response guidelines (URG), with respect to the plant specific features. The development of the URG is subject to review by the AEC.

Application of initial fuel loading (IFL) authorization for Unit 1 of the Lungmen NPP has been submitted to AEC by TPC on Dec. 31, 2013 and is not yet granted. In response to the Fukushima Daiichi accident in 2011, the Taiwan government issued a new nuclear energy policy in the President's remarks at a press conference on November 3, 2011 that no life extension of the three operating NPPs will be granted and the nuclear power will be gradually reduced under three major principles. These principles are no power rationing, reasonable electricity price maintained, and pledges to the international community to reduce this nation's carbon emission fulfilled. The government further announced on April 28, 2014 that the Lungmen NPP units 1 and 2 would be mothballed and whether to operate this NPP in the future or not would be decided by a referendum. The Lungmen NPP was formally mothballed on July 1, 2015 for a temporary period of 3 years.

On the other hand, with an intention to make the government structure smaller, an "Amendments to the Law Governing Organization of the Executive Yuan" was passed by the Legislative Yuan (LY) (i.e., the Congress) of Taiwan in January 2010, reducing the legal number of second-tier ministries from 37 to 29. Accordingly, a Government Reform Program for the Executive Yuan (EY) started in January 2012. In the governmental structure proposed by this reform program, the authority of nuclear regulation is no longer a 2nd tier ministry member of the Cabinet (Executive Yuan), but a 3rd-tier independent organization called the "Nuclear Safety Commission." In addition, the "Institute of Nuclear Energy Research (INER)," which is the primary research and development (R&D) organization to technically support the AEC in carrying out the nuclear regulation affairs, will no longer be an affiliated institute to the new nuclear regulatory body. To compensate the loss due to the leaving of INER, a task force called the "Nuclear Regulatory Technology Support Center" was formed to give the technical support to the AEC. Whether this downgrade in the nuclear regulatory body's level in the governmental structure will affect the efficiency and effectiveness of nuclear regulation is to be seen. A draft of "Nuclear Safety Commission Organization Act" approved by the Executive Yuan (EY) on February 21, 2013 was submitted to the Legislative Yuan (LY) on the same day for review and a formal legislative approval is still pending.

In conclusion, Taiwan has demonstrated its fulfillment of the Principles of the Vienna

Declaration on Nuclear Safety (VDNS), consideration of challenges identified during the 6th review meeting under the CNS, and implementation of lessons learned from the Fukushima Dai-ichi accident through the activities of the AEC and its licensee, the TPC, in all aspects of three domestic operating NPPs. Detailed information can be found in the following Articles of this report.

Table of Contents

LIST OF FIGURES

LIST OF TABLES

INTRODUCTION

A. General

The Republic of China (hereafter referred to as the ROC or Taiwan) is not a Contracting Party to the Convention on Nuclear Safety (hereafter referred to as the CNS or the Convention) 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 positive in promoting the nuclear safety here, Taiwan is willing to participate, if possible, and fulfill the relevant obligations. This ROC's National Report for the CNS showed how the obligations under the terms of the CNS were fulfilled by this nation.

The indigenous energy resources are severely scarce in Taiwan. In 2014, the total primary energy supply was about 147.453 x 10^6 kilo-liter oil equivalent (KLOE), an increase of 3.02% as compared to that in the previous year. 98.04% of the total energy supply is imported. The indigenous energy supply contributed only 1.96% of the total, a decrease of 3.75% as compared to that in the previous year. When classified by the kinds of energy sources, the nuclear power constituted about 8.33% of the total, an increase of 1.80% as compared to that in the previous year. The fossil power contributed 89.95% to the total, including coal 29.20%, oil 48.52% and natural gas 12.23%. As for the renewable energy, the hydropower constituted about 0.28%, power from biomass and waste 1.24%, while the combined contribution from the wind power and solar power was about 0.21%. Since 2001, coal was no longer produced domestically and the supply of it depended totally on imports. As for the crude oil and natural gas, they were very limited in this country.

In the aspect of nuclear power utilization, there are currently three nuclear power plants (NPPs) in operation in Taiwan, namely, Chinshan, Kuosheng and Maanshan, and one NPP in mothballs, Lungmen. The construction work of the Lungmen NPP Unit 1 was nearly completed, but the entire palnt was put into mothballs on July 1, 2015, following the government policy in response to the public environment in this nation after the Fukushima nuclear accident. Each of these four NPPs has two identical nuclear power units. All NPPs in Taiwan are owned and operated (or will be operated) by the Taiwan Power Company (hereafter referred to as TPC or Taipower), a state-owned utility with 96.92% of the its stock held by the government and 3.08% by the private.

As will be shown in Table 6.2, the expiration dates of the 40-years operating license for the Chinshan Unit 1 & 2 are December 5, 2018 and July 15, 2019, while those for the Kuosheng Unit 1 & 2 are December 27, 2021 and March 14, 2023, respectively. According to the regulations, the life-extension application of a NPP must be submitted 5 ~15 years before the expiration of the operating license (OL) to the regulatory body by the licensee. Thus, The TPC submitted the life-extension application of Chinshan NPP in July 2009 to the AEC. This review is the first time of such kind in this country and will normally take 24 ~ 26 months. During the reviewing period, because of the happening of the Fukushima accident in March 2011 and the implementation of the stretch power uprate program of the Chinshan plant in 2012 (Subsection 6.1.2), the TPC requested to postpone the review. This review was re-started in August 2014 and estimated to need about 2 more years. However, due to the Japanese Fukushima Daiichi NPP Accident in March 2011, Taiwan energy policy has been revised in November 2011. According to the previous administration of this nation, there will be no life extension beyond 40 years for all nuclear units, if power rationing is not required, a reasonable power price can be maintained, and pledges to reduce carbon emissions can be fulfilled. However, after the inauguration of the new president on May 20, 2016, the current national policy is “no life extension of nuclear power plants.”

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 or the so-called independent power producers (IPPs) 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, 78.3% in 2003, to about 77% in 2015.

By the end of December 2015, the total domestic installed power capacity in the TPC power grid system was 41.04 GWe as shown in the Table I-1, including the capacity installed by the TPC itself and the independent power producers (IPPs). The installed power capacity by the TPC was 31.424 GWe, contributing about 76.6% to the total. The private power plants contributed about 9.620 GWe or 23.4%. Among the installed power capacity by the TPC, fossil power contributed 21, 570 MWe or 52.6% of the total, while nuclear power contributed 5,144 MWe (not including the power uprates) or 12.5%, renewable energy including hydro-power (excluding the pumped storage power), wind power and solar power, contributed about 2,110 MWe or 5.1%, and the pumped storage hydro-power capacity stood for about 2,600 MWe or 6.3%.

Table I-1 Installed Power Capacity in Taiwan in 2015

Type of Energy Source	Installed Capacity (MWe)	Percentage (%)
TPC:		
Oil	3,330	8.1
Coal	7,600	18.5
LNG	10,640	25.9
Pumped Storage Hydro	2,600	6.3
Renewable	2,110	5.1
Nuclear	5,144	12.5
(TPC Subtotal:)	(31,424)	(76.6)
IPP:	9,620	23.4
Total	41,040	100

Source: 1. 2016 Taiwan Power Company Sustainability Report, “Corporate Highlight”.

2. 2015 Annual Report of the Bureau of Energy, MOEA.

In 2015, the total electricity generated in Taiwan reached 219.1 billion kWh. Of which, 169.1 billion kWh, representing about 77.2% of the total, was generated by the TPC. The nuclear power generated about 35.1 billion kWh of electricity, representing about 16% of the total. As for the utilization of the renewable energy in 2015, it contributed about 4.2% of the total electricity generation or 9.3 billion kWh (including 4.8 billion kWh from IPP). The electricity generated from the pumped storage hydro-power stood for about 1.4%. Table I-2 shows the electricity generated in Taiwan in 2015.

Table I-2 Electricity Generation in Taiwan in 2015

Type of Energy Source	electricity generation (billion kWh)	Percentage (%)
TPC:		
Oil	10.3	4.7
Coal	57.1	26.0
LNG	59.1	27.0
Pumped Storage Hydro	3	1.4
Renewable	4.5	2.1
Nuclear	35.1	16.0
(TPC Subtotal:)	(169.1)	(77.2)
IPP:	50	22.8
Total	219.1	100

Source: 2016 Taiwan Power Company Sustainability Report, "Corporate Highlight".

All NPPs in Taiwan are owned and operated (or will be operated) by the TPC. With the best efforts done by the TPC staffs, the performance of the three operating NPPs continued to be very good in recent years (refer to Figure 6.6). The weighted annual capacity factor (ACF) for all six operating nuclear units was 94.07% in 2014. Generally speaking, the operations of NPPs in the ROC are quite satisfactory in terms of safety and reliability.

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. There are three affiliated organizations under the AEC, including the Institute of Nuclear Energy Research (INER), the Fuel Cycle and Materials Administration (FCMA), and the Radiation Monitoring Center (RMC). INER is the sole nuclear R&D institute in this country. The major nuclear R&D areas of the INER in recent years consisted of the evaluation of license renewal for a NPP, the medium and small scale power uprates study, level 2 probabilistic risk assessments (PRA) of the operating NPPs, source term evaluation, seismic risk re-assessment of a NPP, high efficient solidification technology (HEST) study for the low level waste (LLW), nuclear facility decommissioning and radioactive waste management, radiobiological medicine R&D, the establishment of the accreditation platform for the nuclear grade industrial technologies, etc. (More information about INER's R&D programs can be found in Subsection 6.3.8.) FCMA has two major responsibilities: first, the safety regulation of the treatment, transportation and final disposal of the radioactive wastes including both LLW and the spent nuclear fuels (SNF); 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.

After the Fukushima Daiichi accident, a national "Programs for Safety Re-assessment" was initiated and later the nuclear stress tests (ST) for each NPP based on the EU "stress tests" specifications were conducted. The robustness of NPPs with respect to the power supply systems among others has thus significantly increased.

This ROC National Report for the CNS of 2016 is a self-standing document and there is no need to get familiar with the earlier reports in advance. In the following Article 6, the nuclear power plants (NPPs) in Taiwan are discussed of their design features, power uprates, power unit performance, power supply sources, etc. Also discussed in Article 6 are the re-assessment of nuclear safety, the stress tests for all Taiwanese NPPs and a variety of regulatory requirements and orders issued by the regulatory body to the licensee with the intension to strengthen the robustness of the NPP after the Fukushima accident.

Article 7 gives an overview of the 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 ROC. Besides the Atomic Energy Act, there are six basic laws, six subsidiary enforcement rules and the related regulations.

Article 8 describes the mission and the structure of the regulatory body. The AEC, in the implementation of regulatory tasks and R&D works, adheres to the following principles: safety first, reasonable control, and convenience to the people. The AEC consists of 15 commissioners besides the Chairman (Minister) and 2 deputy Chairmen (deputy Ministers), four technical departments and four administrative units within the Headquarters in addition to eight advisory committees on nuclear policy and safety. Besides, under the AEC's supervision, there are three affiliated organizations.

In Article 9, the main subjects are the mechanism for the license holder to discharge its prime responsibility for safety and the mechanism for the regulatory body to ensure that the license holder will meet its prime responsibility for safety

Article 10 gives the overview of the arrangements and requirements to prioritize safety, the voluntary activities and good practices related to safety, and the enhanced transparency of nuclear safety information to ensure that"Safety" has always been the top priority in the country.

As for Article 11, the financial requirements of the licensee including the financing of safety improvements, financial provisions for decommissioning and radioactive waste management, financial protection program for liability claims arising from nuclear accidents, etc. as well as the manpower resources of the licensee are addressed.

Article 12 presents the overview of human factors and organizational issues for the safety of NPPs, the human factors in the design of NPPs and subsequent modifications, the methods to prevent, detect, and correct human errors, the managerial and organizational issues, the role of the regulatory body and facility operator, and the lessons learned from the accidents at Fukushima to ensure the good human performance in the NPPs.

Article 13 describes the quality assurance (QA) policy, the requirements and programs which are implemented for the NPPs in stages of design, procurement, manufacturing, construction, commissioning, operation and maintenance, and the lessons learned from the accidents at Fukushima to ensure that the regulatory body and the license holder will meet its prime responsibility for safety.

Article 14 depicts the comprehensive and systematic safety assessments throughout the plant life, the verification of plant safety by analysis, surveillance, testing and inspection, and the lessons learned from the accidents at Fukushima to ensure the prevention of disaster resulted from combined accidents of the TPC's NPPs.

Article 15 mentions about the purpose of the Ionizing Radiation Protection Act (IRPA) which 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. In this article the spirit of "As Low As Reasonable Achievable" (ALARA) is emphasized for reduction of occupational radiation exposure and protection of radiation exposure for members of the public.

Article 16 focused on the emergency preparedness (EP), for either on-site or off-site of any nuclear reactor facility. Based on "The Nuclear Emergency Response Act", the response mechanisms have been established. The emergency response organizations include (1) National Nuclear Emergency Response Center, (2) Radiation Monitoring and Dose Assessment Center, (3) Regional Nuclear Emergency Response Center and (4) Nuclear Emergency Support Center. After Fukushima accident in 2011, the emergency planning zones (EPZ) for the three NPPs were all enlarged from 5 to a new 8 kilo-meters from the center of the nuclear power station.

Article 17 describes the evaluation of site-related factors affecting the safety of a NPP, the evaluation of the safety impact on individuals, society, and environment, and the reevaluation of site related factors after Fukushima accident to cope with extreme natural disasters including earthquake, tsunami, and flooding.

Article 18 presents the protection (defense in depth) of NPPs against the release of radioactive material, the application of proven technologies to assure the safety of NPPs, and the Fukushima lessons learned to re-visit the design basis to verify the capability of NPPs in response to both the design basis accident (DBA) and the beyond design basis accident (BDBA).

Finally, activities related to the operation of NPPs in Taiwan are thoroughly evaluated in Article 19 including the development of the ultimate response guidelines (URG) procedure by the TPC to prevent the reactor core from melting in the beyond design basis conditions.

It is worth noting that the operational experience feedback (OEF) for lessons learned from the Fukushima accident are discussed in several subsections including, for example, Subsections 6.1.4.5, 6.4, 12.6, 13.4, 14.3, 15.3, 16.4, and 18.4.

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.

B. Lessons Learned from the Fukushima Daiichi Accident

Since the Fukushima Dai-ichi accident, a number of technical and administrative measures to increase the plant robustness have been taken at Taiwanese NPPs. These measures were mainly identified during Programs for Safety Reassessments and Stress Tests for NPPs in Taiwan. Examples of these safety enhancement measures include the AC and DC power supply sources enhancement, the safety enhancements against seismic/tsunami hazards, the issuance of new regulatory orders based on results from NPP reassessment and stress tests, and the development of the ultimate response guidelines (URGs). For more information, please see Subsections 6.2.3.5, 6.4, 12.6, 13.4, 14.3, 15.3, 16.4, 18.4 and 19.8.4.

C. Adoption of Findings from Peer Review Missions

Findings and recommendations from international peer reviews such as those from the OECD/NEA and EC/ENSREG peer review teams on the NPP stress tests have been adopted by the regulatory authority (AEC) and the licensee (TPC). For more information, please see Subsections 6.2.3.6, 6.4.1, 10.1.5, 14.3, 17.3, 18.4, etc.

D. Challenges Identified by the Special Rapporteur at the 6th Review Meeting

During the 6th review meeting of the CNS (24 March – 4 April 2014), five challenges were identified by the special rapporteur, Mr. Petteri Tiippana (Finland), on Fukushima for consideration of the contracting parties in their next national reports under the CNS:

Challenge 1: How to minimize gaps between Contracting Parties' safety improvements?

Challenge 2: How to achieve harmonized emergency plans and response measures?

Challenge 3: How to make better use of operating and regulatory experience, and international peer review services?

Challenge 4: How to improve regulators' independence, safety culture, transparency and openness?

Challenge 5: How to engage all countries to commit and participate in international cooperation?

Actions already taken in Taiwan on Challenge 1 included being a member of international organizations such as WANO and INPO as well as participating various international cooperation programs such as CSARP, CAMP, CPD, and CODAP. For several decades, bilateral conference between Taiwan and Japan nuclear communities as well as the TECRO-AIT Joint Standing Committee meeting on civil nuclear cooperation are being held annually, which are beneficial to either sides. For more information, please see Subsection 6.3.5.1.

To harmonize emergency plans and response measures with respect to Challenge 2, each NPP in Taiwan conducted a nuclear emergency drill every year, while a national nuclear emergency exercise (NEE) was conducted annually with the reference NPP being that, in turn, in the south or north. For more information, please see Subsections 16.1.2 and 19.8.4.

As for Challenge 3, the operational experience feedback (OEF) programs conducted in Taiwan are beneficial to the safety improvement of the NPPs via the periodic safety review (PSR), which is called the 10-year integrated safety assessment (ISA) in this nation. For more information, please see Subsections 6.2.3.4, 10.1.4.1, and 19.3.

Referring to Challenge 4, the AEC has implemented a reactor oversight process (ROP) system and published the performance indicators and inspection indicators of the Operating NPPs in Taiwan in its website in order to improve the regulator's transparency and openness. Besides, all regulations and more and more information about the AEC activities are also put in the AEC's website. On the other hand, the TPC also presents information about the company in six aspects, including the information on management, power generation, demand & supply of electricity, customers, environment, and construction engineering, to the public through the TPC's web site. Furthermore, an advanced safety culture program has been developed by TPC since June 2011 to foster a high level of nuclear safety to ensure the health and safety of the general public. This advanced SC program includes 4 major areas: management effectiveness, contractor management, risk management and personnel performance. For more information, please see Subsections 6.2.1, 6.2.3.3, 10.1.3, 12.6.1 and 19.10.

Like Challenge 1, Challenge 5 has been addressed by being a member of international organizations and participating the international cooperation programs. For more information, please see Subsection 6.3.5.1.

E. Principles of the Vienna Declaration on Nuclear Safety

In the 2015 Vienna Declaration on Nuclear Safety (VDNS), three principles for implementing the objective of the CNS were addressed as follows:

1. New nuclear power plants are to be designed, sited, and constructed, consistent with the objective of preventing accidents in the commissioning and operation and, should an accident occur, mitigating possible releases of radionuclides causing long-term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long-term protective measures and actions.

2. Comprehensive and systematic safety assessments are to be carried out periodically and regularly for existing installations throughout their lifetime in order to identify safety improvements that are oriented to meet the above

objective. Reasonably practicable or achievable safety improvements are to be implemented in a timely manner.

3. National requirements and regulations for addressing this objective throughout the lifetime of nuclear power plants are to take into account the relevant IAEA Safety Standards and, as appropriate, other good practices as identified inter alia in the Review Meetings of the CNS.

These three principles of the VDNS are all fulfilled in Taiwan. The fulfillment of Principle 1 is addressed in more detail in Subsections 6.2.3.1, 10.1.4.1 and 19.1 (although the new NPP being constructed was put into mothball). Principle 2 was adopted and implemented through the carrying out of the 10-year integrated safety assessment (ISA) program and addressed in more detail in Subsections 6.2.3.4 and 14.1.2(2). On the other hand, Principle 3 was enhanced through the conducting of “Programs for Safety Reassessments and Stress Tests for NPPs” in Taiwan, which resulted in the issuance of new regulatory orders and the licensee’s safety enhancement measures in accordance. For more information, please see Subsections 6.4.1, 6.4.2, 6.4.3, 10.1.5.2, and 14.3.4.

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 Nuclear Power Plants in Taiwan

There are a total of four land-based civil nuclear power plants (NPPs) in Taiwan. Among them, the Chinshan (CS), Kuosheng (KS) and Maanshan (MS) NPPs are currently in operation, while the Lungmen (LM) NPP whose construction was nearly completed is in mothballs. CS, KS, MS and LM NPPs are also named in this nation as the First, Second, Third and Fourth NPPs, respectively, according to the sequence of their corresponding project starting dates. Each of these four NPPs has two identical nuclear power units.

By the end of December 2014, the total installed rated capacity of nuclear power was 5,144 MWe (not including the power uprates) from the 3 operating NPPs, representing about 12.61 % of the total installed capacity in Taiwan's power system which reached about 40,787 MWe. The total installed capacity of the power plants of the Taiwan Power Company (TPC or Taipower) was about 31,651 MWe. Should the Lungmen NPP be able to come into operation someday in the future, the total installed rated nuclear power capacity will become 7,844 MWe.

In 2014, the electricity generated from the nuclear power was about 40.801×10^9 kWh, which contributed about 18.61% to the total domestic supply of electricity.

When the Fukushima nuclear accident happened in March 2011, the construction of the Lungmen NPP with two ABWR units was nearly completed. Application for the initial fuel loading (IFL) of the Lungmen NPP Unit 1 was submitted to the Atomic Energy Council (AEC) by TPC on Dec. 31, 2013. However, after the Fukushima accident, the Government of Taiwan revised its nuclear policy and announced to mothball the Lungmen NPP on April 28, 2014. The Lungmen NPP was formally put into mothballs on July 1, 2015 with a period set at 3 years.

All nuclear power units in Taiwan are owned and operated or will be operated by the TPC which can be viewed as a state-owned utility (with 96.92% of its stock held by the government and 3.08% by the private).

6.1.1 Plant Site, Ground Elevation and Tsunami Runup

Chinshan, Kuosheng and Lungmen NPPs are located along the northern coast of the Taiwan Island, while Maanshan NPP is situated in the southern coast of Taiwan, as shown in Figure 6.1. The distance between Chinshan and Kuosheng NPPs is about 12 kilometers (km) with Chinshan located to the west of Kuosheng, while Lungmen lies to the east of

Kuosheng.

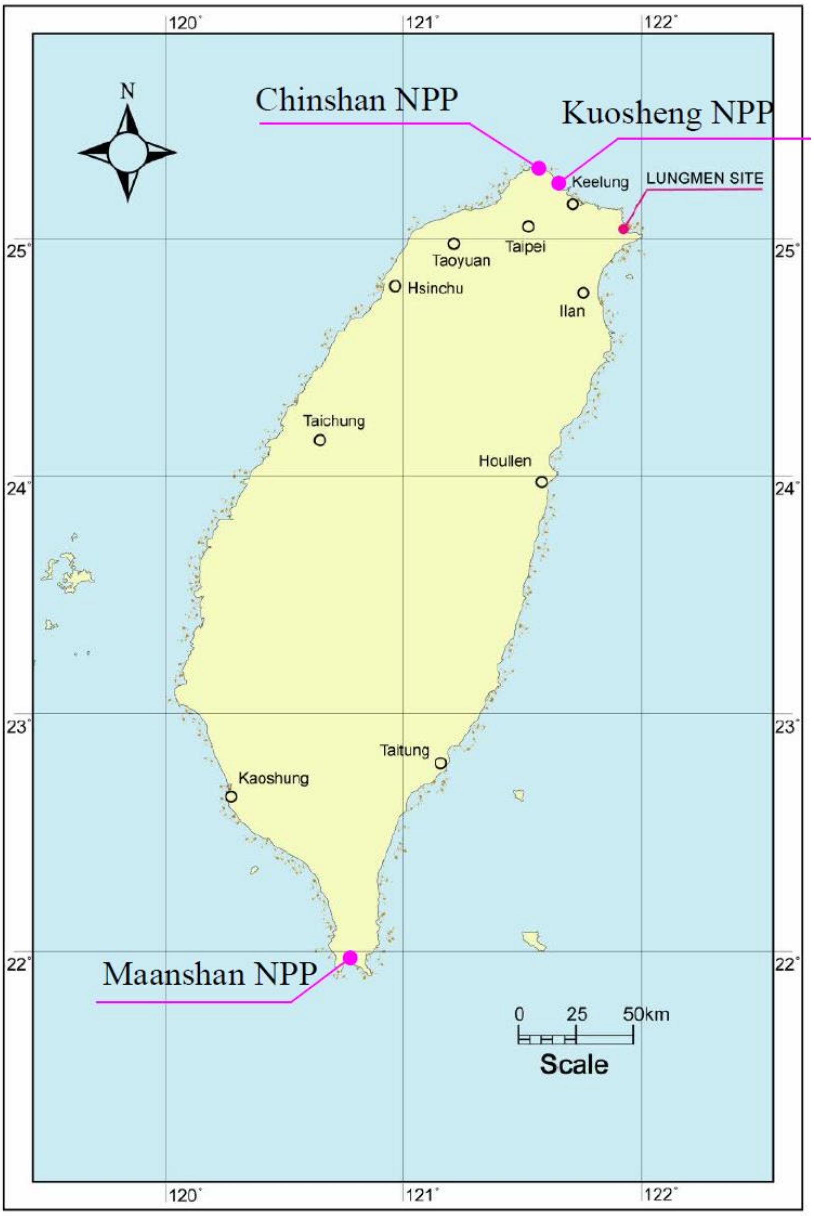


Figure 6.1 Site Locations of the NPPs in Taiwan

(1) <u>Chinshan NPP</u>

The Chinshan NPP (CSNPP), located at the north end of Taiwan, fronts the East China Sea. It is about 500 meters (m) from the sea shore. The ground elevation of the plant site, according to the most recent survey, is about 11.2 m above the mean sea level (MSL).

There are two creeks passing through the nearby area of the Chinshan plant. The Hsiokeng Stream, located to the east of the plant, is a 4-km long stream and about 3.5 km away from the plant. According to the Mudslide Disaster Prevention Center (of the Soil and Water

Conservation Bureau, Council of Agriculture, Executive Yuan), the Hsiokeng Stream is not a mudslide-prone stream. The other creek, the 8-km long Chien-hua Stream located to the west of the plant whose potential of mudslide was judged as medium, is about 4.2 km away from the plant. In order to protect against mudslide due to the Chien-hua Stream, the stream path in the downstream portion was changed to an artificial drainage trench with jetty, protection slope. Both streams flow into the East Sea and were evaluated with a result of no mudslide potential.

According to the CSNPP's final safety analysis report (FSAR) (in Section 2.6.1.1), the potential maximum tsunami run-up height was 9 m. After adding the height of tide wave 1.73 m, the total potential maximum tsunami run-up height becomes 10.73 m which is set as the design basis (DB) tsunami height. Therefore, the FSAR recommended that the ground elevation of the plant buildings should be higher than 11 m. Accordingly, during the construction period the target elevation of the main area in the plant was set at 12 m above the MSL. However, according to the recent re-evaluation result done by the Sinotech Engineering Consultants, Ltd in 2011, the Chinshan plant ground elevation was 11.2 m above the MSL, still higher than the potential maximum tsunami run-up height of 10.73 m. Figure 6.2 shows the elevations of major facilities in the Chinshan NPP.

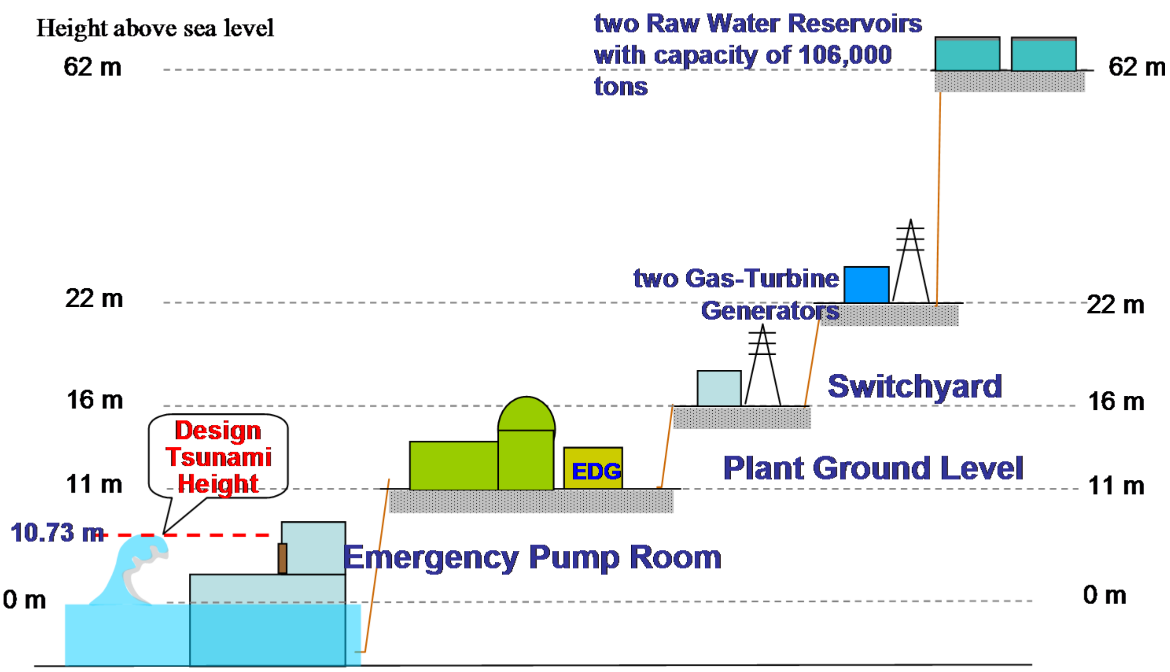


Figure 6.2 Elevations of Major Facilities in Chinshan NPP

(2) <u>Kuosheng NPP</u>

Similar to the Chinshan NPP, the Kuosheng NPP (KSNPP) is also located on the northern coast of Taiwan and fronts the East China Sea. Both NPPs are separated by a distance of about 12 km. The ground elevation of the Kuosheng NPP is about 12 meters above MSL and the plant is about 500 m from the shore.

There are two streams passing by the Kuosheng NPP, the Yuantan Creek and Mashu Creek, both flowing to the East China Sea. These two streams are separated from the plant by a hill (the sand dunes rising up to several tens of meters above MSL). Both streams were evaluated with the result of no mudslide potential.

According to the evaluation result of the KSNPP's FSAR, the potential maximum tsunami run-up height was 7.78 meters. If tides and geographical landscape are taken into account, the potential maximum tsunami run-up height will be 10.28 meters, which is lower than the plant ground elevation of 12 meters. Figure 6.3 shows the elevations of major facilities in the Kuosheng NPP.

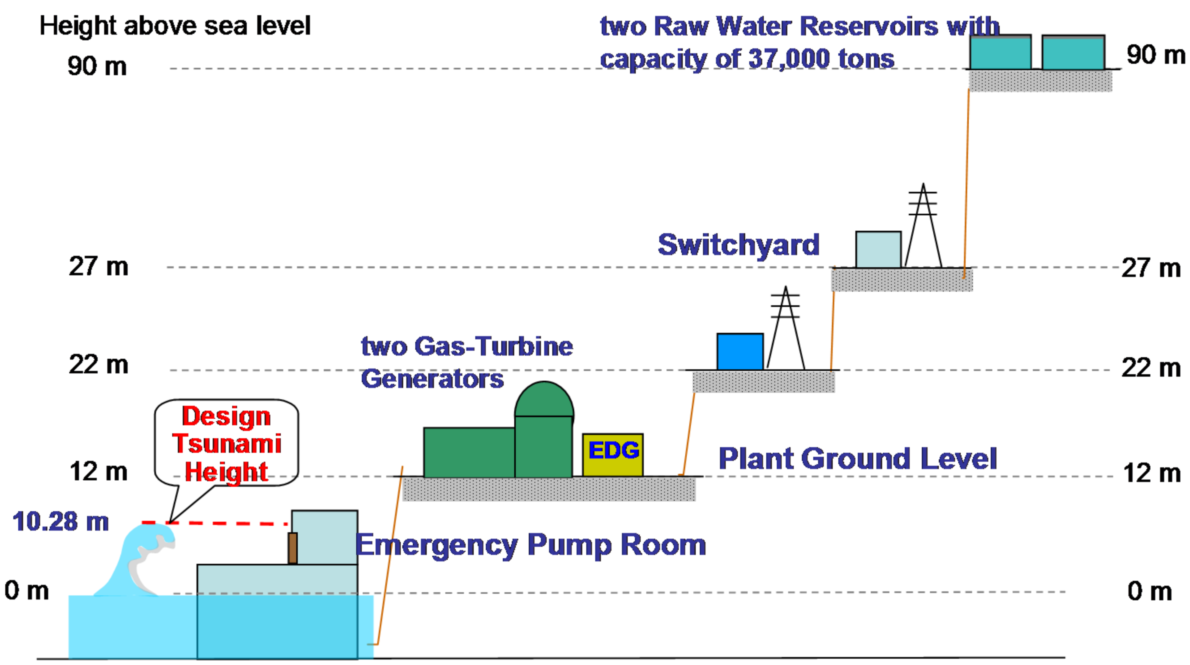


Figure 6.3 Elevations of Major Facilities in Kuosheng NPP

(3) Maanshan NPP

The Maanshan NPP (MSNPP) is located at the southern tip of Taiwan as shown in Figure 6.1. It is about 300 meters from the shore. The plant site elevation is about 15 meters above MSL. There is no creek, no water dam, and no potential mudslide inside the plant site. However, a nearby lake called Lung-luan pond is located to the north of the plant. The Maanshan plant is protected from the overflow of this lake by a mountain in between the lake and the plant site.

The maximum tidal range in the historic record of the neighboring oceanic area is 4.03 m, which happened at the Kaohsiung Port. Assuming a magnitude 8.0 (M_w) earthquake induces a tsunami with 5 m high wave at the epicenter location, the height of the tsunami wave at the coast of Maanshan will be approximately 11 m, and the tsunami run-up height near the plant will be approximately 8 m. Conservatively adding a 4.03 m spring tide rise and 0.5 m safety margin, the potential maximum tsunami run-up level at the plant will be 12.53 m, which is the design basis (DB) tsunami height. Comparing to the ground elevation of the main building area (15 m above MSL), the plant thus has enough protection against tsunami. Figure 6.4 shows the elevations of major facilities in the Maanshan NPP.

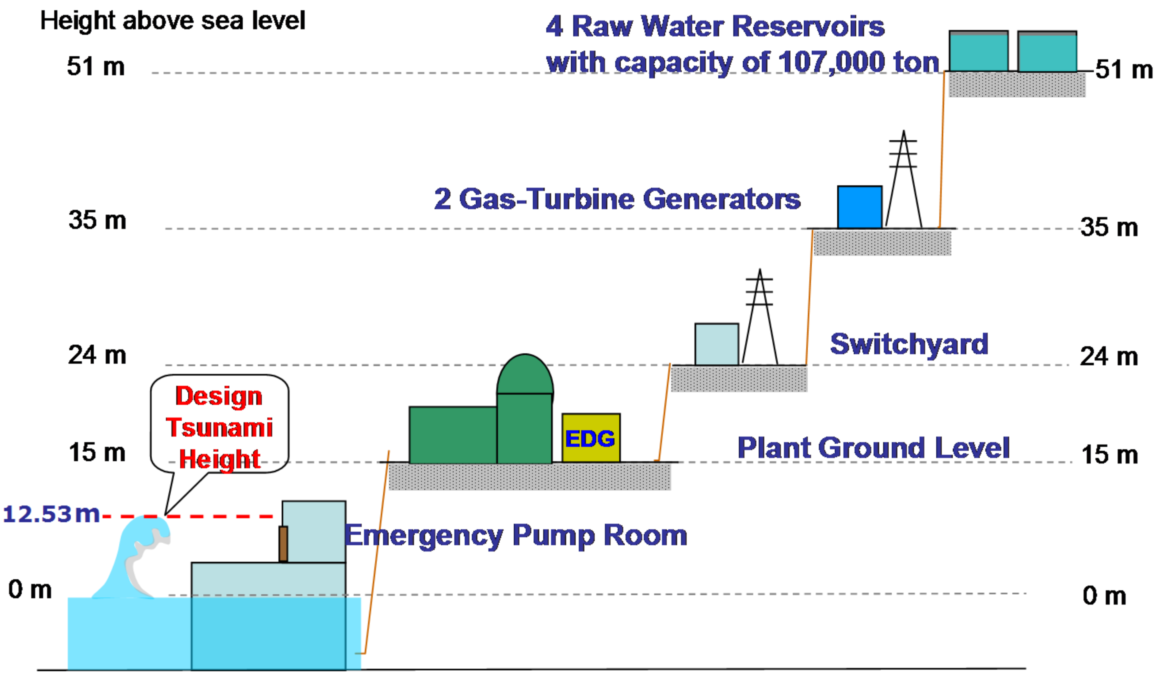


Figure 6.4   Elevations of Major Facilities in Maanshan NPP

(4) Lungmen NPP

The Lungmen NPP (LMNPP) site has a concave shoreline facing the Pacific Ocean, while the other sides (the western part of the plant) are surrounded by hills. Most of the site elevations are about 12 to 30 m above MSL and the plant is about 600 m from the shore. The ground elevation of the main building area is 12 m above MSL. There are two streams in the vicinity of the plant site with the Shihting Creek located in the north side and Shuang-Chi Creek in the south side of the plant. These two creeks were not included in the list of potential mudflow areas determined by the Soil and Water Conservation Bureau of the Council of Agriculture. There is a raw water reservoir located on the west side hill.

According to the evaluation result of FSAR for Lungmen NPP, the potential maximum tsunami run-up height is 7.5 m. If change of tides is considered, the potential maximum tsunami run-up is about 8.07 m. Conservatively adding another safety margin 0.5 m of freeboard, the design basis (DB) tsunami run-up height becomes 8.57 m, which is lower than the ground elevation of the main building area (12 m above MSL). Figure 6.5 shows the elevations of major facilities in the Lungmen NPP.

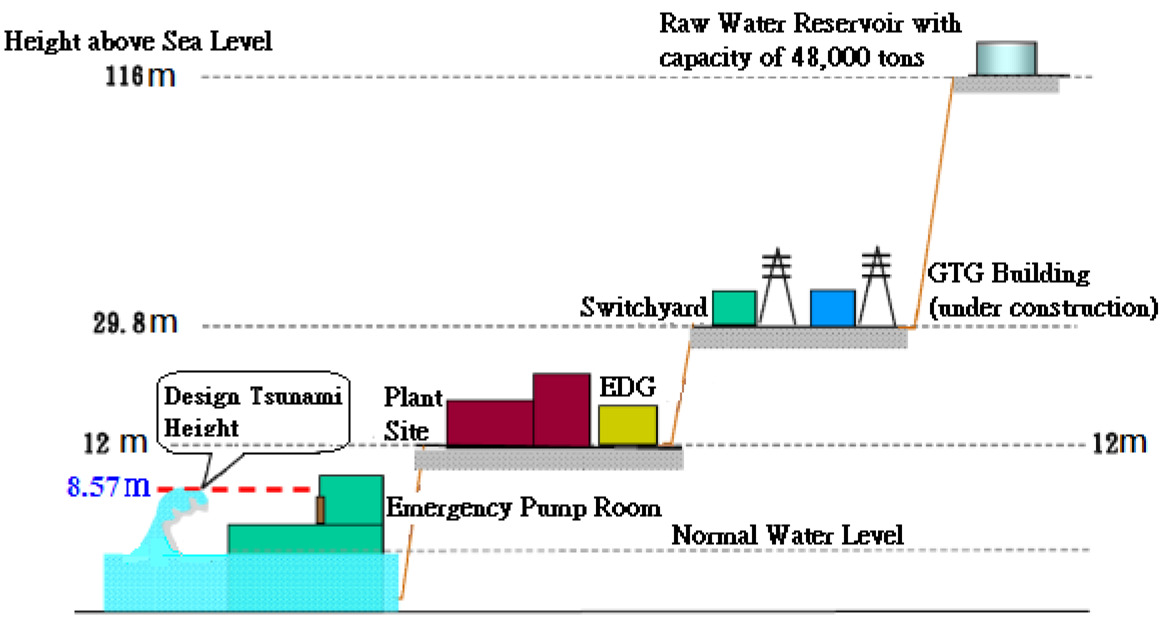


Figure 6.5 Elevations of Major Facilities in Lungmen NPP

Table 6.1 Summary of Design Basis Tsunami Heights and Plant Ground Elevations of NPPs in Taiwan

Plant Name	Chinshan	Kuosheng	Maanshan	Lungmen
Potential maximum tsunami run-up height (by FSAR)	10.73 m	10.28 m	12.53 m	8.57 m
Plant ground elevation	11.2 m	12.0 m	15.0 m	12.0 m

In summary, Table 6.1 lists the plant ground elevations and the potential maximum tsunami run-up heights of existing NPPs in Taiwan.

6.1.2 Nuclear Power Unit Characteristics, SSCs Shared and Power Uprates

Among the 3 operating NPPs in Taiwan, the CSNPP is equipped with a boiling water reactor of type 4 (BWR-4) with Mark I containment per unit and each unit of the KSNPP has one BWR-6 with Mark III containment, while each unit of the MSNPP is equipped with one 3-loop pressurized water reactor (PWR) with the steel-lined pre-stressed post-tensioned large dry reinforced concrete containment.

As for the 4th NPP (LMNPP), which was put into mothballs, it has two advanced boiling water reactors (ABWR). The containment vessel of the LMNPP is a cylindrical steel-lined reinforced concrete structure integrated with the reactor building, i.e., the so-called reinforced concrete containment vessel (RCCV). The reactor building provides a secondary confinement around the primary containment vessel.

The nuclear steam supply systems (NSSS) of the four BWRs of Chinshan and Kuosheng

NPPs as well as the two ABWRs of the Lungmen NPP were all designed and manufactured by the General Electric Company (GE). However, the vendor of all the main turbine generators (TG) for Chinshan and Kuosheng NPPs was the Westinghouse Electric Corporation (W), while the main turbine generators of the Lungmen NPP were manufactured by the Mitsubishi Heavy Industries (MHI). On the other hand, the NSSS of the two PWRs of the Maanshan NPP were supplied by W with GE manufacturing the main TG sets.

The first nuclear power unit installed in this country was the Chinshan Unit 1, which was initially critical on October 16, 1977 and started its commercial operation on December 6, 1978. The initially critical and first commercial operation dates of the Chinshan Unit 2 were November 9, 1978 and July 16, 1979, respectively. Units 1 and 2 of the Kuosheng NPP were initially critical on February 1, 1981 and March 26, 1982, respectively, and were first commercially operated on December 28, 1981 and March 15, 1983, respectively. As for the Maanshan NPP, Units 1 and 2 were initially critical on March 30, 1984 and February 1, 1985, respectively with the first commercial operation on July 27, 1984 and May 18, 1985, respectively.

The two ABWR reactors of the Lungmen NPP are of a standard design certified by the Nuclear Regulatory Commission of the United States (USNRC). However, 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 (construction and operating) licenses (COL), is not applicable in Taiwan. The nuclear reactor licensing regulations of this nation still follow a two-step licensing process similar to that of the USNRC 10CFR50. The application for constructing the Lungmen NPP with two ABWR units was submitted to the 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. Although the construction was almost completed at this moment, the Lungmen NPP was put into mothballs on July 1, 2015 due to the adverse public environment which became even worse after the Fukushima Daiichi accident in March 2011. Whether the Lungmen NPP can start operation or not in the near future is uncertain.

A brief summary of the basic design data of all the existing NPPs in Taiwan is given in Table 6.2. An overview of the main technical characteristics of these NPPs is presented in Annex 1.

Table 6.2 Basic Data of the Nuclear Power Units in Taiwan

	Chinshan		Kuosheng		Maanshan		Lungmen	
	Unit 1	Unit 2	Unit 1	Unit 2	Unit 1	Unit 2	Unit 1	Unit 2
Construction Permit Issued Date	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 Date	Dec. 6, 1978	July 16, 1979	Dec. 28, 1981	Mar. 15, 1983	July 27, 1984	May 18, 1985	–	–
Expiration	Dec. 5,	July 15,	Dec. 27,	Mar. 14,	July 26,	May 17,		

Date	2018	2019	2021	2023	2024	2025	–	–
Reactor Type (Vendor)	BWR-4 (GE)		BWR-6 (GE)		3-loop PWR (W)		ABWR (GE)	
Rated Thermal Power	1,840 MWt		3,001 MWt		2,822 MWt		3,926 MWt	
Installed Electrical Power capacity	636 MWe		985 MWe		951 MWe		1,350 MWe	
TG Vendor	W		W		GE		Mitsubishi	
A/E	Ebasco		Bechtel		Bechtel		S & W	
Containment	Mark I		Mark III		Large, Dry Post-Tensioned Reinforced Concrete		Reinforced Concrete Containment Vessel	
DBE: SSE OBE	0.3 g (PGA) 0.15 g (PGA)		0.4 g (PGA) 0.2 g (PGA)		0.4 g (PGA) 0.2 g (PGA)		0.4 g (PGA) 0.2 g (PGA)	

Note. TG ─ turbine generator; A/E ─ architect/engineer;
DBE ─ design basis earthquake; SSE ─ safe shutdown earthquake;
OBE ─ operating basis earthquake; PGA ─ peak ground acceleration.

SSCs Shared:

Basically the structures, systems and components (SSCs) of the twin units in a NPP in Taiwan are all identical. Their designs are all based on the concepts of redundancy and defense-in-depth. All safety-related systems are designed to have two trains of components and equipments. These two trains are physically separated as well as electrically and mechanically independent from each other. Any single component failure or malfunction will not affect the functioning of the related safety system. It is designed that the safe shutdown of reactor can be achieved as long as one train is normally functioning. However, there are still some important SSCs shared by both units. The potential negative impacts due to the sharing of these SSCs have already been assessed and were considered in the related operating procedures.

For example, in all 4 NPPs besides the multiple water-cooled emergency diesel generators (EDGs) separately equipped for the multiple trains in each unit, there is an additional air-cooled swing EDG (the 5th EDG in the CS, KS or MS NPP, while the 7th EDG in the LM NPP) installed in common for both units. Originally the swing EDG was designed to take over the design function of the water-cooled EDGs of either unit 1 or unit 2 when they are not available, but not for both units. After the Fukushima accident, the Station Blackout (SBO) Procedure has been revised so that under appropriate control and management, the swing EDG can supply power to the safety-related 4.16kV essential bus of both units simultaneously in case of emergency.

Another example of the SSCs shared is that in Maanshan NPP, not only the 5th EDG is in common for both units, the nuclear service cooling water (NSCW) system building is also shared by both units, but all systems and equipments in this building belonging to units 1 and 2, respectively, are separated.

<u>Power Uprates</u>:

A power uprate (PU) program was first completed in the Kuosheng NPP for Unit 2 in July 2007 and Unit 1 in November 2007 with the use of the measurement uncertainty recapture (MUR) technique by installing an ultrasonic feedwater flow rate measurement system. Similar MUR power uprate programs were also implemented and 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.

Later, a further power uprate program, the stretch power uprate (SPU), was conducted in Chinshan and Kuosheng NPPs. The SPU programs of about 2% increases in the thermal power outputs for Chinshan Units 1 & 2 were completed on November 23 and November 29, 2012, respectively with the electrical power output increases of about 12.0 and 12.4 MWe, respectively. On the other hand, the SPU programs of about 1.2% increase in the thermal power outputs for Kuosheng Units 1 & 2 were completed on July 9 and July 18, 2014, respectively and the increases in the electrical power outputs were 7.1 and 8.4 MWe, respectively. The total power uprated due to these SPU programs was about 40 MWe, which is equivalent to an annual increase in total electricity generation of 0.31 x 10^9 kWh, assuming a plant capacity factor (CF) of about 0.9. More information about the power uprate programs in Taiwan is described in Subsection 14.1.2(5) of this report.

6.1.3 Performance of the Operating NPPs

With the best efforts done by the TPC staff, the performance of the three operating NPPs was steadily maintaining excellent records in recent years. In terms of the performance indicators (PI) of the World Association of Nuclear Operators (WANO), the PI values of all six operating nuclear units in 2014 were better than the WANO median values in the areas of the unit capability factor (UCF), forced loss rate (FLR), and fuel reliability (FR) as shown in Table 6.3. Currently these performance indicator items remain the major areas targeted by the three operating NPPs to improve their performance.

Table 6.3 Comparison of the Performance Indicators of the TPC's Nuclear Power Units with the WANO Median in 2014

PI [(a)]	CS-1	CS-2	KS-1	KS-2	MS-1	MS-2	WANO-2014 (median)
1. UCF (%)	86.98	91.99	92.37	93.92	89.83	90.15	86.78 (3 yr av)*
2. UCLF (%)	0.26	0.71	0.04	0.30	0.62	0.53	1.69
3. FLR	0.29	0.51	0.04	0.33	0.54	0.65	0.89

4. UA7	0.00	0.00	0.00	0.00	0.00	0.00	0.00 (world av)*
5. SSP (%)							
BWR(HPSI)	0.00	0.02	0.00	0.00			0.80 (3 yr av)
(RHR)	0.28*	0.22*	0.07	0.01			0.50 (3 yr av)
PWR(HPSI)					0.07	0.00	0.10 (3 yr av)
(AFS)					0.61	0.02	0.00 (3 yr av)
EPS	0.010	0.787	0.01	0.001	0.843	0.000	0.20 (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-6	1.9E-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	1.14	1.25	1.49	1.44			1.11 (3 yr av)
PWR					0.56	0.60	0.46 (3 yr av)
9. ISAR ($2x10^5$man-hrs)	0.16	0.16	0.00	0.00	0.00	0.00	0.00
10. CISAR ($2x10^5$man-hrs)	0.00	0.00	0.00	0.00	0.00	0.00	0.00
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, which are for use in this Table only:

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

In the meanwhile, the average annual capacity factor (CF or ACF) of all six operating nuclear units in recent years has been maintained at around 90% as shown in Figure 6.6. The average CF of all 6 operating units in 2014 was 94.07%. Figures 6.7 and 6.8 show the trends of the average annual numbers of the reportable event reports (RER) per unit and the automatic scrams per unit, respectively.

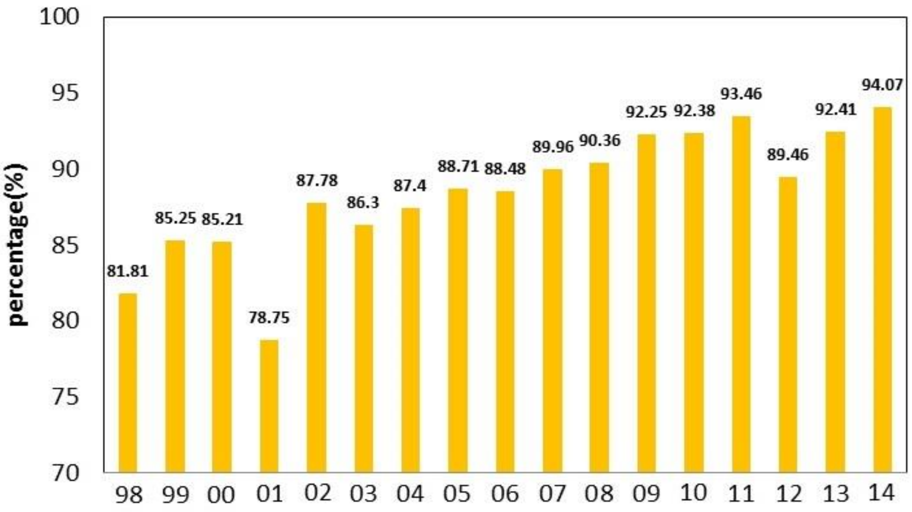


Figure 6.6 Average Annual Capacity Factor of the Operating NPP Units in Taiwan (up to the year 2014)

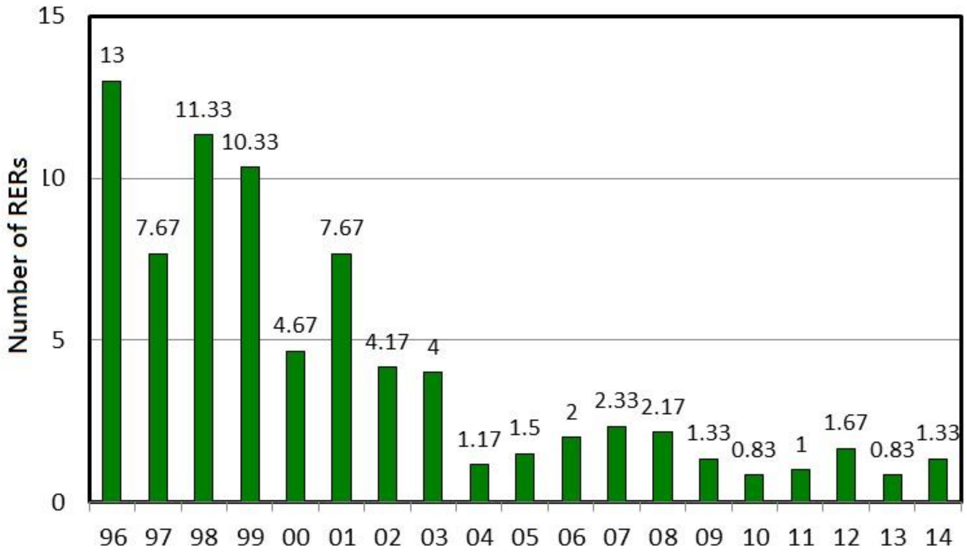


Figure 6.7 Average Annual Number of RERs per Unit for the Operating NPPs in Taiwan (up to the year 2014)

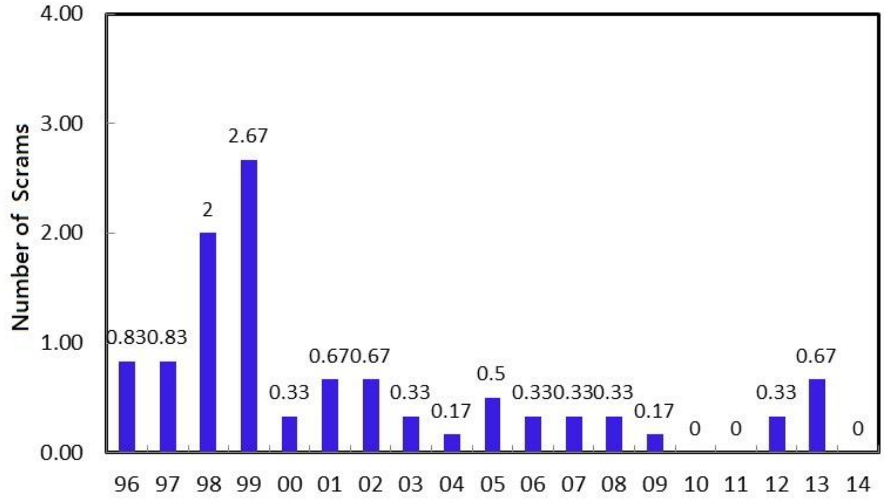


Figure 6.8 Average Annual Number of Scrams per Unit for the Operating NPPs in Taiwan (up to the year 2014)

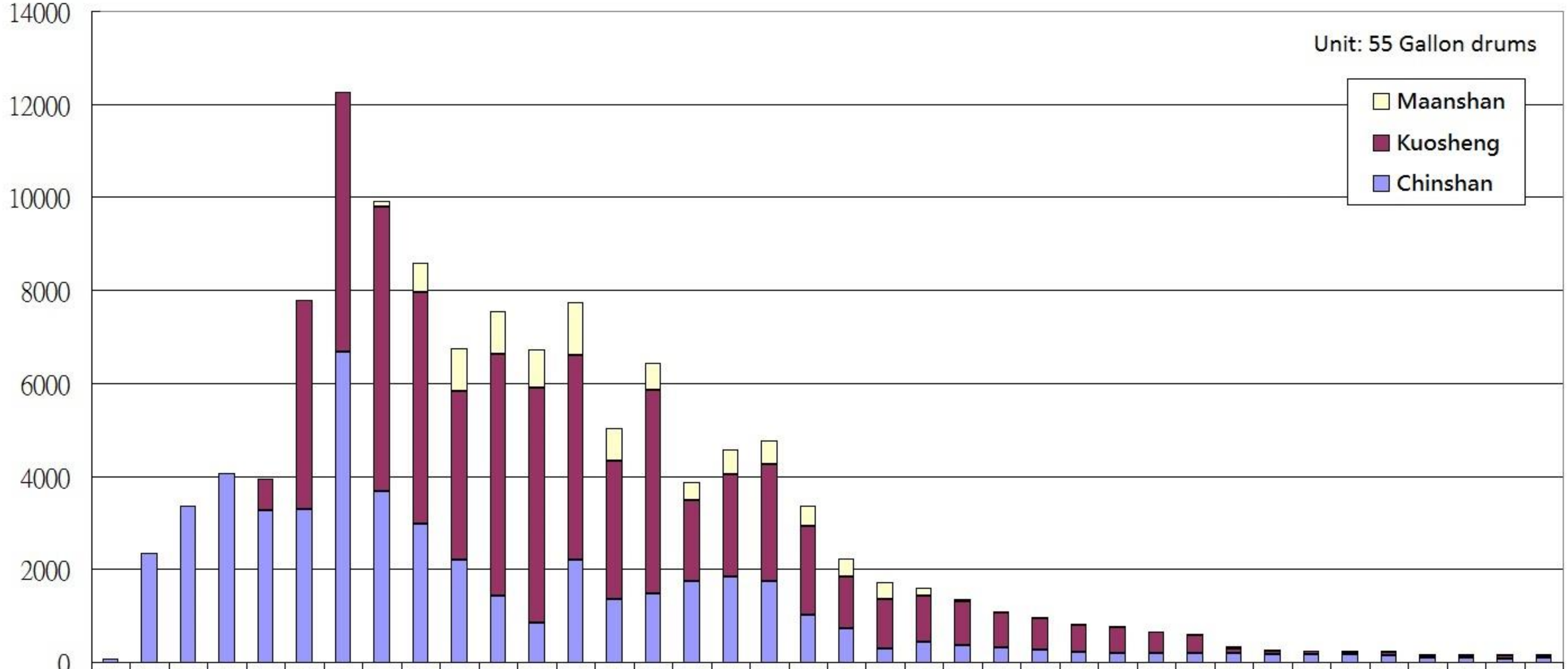
More than 90% of the low level waste (LLW) (or low level radioactive waste, LLRW) generated by volume in this nation came from the three operating NPPs. With the use of the high efficiency solidification technology (HEST) developed by the Institute of Nuclear Energy Research (INER) and as a result of the plant staff's efforts, the annual generation of the solidified LLW from these NPPs drastically dropped from a peak of nearly 12,000 drums (200 liters each) in 1983 to only about 176 drums in 2014, as shown in Figure 6.9. (More information in Subsection 19.9.1.)

6.1.4 Power Supply Sources of the NPPs

The power distribution systems of nuclear power plants (NPPs) in Taiwan are divided into two categories: safety-related and non-safety-related. Non-safety-related power systems provide various auxiliary load power needed for the operation of a nuclear power unit, while safety-related power systems provide power to the reactor protection system (RPS) and emergency cooling systems (ECS) to ensure safe shutdown of the reactor and cooling of the core. Safety-related power systems thus must meet the requirement of the seismic category I and electrical class 1**.**

When a NPP unit is in normal operation, its service power can be provided from the main turbine generator (TG) through the unit auxiliary transformer (UAT). In the meanwhile, in each NPP in Taiwan there are two offsite power sources (either 345kV & 69kV or 345kV & 161kV) for providing the startup power through the respective startup transformers.

In each unit of the 3 operating NPPs in Taiwan, there are two redundant safety-related water-cooled emergency diesel generators (EDGs) of seismic category I design with voltage 4.16kV, while in each unit of the 4th NPP (Lungmen NPP) there are three such EDGs. All these EDGs act as the on-site backup alternating current (AC) power sources to provide power to the RPS and ECS in case of loss of offsite power (LOOP). When a LOOP event occurs, the power supply of safety-related essential systems will be



	1977	1978	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995
Maanshan								138	607	935	919	842	1147	686	586	381	529	500	449
Kuosheng					673	4494	5576	6107	4982	3625	5204	5054	4405	2983	4391	1757	2214	2525	1904
Chinshan	83	2345	3365	4069	3258	3293	6682	3680	2984	2191	1425	838	2194	1349	1463	1736	1833	1731	1010

	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010	2011	2012	2013	2014
Maanshan	388	356	176	34	24	18	17	27	18	19	30	17	12	22	18	15	30	35	35
Kuosheng	1114	1070	983	938	741	685	582	538	446	387	107	62	61	59	68	57	58	66	51
Chinshan	729	292	444	374	316	260	219	200	200	195	190	180	180	170	148	90	90	80	90

Figure 6.9 Annual Number of Drums of Solidified LLW Generated from the Three Operating NPPs in Taiwan (up to the year 2014)

automatically switched to the water-cooled EDGs. The water-cooled EDG can reach its rated speed and voltage within 10 seconds and then the power output circuit breaker will be closed automatically to provide power to the essential bus.

In addition, each NPP in Taiwan has one additional safety-related air-cooled swing EDG (SEDG) of seismic category I design. Furthermore, each of the 3 operating NPPs has another two redundant non-safety grade non-seismically qualified air-cooled gas turbine generators (GTG)(or simply gas turbines, GT) with black-start capability. These air-cooled swing-EDG and GTs can take over the function of onsite backup water-cooled EDGs in supplying power to the essential safety systems in emergency. In case of loss of both offsite power and water-cooled EDGs, which is the so-called “station blackout (SBO)” event, the swing EDG and GTGs can act as the diverse emergency backup power sources. Construction of two non-seismically qualified air-cooled GTGs in Lungmen NPP was planned.

The capacity of the battery system in each unit of the NPPs is originally designed to be capable of providing the essential direct current (DC) power supply for at least 8 hours (h) in case of loss of AC power. After the Fukushima Daiichi accident, the relevant plant procedures have been modified to extend the DC battery duration time from 8 h to 24 h. Furthermore, a variety of additional large scale (4.16kV/1500kW) mobile diesel generator (MDG) power trucks and medium scale (480V/500kW or 480V/200kW) MDGs are also available in each NPP for alternately supplying the essential power in emergency.

In the following, the power supply sources of the Taiwanese NPPs are to be discussed.

6.1.4.1 Offsite Power

In each of the existing NPPs in Taiwan, the offsite power supply systems are connected to two external grids (345 kV and 69 kV grids or 161 kV grids) with several transmission lines each:

- CSNPP: Four 345kV transmission lines and four 69kV transmission lines
- KSNPP: Four 345kV transmission lines and two 69kV transmission lines
- MSNPP: Four 345kV transmission lines and two 161kV transmission lines
- LMNPP: Four 345kV transmission lines and two 161kV transmission lines

(1) <u>Chinshan NPP</u>

In the Chinshan NPP, there are four 345kV offsite power sources with two transmission lines from the Ting-Hu extra high voltage (EHV) substation, one from the Hsichih EHV substation and another one from the neighboring Kuosheng NPP. In addition, there are four 69kV offsite power sources with two transmission lines from the Hsing-Jen secondary substation and another two from Chin-Shan secondary substation. As to be mentioned in more detail in Subsection 6.1.4.3, there are also two gas turbine generators (GTGs) on site to provide emergency backup power source to the 69kV switchyard. Consequently, the 69kV start-up transformer can be available even if one of the offsite power lines or breakers (BKR) trips.

The power source of the non-safety-related power systems (4.16kV Bus #1 and Bus #2) in Chinshan NPP can be selected from the unit auxiliary transformer (UAT) which is connected to the main generator or from the startup transformer connected to the off-site 345kV or 69kV system grid. During normal operation, usually power supply from the main generator through the UAT is selected. When necessary, the power source can be manually switched from the main generator to the offsite power via the 345kV or 69kV startup transformer. In case the reactor scrams or the UAT trips, the power supply for the non-safety-related 4.16kV buses will be switched to the 345kV or 69kV startup transformer automatically.

The power source of the safety-related essential buses (4.16kV Bus #3 and Bus #4) in Chinshan NPP is from the non-safety-related power systems (4.16 kV Bus #1/Bus #2) which is normally powered from the main generator of the nuclear unit and can be manually switched to become being powered from the offsite 345kV system grid or 69kV system grid. If the reactor scrams or the UAT trips, the power source of the safety-related essential 4.16kV buses will be automatically transferred to the offsite power network system through the 345kV startup transformer or 69kV startup transformer, or to the water-cooled EDGs in case the offsite power is lost.

(2) <u>Kuosheng NPP</u>

The offsite power supply for the Kuosheng NPP is provided by two physically independent transmission systems, the 345kV system and 69kV system. The 345kV system has four 345kV transmission lines from the Hsichih substation (2 lines), Hsiehho substation (1 line) and the Chinshan NPP (1 line) in the neighborhood. On the other hand, the 69kV system has two 69kV transmission lines from the Chin Shan/Chung Fu and Wu Lun/Chien Hua substations.

The power for the engineered safety features (ESF) (i.e., essential) buses of both units can be supplied from either the 345kV or 69kV offsite power systems. It is via the off-site 345kV or 69kV start-up transformers to step down the 345kV or 69kV off-site power to the required 4.16kV for the essential buses. Each of the essential buses is also connected to an independent water-cooled EDG to provide power to the emergency cooling systems (ECS) in case of loss of offsite power (LOOP).

(3) <u>Maanshan NPP</u>

The Maanshan NPP also has two offsite power systems: 345kV and 161kV systems. The 345kV system has two buses (Bus-1 and Bus-2) and is connected to the electric grid by four transmission lines with three lines connected to the Dapeng EHV substation and one line to the Meelee EHV substation. The 161kV system has one single bus (161kV Bus) and two loops with one connected to the Fengkang substation and another circuit connected to the Kenting substation and then to the Fengkang substation. The plant has two 345kV start-up transformers and two 161kV start-up transformers (one for Unit 1 and the other for Unit 2 and they can support each other in emergency) to provide the power required for the start-up and shutdown for both units. When the unit is in normal operation, its service power can be provided from the UAT.

(4) <u>Lungmen NPP</u>

Similar to the other 3 NPPs in Taiwan, there are also two offsite power systems in the Lungmen NPP. The 345kV offsite power is provided by four transmission lines from the Shenmei (2 lines) and Longtan (2 lines) substations, while the 161kV offsite power is provided by two transmission lines from the Nangang -Pinglin and Aodee substations.

6.1.4.2 Onsite Backup Power ─ Water-Cooled EDGs

Each nuclear unit of the 4 NPPs in Taiwan has redundant and qualified safety-related, seismic category I, water-cooled 4.16kV EDGs to provide the onsite backup power in case of LOOP. Each unit of the 3 operating NPPs, Chinshan, Kuosheng and Maanshan, has 2 redundant water-cooled EDGs with Kuosheng having one extra water-cooled EDG which is used only to provide emergency power to the high pressure core spray (HPCS) system. On the other hand, the Lungmen NPP is equipped with 3 identical water-cooled EDGs per unit.

These redundant water-cooled EDGs supply the back-up AC power to those electrical loads needed to safely shutdown the reactor and maintain cooling for the reactor core and spent fuel pool (SFP) during the design basis events (DBE). In the event of loss of normal sources of power, for example a LOOP, the EDGs will start and be put into service automatically.

(1) <u>Chinshan NPP</u>

In the Chinshan NPP, each unit is equipped with two independent 4.16kV/3600kW safety-related water-cooled EDGs (EDG A and B) of seismic category I design. The two EDGs belonging to the same unit can backup each other. All 4 EDGs of the plant are located at an elevation of 11.2 meters above the mean sea level (MSL) (which is the plant ground elevation) in the combined structure building, higher than the potential maximum tsunami run-up height of 10.73 m. The two EDGs for Unit 1 are named EDG 1A/1B and those for Unit 2 are called EDG 2A/2B. The starting and control power of the EDGs A and B is supplied by the 125VDC power. Each EDG has one independent air starting system.

The EDG system in each unit is divided into two segregated and separated divisions - DIV I and II, independently providing power to the essential bus corresponding to each division. The downstream emergency loads of these two divisions, including the 4.16kV essential buses and their associated power centers (PC) and motor control centers (MCC), are redundant to each other. Each division has one EDG supplying the emergency power to one essential bus in case of LOOP. The design capacity of each EDG is 100% so that, as long as one EDG is successfully operated, it can provide sufficient power needed to the essential bus.

In case of a LOOP event, the EDG attached to the DIV I or II 4.16kV essential bus will start automatically and provide 4.16kV emergency power to the essential bus. The EDG can reach its rated speed and voltage within 10 seconds after being started and then the power output circuit breaker (BKR) will be closed automatically to provide the essential bus for emergency cooling. The capacity of each EDG division is enough for safe reactor shutdown and cooling. In case of a reactor trip, the normal 4.16kV bus power source will be switched automatically to the 4.16kV essential buses within 10 cycles.

Cooling of the water-cooled EDGs is provided by the combined structure cooling water (CSCW) system with heat transferred to the essential service water (ESW) system and finally to the sea ultimate heat sink (UHS).

Cooling of the EDGs without ESW can be assured by alternatively cooling via CSCW to the SFP and then to the atmosphere via the SFP additional cooling system (SFPACS) cooling tower.

(2) Kuosheng NPP

In each unit of the Kuosheng NPP, there are two 4.16kV/3600kW and one 4.16kV/2200kW water-cooled safety-related EDGs of seismic category I design, located in separate compartments of the EDG building at 12 meters above MSL (the plant ground elevation), higher than the potential maximum tsunami run-up height of 10.28 m.

The EDG system of KSNPP is divided into three divisions - Div. I, II, and III. The essential bus of each division is powered by a corresponding EDG. The Div. I and II are connected to the two 4.16kV/3600 kW EDGs. Either EDG together with its corresponding 4.16 kV essential bus has a capacity enough to ensure reactor safe shutdown and cooling. For the essential bus downstream loads of Div. I and II, these two divisions can backup each other. Div. III is connected to the 4.16kV/2200 kW EDG. This Div. III EDG and its corresponding essential bus provide power only to the high pressure core spray (HPCS) system. All 3 EDGs in a unit can reach the rated speed and voltage within 10 seconds after being started.

In case of a LOOP, the two EDGs belonging to Div. I and II, respectively, in the unit will start automatically and provide the necessary power to safely shutdown the reactor.

(3) Maanshan NPP

In the Maanshan NPP, each unit has two 4.16kV/7000kW water-cooled safety-related emergency diesel generators (EDGs A and B) of seismic category I design situated in the EDG building at an elevation of 15 meters above the MSL (which is the plant ground elevation), higher than the design basis (DB) tsunami height (or the potential maximum tsunami uprising height) of 12.53 meters.

In each unit, there are two safety-related essential buses (Buses A/B-PB-S01) with each bus equipped with one water-cooled EDG to provide the emergency power to the corresponding safety-related 4.16 kV load group.

Upon loss of voltage (LOV) on the safety related buses A/B-PB-S01 (or in case of a LOOP), the connected EDGs will automatically start and, based on the loading sequence, supply power to the ESF equipments to safely shutdown the reactor. The cooling water of these water-cooled EDGs is provided by the nuclear service cooling water (NSCW) system.

(4) Lungmen NPP

In each unit of the Lungmen NPP there are three 4.16kV safety-related buses (Buses A4, B4 and C4) and corresponding downstream 480V PC/MCC. Each of these 3 safety-related buses is connected to a 4.16kV/7500kW water-cooled safety-related emergency diesel generator (EDG A, B or C) of seismic category I design in order to provide the emergency power needed.

The three EDGs (EDGs A, B and C) in a unit are located on 3 corners of the reactor building in separate compartments at an elevation of 12.3 m above MSL (which is about the ground elevation of the plant), higher than the potential maximum tsunami run-up height of 8.57 m.

In case of LOOP, power can be provided from EDG A/B/C to the safety-related 4.16kV Bus A4/B4/C4 and downstream PC/MCC. These EDGs are water-cooled via the cooling chain of the reactor building cooling water (RBCW) and reactor building service water (RBSW) systems. In other words, the water to cool these EDGs is first provided by the safety-related RBCW system and then the heat absorbed by the RBCW is removed by the RBSW system serving as the ultimate heat sink (UHS).

6.1.4.3 Emergency Backup Power ─ Air-Cooled Swing EDG and Gas turbines

On every NPP site in Taiwan, there is one additional 4.16kV safety-related air-cooled EDG of seismic category I design. This air-cooled EDG is also called a swing EDG, since it can provide the emergency AC power to either one of the 2 units on the site like a swing diesel generator (DG). This swing EDG is located at the plant ground elevation. It can be aligned manually to backup either of the two units. In Chinshan, Kuosheng and Maanshan NPPs, the swing EDG is also called the 5th EDG because it is the fifth EDG installed on the site. Similarly, it is also termed the 7th EDG in Lungmen NPP.

Same as the onsite water-cooled EDGs, the design capacity of the swing EDG is 100 % such that it is sufficient to provide power to the essential bus. Besides, the swing EDG is designed for common use for both units, but for only one unit at a time according to the original arrangement. It can provide the full power required for a unit. However, after the Fukushima accident, the related plant operating procedures were modified to make the swing EDG able to provide power for both units under load control if required.

Furthermore, on each of the 3 operating NPP sites, there are 2 non-safety grade non-seismically qualified air-cooled gas turbine generator (GTG) sets (or simply gas turbines or gas turbine generators) with black-start capability. Each gas turbine (GT) is equipped with an appurtenant 4.16kV/1100kW diesel generator (DG) to provide the black-start power to it. The GTGs are located on an elevation higher than the plant ground elevation and can provide power to the safety buses. In all 3 operating NPPs, the AC power supply can be taken over by these two GTGs. Installation of two GTG sets for the Lungmen NPP under construction was planned.

In case of loss of both off-site power and onsite backup water-cooled EDGs simultaneously (SBO), the swing EDG and the two GTGs can be used as the diverse sources to provide emergency backup power. It is worth noting that the swing EDGs and the GTGs are all air-cooled and thus no cooling water is needed.

(1) <u>Chinshan NPP</u>

Besides the 4 water-cooled EDGs on-site (2 for each unit), the Chinshan plant has one additional 4.16kV/4000kW air-cooled safety-related and seismically qualified swing EDG situated in the 5th DG building at the plant ground elevation (i.e., 11.2 meters above MSL). The 5th EDG (i.e., the swing EDG) is used to serve as the backup of the 4 water-cooled EDGs of both units to provide the essential power needed in case the EDGs fail. It can be started manually in the control room to substitute any EDG of either unit and perform the same function as that of the EDG being substituted. In other words, in case of a design basis event (DBE) and failure of the water-cooled EDGs, the swing EDG can provide the required power to ensure safe reactor shutdown and cooling. The 125 VDC power used to start and control the 5th EDG is provided by an independent battery set.

Furthermore, on the Chinshan plant site, there are two 50 MW air-cooled non-safety grade non-seismically qualified GTG sets with black-start capability located at an elevation of 22.4 meters above MSL. The two gas turbines (GTs)(or GTGs) can be used as a supplementary source to provide power to the 69 kV buses through their step-up main transformers. Each GT can provide power to the safety buses and has a capacity of 69.6 MVA, enough to serve as the 69kV offsite power for the two units' safe shutdown. Each GT is equipped with an appurtenant DG with capacity of 1,500 kVA and voltage 4.16kV used to startup the GT. The two startup DGs associated to the two GTs are independent from each other. However, if one of these two DGs is unavailable, the other can be used to support starting the GT with malfunctioned DG via the closure of the tie-breaker. This will not affect the SBO startup capability of the GTs.

The two GTGs are primarily for providing the emergency startup AC power for the station when SBO occurs. They can also provide power during grid peak load period or power shortage due to tripping of a big power unit from the grid. In case of LOOP or SBO, the two GTGs can provide the required AC power by supplying power source to the 69kV switchyard.

Normally, the 4.16kV power needed to start the GT is provided from 69 kV Bus-2 (or Bus-1 if Bus-2 is unavailable) via the main transformer (69kV/13.8kV) and start-up transformer (1500 kVA, 13.8kV/4.16kV) to step down the voltage to 4.16kV. This 4.16kV power is used to power the startup motor to actuate the GTG.

The starting power for the two appurtenant DGs to the GTs is provided by the 24 VDC battery sets. These two DGs, each with capacity of 1500kVA and voltage 4.16kV, are independent from each other. In case of a LOOP or SBO, the 24 VDC battery set of the 4.16kV/1100kW DG appurtenant to the GT can be used to start the DG to provide the 4.16kV AC power to the GT startup motor. This is the power source to start the GT in case of LOOP or SBO and thus provide the GT with a black start (or SBO start) capability. After being started, it takes about 10 minutes for the GT to engage into the distribution system and another 7.5 minutes approximately to reach full load (50 MW). The power to control the GTG is supplied by an independent DC battery set.

Therefore, in case loss of offsite power (LOOP) and loss of water-cooled EDGs occur simultaneously (i.e., SBO) in a Chinshan unit, there are emergency backup power systems, including one air-cooled swing EDG and two air-cooled GTG sets, available to take over the function of onsite backup EDGs in supplying power to the essential safety systems.

(2) Kuosheng NPP

The Kuosheng NPP has one 4.16kV/3910kW air-cooled safety-related swing EDG (the 5th EDG) of seismic category I design installed on site in addition to the 4 redundant water-cooled EDGs and is common to both units as a backup emergency power source. It is situated in a separate building at an elevation of 12 m above MSL (the plant ground elevation) and can be used as a diverse power supply source to substitute any of the Div. I or II EDGs of either unit.

When the Div. I or Div. II EDG of any unit fails, the 5th EDG can be manually put into operation in the control room to substitute the malfunctioned water-cooled EDG to provide enough essential power supply to maintain the unit in safe shutdown condition. The original arrangement of this 5th EDG is to provide power only to one division essential bus of any unit. After Fukushima accident, the corresponding plant operating procedure has been modified to make the 5th EDG able to simultaneously provide power to both units with controlled loads.

Furthermore, the Kuosheng plant has two air-cooled non-safety grade non-seismically qualified GTGs located at an elevation of 22 meters above MSL with the black-start capability and a capacity of 50 MW. Each GTG has an associated 4.16kV/1100kW DG for black-start and is able to provide enough power needed for the load of essential buses as well as non-safety-related buses in both units to maintain the units in safe shutdown condition. It takes about 20 minutes to startup the GTG and to engage to the 69kV system.

The 5th EDG or any one of the GTGs can provide power for the loads on the essential buses, including power for the essential and non-essential DC chargers. Therefore, when any of the Kuosheng NPP twin units loses its off-site power and on-site backup power (Div. I, II, III EDGs), it still has the diverse emergency backup power supply from one air-cooled swing EDG and two air-cooled GTGs to provide the required power supply to ensure safe shutdown and adequate cooling of the reactor.

(3) <u>Maanshan NPP</u>

The Maanshan NPP also has one swing EDG (the 5th EDG) which is a 4.16kV/7000kW safety-related air-cooled DG of seismic category I design. It is situated in the 5th EDG building at plant ground elevation (15 m above MSL).

In addition, the plant has two non-safety grade non-seismically qualified 69.6 MVA, 50 MW air-cooled GTGs at an elevation of 35 meters above MSL. Each GTG set is equipped with one 4.16kV/1100kW black-start DG. If offsite power is available, power for the control of the GTG will be provided by the offsite power. If offsite power is unavailable, the diesel engine of the black-start DG will provide the power for control at the startup stage of the GTG.

In case of loss of both offsite power and onsite back-up power (i.e., SBO), the 5th EDG and the two GTGs can be used as diverse emergency power sources to provide the essential AC power. The 5th EDG can be started manually in control room to substitute any EDG of either unit. There is no need of cooling water for the 5th EDG and the gas turbine generators, since they are all air cooled.

If loss of offsite power and loss of on-site back-up power occur simultaneously, the 5th EDG, the GTGs, or the 3rd power source can provide AC power through safety-related bus to the spent fuel pool (SFP) cooling and emergency make-up water pumps. These pumps can be started by operator when necessary to protect spent fuel from uncover.

(4) <u>Lungmen NPP</u>

In addition to the 6 water-cooled EDGs on the Lungmen NPP site (3 EDGs per unit), there is another 4.16kV/7800 kW air-cooled safety-related EDG of seismic category I design common to both units. This 7th EDG, also called the swing EDG, is situated in the auxiliary fuel building (AFB) at elevation 12.3 meters above MSL (about the plant ground elevation) and can provide emergency power to any safety-related bus of both units.

Each of the two Lungmen NPP units has three safety-related 4.16kV buses (A4/B4/C4) and downstream 480V PC/MCC with each bus connected to one water-cooled EDG: Bus A4 is connected to EDG A, Bus B4 to EDG B, and so forth. In addition, in Lungmen NPP there is another 4.16kV safety-related bus (Bus S4) connected to the swing EDG (i.e., the 7th EDG) and its corresponding downstream 480V PC/MCC. Bus S4 is common to both units.

In case of LOOP, emergency power can be provided from EDGs A/B/C to 4.16kV Buses A4/B4/C4/S4 and 4.16kV downstream PC/MCC. The 7th EDG is the backup for the 6 EDGs of units 1 and 2. If any one of these 6 EDGs fails, the 7th EDG can take over the design function of the malfunctioned EDG. The target EDG to be substituted by the 7th EDG has to be pre-selected. The 7th EDG, in full load operation, can provide all the AC power required by the emergency core cooling systems (ECCSs). In case of loss of both offsite power and onsite backup power (EDG A, B and C), the 7th EDG will be used to provide power required to safely shutdown and adequately cool the reactor. The start and control power of the 7th EDG is 125VDC provided by an independent battery set.

Besides, it was planned to install two 12MVA air-cooled non-safety grade non-seismically designed GTGs in the Lungmen station as another diverse emergency backup power in case of SBO.

6.1.4.4 DC Power and Mobile Diesel Generators

In case of SBO, for the BWR-type Chinshan, Kuosheng and Lungmen NPPs, sufficient reactor water level and appropriate pressure can be maintained by operation of the reactor core isolation cooling (RCIC) system to provide feedwater (FW) to the reactor pressure vessel (RPV) and by actuation of the safety relief valves (SRVs) on the main steam lines to release the steam pressure. The RCIC pump is turbine-driven (i.e., steam-driven).

On the other hand, the coping strategy for SBO by the PWR-type Maanshan NPP is to inject feedwater (FW) into the secondary side (or shell side) of the steam generators (SG) by the turbine-driven auxiliary feedwater pump (TDAFP) and/or diesel engine-driven auxiliary feedwater pump (AFP), and to actuate the power operated relief valves (PORVs) on the main steam lines (MSLs) to relieve the steam pressure.

Both RCIC system and TDAFP rely on the 125 VDC power.

In case of LOOP, without considering the availability of backup power sources, the battery set of the plant can provide the required DC power for at least 8 hours. However, after the Fukushina accident, the DC power supply was enhanced and can now last for at least 24 hours. During this period of time, it is believed that the station’s normal onsite backup power supply (i.e., water-cooled EDGs) and/or the alternative power supplies including the air-cooled swing EDG and GTGs as well as the mobile power supplies can be set up.

In case of SBO along with loss of both the swing EDG and the GTGs, the DC power and the mobile AC diesel generators (DGs) will be the only available power sources left to power the NPP. Based on lessons learned from the Fukushima accident, a variety of large scale (4.16kV/1500kW) mobile diesel generator (MDG) power vehicles and medium scale

(480V/500kW or 480V/200kW) MDGs were added to the NPPs. The MDGs are for power supply to critical systems and for battery recharging. In other words, if all permanently installed AC power sources were unavailable, the NPPs have provisions of DC power and a number of on-site mobile AC diesel generators.

Redundant battery chargers are provided for each independent battery system. As long as the battery charger is available, the battery sets will be available and so are the vital DC and AC power, via the uninterruptible power supply (UPS) system.

In summary, based on the defense-in-depth concept, the power supply systems of the NPPs in Taiwan have several lines of defense as follows to prevent the occurrence of loss of power to the plant essential systems:

- Multiple offsite electric grid transmission lines,
- Multiple water-cooled, safety-related, seismically qualified EDGs for onsite backup power,
- One air-cooled, safety-related, seismically qualified swing EDG for diverse emergency backup power,
- Two air-cooled, non-safety grade, non-seismically qualified GTG sets with black start capability for diverse emergency backup power,
- Batteries, and
- Large scale (4.16kV/1500kW) MDG power vehicles and medium scale (480V/500kW or 480V/200kW) MDGs for power supply to critical systems and battery recharging systems in emergency.

(1) <u>Chinshan NPP</u>

For coping with a SBO scenario, the Chinshan NPP relies on the reactor core isolation cooling (RCIC) system or the high pressure coolant injection (HPCI) system and the operation of the safety relief valves (SRVs) on the main steam lines (MSL). The RCIC and HPCI pumps are both turbine-driven.

The operation of RCIC relies on the 125V DC power. Originally, the battery set can provide DC power for at least 8 hours. However, after the improvement due to lessons learned from Fukushima, it can now last for up to 24 hours. A modified SBO procedure to extend the battery power provision time has been established accordingly. The corresponding time for HPCI is also at least 8 h. Air supply for the SRVs is sufficient for 43.2 h (after a design change has been completed with N_2 bottles connected to fix air supply pipes).

Manual operation of RCIC system without DC power has been successfully tested and included in the palnt operating procedure as an accident management measure.

(2) <u>Kuosheng NPP</u>

The safety-related DC power supply to each of the Kuosheng NPP units consists of five 125V DC power supply systems, namely battery sets A, B, C, D and G. Among these 5 safety-related battery sets, A and C belong to Div. I load group, B and D belong to Div. II load group, and G belongs to Div. III load group. The design feature is to ensure that any system failure does not impact the reactor protection system (RPS) and the essential equipments to perform their intended functions.

To cope with SBO, a Kuosheng NPP unit has to rely on the operation of the RCIC system (whose pump is steam-driven) and the actuation of SRVs on the MSLs. In case of SBO along with loss of the 5th EDG and the GTGs, the reactor water level can be maintained by running the RCIC system. The operation of RCIC relies on 125V DC power. The safety-related batteries for RCIC operation and SRV actuation are designed to last at least 8 hours. However, the capacity of this DC power source can now be extended to 24 hours. In addition, a dedicated MDG with its associated rectifier can supply DC power to RCIC and SRVs while a mobile air compressor can supply service air to SRVs.

Because the 5th EDG or either one of the GTGs can provide power to the loads on essential buses, including power for the essential and non-essential DC chargers, the battery can have a DC power capacity for long term use.

(3) <u>Maanshan NPP</u>

The auxiliary feedwater system (AFS) of Maanshan NPP is equipped with one turbine-driven auxiliary feedwater pump (TDAFP) and one diesel engine-driven AFP in addition to the motor-driven AFPs. Without AC power supply, the motor-driven AFP will fail to operate, while the turbine-driven AFP can be operated.

In case of SBO, the Maanshan NPP has to rely on the removal of heat via the secondary side (or shell side) of the steam generators (SGs) by actuating the PORVs on the MSLs to relieve the steam, and the injection of feedwater (FW) into the SG secondary side by the turbine-driven and/or diesel engine-driven AFPs in the AFS. Operation of the turbine-driven (i.e., steam-driven) AFP relies on the 125 VDC power.

The rated capacity of each of the battery trains A and B is 1650 AH and that of the battery trains C and D is 900 AH. Based on minimum loads needed (i.e., with some non-essential loads being isolated), the designed battery capacity of battery trains A, B, C, and D can continuously provide DC power for at least 8 hours in case of SBO.

Should all DC power be unavailable, the steam-driven pump of the auxiliary feedwater system, together with the PORVs on the MSLs, can be operated manually as an accident management measure.

The lead acid battery set for the gas turbine (GT) black start diesel generator (GTBSDG) consists of 6 batteries with capacity 12VDC/200AH, providing the 24VDC required by the GTBSDG. This battery set is maintained monthly to ensure that the DC batteries are in normal standby condition. Once the GT is started, it can be used in turn to provide the AC power to charge the battery set of GTBSDG used to startup the GTs.

The capacity of the battery for the 5th EDG is 125VDC/220AH. It can provide emergency power for 2 hours. Once the 5th EDG is started, it can in turn provide the AC power to charge the battery used to startup the 5th EDG.

(4) <u>Lungmen NPP</u>

Each unit of the Lungmen NPP has 3 safety-related 4.16kV buses (A4/B4/C4) and their associated downstream 480V PC/MCC with each bus connected to one water-cooled EDG. In addition, the Lungmen NPP has another 4.16kV bus (S4) and its associated downstream

480V PC/MCC, common to both units. Bus S4 is equipped with the swing EDG (the 7th EDG), common to both units. Each of Buses A4, B4 and C4 has 5 power sources (including the corresponding EDG), while Bus S4 has only 3 power sources (including the associated swing EDG). Buses A4, B4, C4 and S4 provide power to their respective downstream 480V PCs which in turn provide power to their corresponding downstream 480V MCCs.

In case of SBO (loss of offsite power and EDGs A/B/C) and loss of the 7th EDG, the station still has the DC power remaining. In this case, reactor water level and pressure can be maintained by running RCIC whose operation relies on the 125V DC power. The DC power supply from batteries in Buses A4, B4 and C4 is designed to be available for at least 8 hours and can provide the control power for SRV actuation. The RCIC turbine governor valve (GV) is mechanically operated and needs no electrical power. Thus the turbine-driven (i.e., steam-driven) RCIC pump can be run by only manually opening the steam inlet valve (E51-MBV-0103) and water discharge valve (E51-MBV-0004). The water source for the RCIC pump to take from is the suppression pool or condensate storage tank (CST). The capacity of DIV 0 125V DC battery (Battery S4) connected to the 7th EDG is 1800 AH. Based on Lungmen NPP's final safety analysis report (FSAR), this capacity can provide DC power for at least 3.5 hours. The battery duration can be prolonged if DC loads are controlled.

6.1.4.5 Enhancement of Power Supply Systems after Fukushima

In response to lessons learned from the Fukushima Daiichi accident, a variety of power supply enhancement measures have been implemented by the TPC for its NPPs.

On each NPP site in Taiwan, there are multiple transmission lines that provide offsite power to normal and emergency plant loads. In the event of loss of normal sources of offsite power (LOOP), each unit of the 3 operating NPPs has two water-cooled 4.16kV EDGs, while each Lungmen unit has 3 water-cooled EDGs, that will start and be put in service automatically. Besides, each Kuosheng NPP units has a third water-cooled EDG dedicated to the high pressure core spray (HPCS) system.

In the case of a LOOP together with a loss of water-cooled EDGs (i.e., SBO), each site of the 3 operating NPPs has an air-cooled 4.16kV EDG (the swing or the 5th EDG) and two air-cooled gas turbine generators (GTGs) that can be used as the diverse emergency power sources. In the case of Lungmen NPP, currently it has only one additional air-cooled EDG (the swing or 7th EDG). Installation of two air-cooled GTGs was planned.

After the Fukushima accident, in order to protect its NPPs against a Fukushima-like compound accident, the TPC headquarters had purchased six 4.16kV/1500kW mobile diesel generator (MDG) power vehicles and twenty-six 480V MDGs. Two of these six 4.16kV/1500kW MDG power trucks are allocated for each of the 3 operating NPPs. TPC also bought several air compressors, nitrogen bottles, etc. Equipped with these mobile equipment/devices including power vehicles, diesel generators (DGs), air compressors, nitrogen bottles, etc., the protection capability against a prolonged SBO accident can thus be significantly enhanced.

In addition, based on the lessons learned from the accident at the Fukushima Daiichi NPP, the following enhancements to the electrical power systems for the operating plants in Taiwan have been implemented:

- The swing EDG can now supply the necessary emergency loads for both units simultaneously.
- The two black-start DGs used to start the two GTGs can now supply the necessary emergency loads for both units simultaneously.
- The power cables and corresponding procedures have been prepared to provide emergency power for the 4.16kV safety-related essential buses of both units from the two black-start 4.16kV DGs associated with the GTGs.
- The one or two 4.16kV/1500kW MDG power vehicles newly provided to each site can supply the necessary emergency loads for both units simultaneously.
- Multiple sets of 480 V portable DGs have been procured for each site to power the emergency 480 V buses.
- The coping time of DC power in response to SBO events can now be extended from the originally designed 8 hours to 24 hours.

(1) Chinshan NPP

To cope with a compound "loss of power" event occurred simultaneously in both units of the Chinshan plant, the plant operating procedures have been modified for guiding the 5th EDG to supply powers to two units simultaneously in emergency.

In addition, the station has prepared one set of large scale 4.16kV/1500kW mobile container type DG power trucks and 12 sets of medium scale DGs, including 4 sets of 480V/200kW DGs and 8 sets of 480V/500 kW MDGs, which are stationed on the roof of seismically qualified buildings located at a higher elevation. The two 4.16kV/1500kW MDGs are able to provide power to the 4.16kV safety-related essential buses, while the 480V/500kW MDGs are to be connected to the battery charger to provide DC power to the 125VDC bus, vital UPS, and 480V loads. The Chinshan plant also prepared many sets of small mobile gasoline/diesel generators, air compressors, combustion water pumps, etc.

The power cables and corresponding procedure have been prepared to provide emergency power for the two units' 4.16kV safety-related essential power buses from the two black-start 4.16kV/1100kW DGs associated with the GTGs.

The power transformer of the Technical Support Center (TSC) building lighting system has been moved from the first floor outdoor to an elevated position in October 2015 and quick connections to the MDGs was added in the procedures.

All EDGs, including 4 onsite backup EDGs and one swing EDG, are located at an elevation about the plant ground level. To enhance the resistance capability against flooding/tsunami, watertight equipment will be installed at doors and openings of the 5th EDG building.

(2) Kuosheng NPP

Five sets of small scale 120V mobile generators and power suppliers are additionally procured to provide DC power to the RCIC system and SRVs as well as other essential instrument control power in case of emergency. One of the 120V mobile generators is stationed at the roof of the control building with its related rectifiers and cables equipped with quick connectors and stored in two boxes in the control room to facilitate the support for RCIC and SRVs in case of emergency. In addition, two sets of mobile diesel-driven air compressors and two sets of high pressure boosters are procured to provide the required compressed air for SRV emergency operation.

In order to increase the robustness of the plant power supply system against the Fukushima-type severe compound accident, the following measures are taken to provide the emergency 4.16kV/480V AC power:

- The station has amended the emergency plan so that, via the plant ultimate response guidelines (URG) procedure 1451, the 5th EDG can be put into service to supply emergency power to essential buses of both units.
- The operating procedure for the EDGs has been amended to make the water-cooled EDGs of one unit able to support the other unit of the same NPP.
- The design change requests (DCRs) to modify the two auxiliary black-start DGs (4.16kV/1100kW/1500kVA) appurtenant to the GTG sets to provide power to the essential buses of both units were already completed.
- The two 480V/200kW DGs at the Technical Support Center (TSC) and Operation Support Center (OSC) have been lined up from the TSC/OSC to the load centers (LCs) 1B3/2B3.
- Four sets of 480V/200kW MDGs have been added to provide power to LCs 1B3/1B4/2B3/2B4 of both units. This can provide additional emergency power supply for the essential 480V loads such as uninterruptible power supply (UPS), hydrogen ignition system (HIS), etc., and the 125VDC battery charger which in turn provides power supply for the 125VDC bus.
- TPC headquarters already procured 6 sets of 4.16kV/1500kW mobile DG power engines. Among them, one is for Kuosheng Station and three for Chinshan and Lungmen. Those MDGs at Chinshan and Lungmen can be used to support Kuosheng, or vice versa.
- The DCR to upgrade the capacities of the safety-related battery sets from the originally designed 8 hours to 24 hours was completed.
- Five sets of small scale 120V mobile generators and power suppliers were additionally procured to provide DC power to RCIC & SRV.
- Two sets of mobile diesel driven air compressors and two sets of high pressure boosters were procured to provide the required compressed air for SRV emergency operation.
- The steam-driven RCIC system can be manually restarted at site if its DC power supply is lost or if the RCIC system trips. The associated operation has been demonstrated to be feasible.
- The RCIC system can also be operated without DC power by manually operating the steam valves. Plant operating procedure for operating the RCIC without DC power has been incorporated into the plant's URG procedure 1451.

As for the enhancement on capability of the plant's power supply resistance against the tsunami/flooding attack with a potential maximum tsunami run-up height of 10.28 meters above MSL, the following measures have been taken by the Kuosheng plant:

- Flood barrier panels are installed at the personnel hatch and equipment hatch of the EDG building to upgrade the waterproofing of this building.

- A waterproof wall of the emergency circulating water (ECW) pump room was added with a height of 11.24 meters reaching the top of ceiling and all penetrations of this room are sealed to prevent the tsunami from passing through the retaining wall and flooding the ECW pump motor area. After the improvement, the protection height of tsunami run-up will be the same as that of the ground level (12 m above mean sea level). It can protect the ECW system and the Div. I, II, III EDGs against the sea water invasion.
- The 5th EDG building is equipped with security doors and the elevation of its vent hole reaches as high as 17 meters.
- The two GTG sets are located at an elevation of 22 m.
- The two 4.16kV black start DGs appurtenant to the two GTG sets are situated at an elevation of 22 m.
- The two newly purchased 4.16kV/1500kW MDG power trucks are stationed at an elevation of 31 m.
- The 480V mobile generators are stored in the warehouse at an elevation of 16 meters.

(3) <u>Maanshan NPP</u>

After Fukushima accident, the Maanshan station implemented several enhancement measures to increase the robustness of the plant power supply systems. The following are some typical ones of this enhancement.

The DCR to upgrade the capacities of the safety-related battery sets from the originally designed 8 hours to 24 hours was completed. Ten 370kVA, 480V MDGs have been procured by the Maanshan station to provide AC power to:

- Battery charger to maintain 125VDC bus power,
- Vital water pumps to supply water to critical equipment/facilities, and
- Important 480V loads such as telephone exchange/TSC/control room in order to maintain the necessary communication, commanding and habitability systems, and other important 480V loads.

The feasibility for the 5th air-cooled EDG to provide emergency AC power to both units simultaneously has been validated. The terminal points for quick connection for this purpose have been set up and the related operating instructions have been included in the plant ultimate response guideline (URG) procedure 1451.

TPC headquarters has procured six 4.16kV/1500kW MDGs, of which two are placed in the Maanshan station. These MDGs can serve as the supporting equipment to provide power to safety-related 4.16kV Buses A/B-PB-S01. Power can then be provided from these buses to the centrifugal charging pump (CCP) for making up water to the reactor coolant system (RCS) and reactor coolant pump (RCP) seal injection and to spent fuel pool (SFP) cooling and emergency make-up water pumps. A corresponding DCR (DCR-M0-4325) has been set up to connect these MDGs with the existing equipments.

The plant has already prepared 24 small mobile generators. They can provide temporary power to submerge pumps, exhaust fans, and radiation monitors to ensure the proper function of those rescue equipments or instruments. Rescue equipments are stored in warehouses at elevation 25 meters above MSL.

In case of SBO, the Maanshan NPP has to rely on the heat removal via the secondary side by actuating the PORVs on the SGs (actually on the MSLs) to relieve steam and operating the steam-driven and/or diesel engine-driven auxiliary feedwater pump (AFP) in the auxiliary feedwater system (AFS) to inject feedwater into the SG secondary side. If all DC power is also lost when SBO occurs, the turbine-driven AFP and the PORVs of the SGs still can be manually operated to make up feedwater into the SGs and to release steam from the SG to the atmosphere. In case the turbine-driven AFP is unavailable, the diesel engine-driven AFP can be operated substitutively to make up water into the SGs. These actions can control the temperature and pressure of the reactor coolant system (RCS) and maintain the core cooling. When all types of AFP are unavailable, diesel engine fire pumps can be operated to makeup water into SG.

The EDG building is of seismic category I design and is located at an elevation of 15 meters above MSL, which is higher than the maximum uprising height of tsunami (12.53 m). For prevention of flooding into other plant buildings, the station has proposed a DCR to evaluate the feasibility of installing flood barrier plate at the ingress/egress gates of the buildings with safety-related systems, so that when the Central Weather Bureau issues warning of typhoon plus heavy rain, or if it is judged that there is possibility of flooding in the site, the flood barrier plates can be installed in advance to prevent water intrusion. The station has also proposed another DCR to improve the water-tight characters for the key fire-proof doors or explosion-proof doors of the buildings with safety-related systems, so as to prevent flood intrusion into key areas or rooms, and thus to prevent faults/failures of the important equipment/facilities.

In addition, the station has procured 3 mobile air compressors to provide working air to individual air-operated valves (AOVs).

(4) <u>Lungmen NPP</u>

In case of SBO and loss of the 7th EDG (i.e., a loss of all AC power onsite and offsite), the reactor water level and pressure can be maintained by running the RCIC. Based on the design bases (DBs), the DC power supply to the RCIC system can last for at least 8 hours. Based on the analysis done in Section 8.3 of the Lungmen NPP's FSAR, the RCIC can run for at least 21 hours during SBO. If some unnecessary DC loads were isolated by controlling the loads, the battery set can supply DC power to RCIC for at least 24 hours. Furthermore, by means of reactor depressurization (through SRVs on SGs) and containment ventilation, more time can be available for the station to recover the AC power.

Additional enhancing measures to provide emergency 4.16kV/480V AC power in case of SBO include:

- Making the 7th EDG able to provide power to Buses A4/B4/C4 of both units simultaneously through Bus S4 in case of emergency.

  To ensure power being able to be provided to the required loads in case of LOOP, plant operating procedure for the operation of the 7th EDG is modified to provide power through Bus S4 to both units simultaneously. The related emergency operating steps have been incorporated into station procedure 1451 "Ultimate Response Guidelines".

- Adding 2 gas turbine generators

  These 2 GTGs can provide AC power to the switchyard 161kV bus and then through transformer to the safety-related 4.16kV Buses A4/B4/C4 and S4 as well as their corresponding downstream 480VAC PC/MCC.

- Preparing five 480V/100kW mobile diesel generators.

  The station has prepared five 480V/100kW MDGs. Anyone of these generators can supply AC power to breaker 1R12-MCC-0140A4-9B. Thus, AC power is available for the battery charger to charge the associated battery set to prolong the battery duration.

The above 3 enhancement measures can ensure the reliable and endurable DC power supply to the RCIC. The emergency operation steps relating to the above arrangements have been incorporated into the plant's URG procedure 1451.

- Connecting the black start DGs of the GTGs (under construction) to the switchyard 4.16kV auxiliary electrical panel to provide power to safety related 4.16kV bus and safety 480V PC/MCC.
- Making the water-cooled EDGs of both units able to support each other in case of emergency so that the operable EDGs of a unit can provide support to the other unit with malfunctioned EDGs. Associate procedure is being prepared.
- Adding 480VAC MDGs to provide power to 480VAC PC/MCC (both Q and non-Q).
- Operating the RCIC pump manually

  The RCIC turbine governor valve (GV) is mechanically operated and no electrical power is needed. The turbine-driven RCIC pump can be run by only manually open the related steam inlet valve and water discharge valve. These two valves are located outside the containment and it will have no problem to operate them. The operating steps have been demonstrated to be feasible and have been incorporated into the station procedure.

- Adding EDG (11.4kV/1250kW) of the security system

  This EDG will provide power to the 4.16kV bus and downstream 480VAC PC/MCC through the switchyard auxiliary electrical panel.

6.1.5 Status of Nuclear Installations in Taiwan

As shown in Table 6.2, the 40-years operation lifetimes of the Chinshan Units 1 and 2 will be due on December 5, 2018 and July 15, 2019, respectively. According to the regulations, the life-extension application of a NPP must be submitted 5 ~15 years before the expiration of the operating license (OL) to the regulatory body by the licensee. Thus, The TPC submitted the life-extension application of Chinshan NPP in July 2009 to the AEC for review. A review of such kind is the first time in Taiwan and will normally take 24 ~ 26 months. During the reviewing period, because of the happening of the Fukushima accident in March 2011 and the implementation of the stretch power uprate program of the Chinshan plant in 2012 (Subsection 6.1.2), the TPC requested to postpone the review. The review was re-started in August 2014 and it was estimated to need about 2 more years.

However, due to the Fukushima Daiichi accident in March 2011, Taiwanese Government's national energy policy has been revised in November 2011. It mentioned that there will be no life extension beyond 40 years for all nuclear units, if there needs no power rationing, reasonable power price can be maintained, and pledges to reduce carbon emissions can be fulfilled.

The Government of Taiwan announced its new policy towards the Lungmen NPP under construction on April 28, 2014 that this NPP will be mothballed after safety examination. The TPC submitted the Mothball Plan of the Lungmen NPP to the AEC on September 1, 2014 which was approved on January 29, 2015. The Lungmen NPP was formally put into mothball on July 1, 2015 for a period of 3 years. Although the application of initial fuel loading (IFL) authorization for Unit 1 of Lungmen NPP has been submitted to AEC by TPC on Dec. 31, 2013, the TPC is required to resubmit the IFL application once the ROC's government decides that the Lungmen NPP program is to be restarted.

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 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 attitude toward nuclear safety.

The nuclear safety culture program in the TPC started in 1988. The implementation of this program was divided into four phases: the Learning Period (1988 ~ 1992), Cultivating Period (1993 ~ 1997), Enhancing Period (1998~2010), and Advanced Period (since 2010).

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:

(1) 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.

(2) 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.

During the Advanced Period, the TPC's president asked in 2010 all its employees as well as organizations to obey the policy that "nuclear safety should be in first priority, without any slack" and to insist on the highest guiding principle of "without safety, there would be no nuclear energy." According to the Fukushima lesson learned, a safety culture advanced program has been developed by TPC since June 2011 to foster a high level of nuclear safety for protecting people's life and property safety. This SC advanced program includes 4 major areas: management effectiveness, contractor management, risk management, and personnel performance. More detailed information about the TPC's safety culture can be found in Subsections 10.1.3 and 12.6.1.

As an illustration of the effects of implementing the nuclear SC, Table 6.4 shows some typical examples of the TPC's nuclear safety-related performance indicators before and after the implementation of the nuclear safety culture program.

Table 6.4 TPC's Safety-Related Performance Indicators during 1991~ 2014 *

Year	No. of scrams per unit	No. of abnormal events per unit	No. of violations more serious than 4th 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
Nuclear SC advanced period began in 2010				
2010	0	0.83	0.33	92.38
2011	0	1	0.67	93.46
2012	0.33	1.67	1	89.46
2013	0.67	0.83	0.5	92.41
2014	0	1.33	0.33	94.07

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

** The 1st degree of violation is the most serious, while the 5th degree is the least one.

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 notification — reporting by phone within 15 minutes after knowing an accident occurred and submitting a written report within one hour,
- Radioactive waste production reports — monthly and annually (referring to Subsection 19.3),
- Performance indicators report resulting from the reactor oversight process — quarterly,
- Radioactive effluent release report — quarterly and annually,
- 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 construction license (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.

More information about the licensing for CL and OL of a NPP can be found in Subsections 10.1.4.1 and 19.1.

6.2.3.2 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. Regulatory inspections of the NPP during the construction stage or operating period include the resident inspections, regular inspections, outage inspections, expert team inspections, and special team inspections as well as the unannounced inspections.

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 performed half a year 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 via the outage inspections 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 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.

More information about regulatory inspection can be found in Subsections 10.1.4.2 and 14.2.1.

6.2.3.3 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 to evaluate the safety performance of operating NPPs. This compact ROP system, adapted from the reactor oversight process (ROP) developed by USNRC, was used for inspecting and assessing the plant performance to ensure the plant's operational safety. At the beginning, it was limited to the strategic performance area of "reactor safety" with three cornerstones only (instead of three strategic performance areas with seven cornerstones in the NRC's ROP) and was thus termed the "compact" ROP. The three cornerstones used in the early stage were the initiating events, mitigating systems, and barrier integrity 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 findings resulting from the AEC's inspections, for the ROP evaluations. The assessment results of this compact ROP will be posted and updated quarterly on the AEC's public English website at "http://www.aec.gov.tw" under the directory of Nuclear Reactor Safety/Reactor Oversight Program.

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 the AEC's ROP system, there are now seven cornerstones, including the emergency preparedness, public radiation safety, occupational radiation safety, and physical protection plus 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.

The AEC's ROP system now consists of two categories of assessment indicators: the performance indicators (PIs) and the inspection indicators. The performance indicators which are 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 PI 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 (SDP) 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.

In the AEC's ROP assessment process, the PIs provided quarterly by the NPPs are combined with the inspection findings of the AEC inspectors. The quarterly assessment result for each indicator is transformed into a "color" similar to the traffic light, according to the degree of safety concerns (or safety significance) of it. A "green" color means "no safety concern," while white, yellow and red colors stand for "minor," "median" and "significant" safety concerns, respectively. An overall assessment of the plant is then based on evaluations of all these indicators. As a follow-up action, the AEC will take regulatory measures based on the "color."

Currently the AEC's ROP system consists of 15 performance indicators (PIs) and 4 inspection indicators. Tables 6.5 and 6.6 give examples of the ROP performance and inspection indicators.

Table 6.5 Performance Indicators of the Operating NPPs in Taiwan

ROP – Performance Indicators (PIs)

Indicators / NPP Units		Chinshan		Kuosheng		Maanshan	
		1	2	1	2	1	2
Initiating Events	Unplanned scrams (automatic or manual) per 7000 critical hours	🟢	🟢	🟢	🟢	🟢	🟢
	Unplanned scrams with loss of normal heat removal	🟢	🟢	🟢	🟢	🟢	🟢
	Unplanned power changes (> 20% rated power) per 7000 critical hours	🟢	🟢	🟢	🟢	🟢	🟢
Mitigating Systems	High pressure cooling system (HPCI/HPCS) unavailability rate	🟢	🟢	🟢	🟢	🟢	🟢
	RCIC unavailability rate (for BWR) or AFW unavailability rate (for PWR)	🟢	🟢	🟢	🟢	🟢	🟢
	RHR unavailability rate	🟢	🟢	🟢	🟢	🟢	🟢
	EDG unavailability rate	🟢	🟢	🟢	🟢	🟢	🟢
	Malfunction of safety systems	🟢	🟢	🟢	🟢	🟢	🟢
Barrier Integrity	Specific activity of RCS	🟢	🟢	🟢	🟢	🟢	🟢
	Leakage rate of RCS	🟢	🟢	🟢	🟢	🟢	🟢
Emergency Preparedness	Performance of drill/exercise	🟢		🟢		🟢	
	Drill participation of emergency response organizations	🟢		🟢		🟢	
	Reliability of alert & notification systems	🟢		🟢		🟢	
Radiation Protection	Occupational exposure control effectiveness	🟢	🟢	🟢	🟢	🟢	🟢
	Public exposure control effectiveness	🟢	🟢	🟢	🟢	🟢	🟢

Note 1: 🟢 — No safety concern, ⚪ — Minor safety concern, 🟡 — Medium safety concern, 🔴 — Significant safety concern.

2: HPCI — high pressure coolant injection (system) HPCS — high pressure core spray (system) RCIC — reactor core isolation cooling (system) AFW — auxiliary feedwater (system) RHR — residual heat removal (system) EDG — emergency diesel generator RCS — reactor coolant system
Page last reviewed/updated on June 3, 2015, at 14:34

Table 6.6 Inspection Indicators of the Operating NPPs in Taiwan

ROP – Inspection Indicators							
Indicators / NPP Units		Chinshan		Kuosheng		Maanshan	
		1	2	1	2	1	2
Initiating Events	Q1 of 2015	●	●	●	●	●	●
	Q4 of 2014	●	●	●	●	●	●
	Q3 of 2014	●	●	●	●	●	●
	Q2 of 2014	●	●	●	●	●	●
Mitigating Systems	Q1 of 2015	●	●	●	●	●	●
	Q4 of 2014	●	●	●	●	●	●
	Q3 of 2014	●	●	●	●	●	●
	Q2 of 2014	●	●	●	●	●	●
Barrier Integrity	Q1 of 2015	●	●	●	●	●	●
	Q4 of 2014	●	●	●	●	●	●
	Q3 of 2014	●	●	●	●	●	●
	Q2 of 2014	●	●	●	●	●	●
Emergency Preparedness	Q1 of 2015	●		●		●	
	Q4 of 2014	●		●		●	
	Q3 of 2014	●		●		●	
	Q2 of 2014	●		●		●	

Note 1: — No safety concern, — Minor safety concern, — Medium safety concern, — Significant safety concern. 2: Q1 ─ 1st quarter, Q2 ─ 2nd quarter, Q3 ─ 3rd quarter, Q4 ─ 4th quarter.
Page last reviewed/updated on July 15, 2015, at 08:21

AEC has also prepared the inspection procedures since 2006. Information in more detail about this evaluation can be found in Subsection 10.1.4.3 of Article 10 of this report.

Since the implementation of this ROP system, the evaluation results showed the safety performances of all six operating nuclear units were quite good. Up to the end of the year 2014, the performance and inspection indicators were green in color in most of the time except in a few cases. As an example of cases with non-green indicators, the indicator for the unavailability rate of the reactor core isolation cooling (RCIC) system of the Chinshan Unit 1 was assigned white in color during the period from the 4th quarter of 2005 to the 4th quarter of 2007. Based on this finding, the AEC required the TPC to take actions to troubleshoot the related SSCs, including areas of maintenance, surveillance, monitoring and so forth as well as root cause analysis. Through the implementation of a series of actions, the TPC finally restored the intended function of RCIC. Since then, the indicator about the RCIC of Chinshan NPP showed green in color.

As another case, the quarterly inspection of Maashan NPP conducted in June 2013 identified the unavailability of the 161 kV offsite power supply of Unit 2. After a risk evaluation by using the SDP process, this finding was classified as “white”, meaning slight deviation from safety. AEC responded with more frequent inspection to this “white’ finding, according to the ROP system until the problem was solved.

A third example of plant safety improvement owing to the ROP in monitoring plant performance is as follows: On March 23, 2011, the AEC’s resident inspector conducted a flood protection inspection in the Maanshan NPP and identified two holes in the nuclear service cooling water (NSCW) pump house. Since flooding through these holes could have safety impact on the ability of all NSCW pumps in performing their design functions of accident mitigation, a Level IV (or 4th degree) violation was issued based on this finding. The reason for this violation was determined as not complying with the requirement of the updated FSAR. Subsequently, the AEC held meetings with the licensee and required Maanshan NPP to take corrective measures. Now these two holes have been sealed according to their design requirements.

6.2.3.4 Integrated Safety Assessment (Periodic Safety Review)

After the operating license (OL) was granted and the plant started its commercial operation, the licensee is required by the Regulations to conduct an integrated safety assessment (ISA) of the 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 comprehensive ISA 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.

The purpose of the AEC's 10-year ISA (or PSR) requirement is to ask the license holder to conduct a re-assessment consistent with the present day state of art knowledge, analytical methods, and equipments (e.g. new seismic methodology and digital seismometer), and to identify potential aging problems. Some examples of important improvements resulting from the 10-year PSR are as follows:

- Each TPC's operating NPP unit has installed an automatic seismic trip system (ASTS) in order to strengthen the ability to safely shut down the reactor in case of a strong earthquake and thus enhance the public confidence in the operation of NPPs. The ASTS is designed to trip the reactor automatically under an earthquake stronger than OBE. To comply with the existing control logic of the reactor protection system (RPS) of each unit, there are 3 or 4 groups of independent channels of seismic sensors in the ASTS.
- System identification program of seismic safety re-evaluation has been established in Kuosheng NPP, and will be installed in Maanshan and Chinshan NPPs.

  The seismic instrumentation of system identification program of seismic safety re-evaluation was first installed in 2006 for Kuosheng Unit 1. In 2015, more seismic sensors were further installed for Kuosheng Unit 1 and also for Unit 2, based on the results of Post-Fukushima Safety Enhancement Review.

Maanshan NPP had already installed the seismic instrumentation of the system identification program at the refueling outage for both Units 1 and 2.

Chinshan NPP upgraded its seismometers in 2005 and added another two sets later in an independent building, while Kuosheng and Maanshan NPPs upgraded their seismometers in 2006 and 2008, respectively.

- Most cables within the drywell of the Chinshan NPP suffering from high temperature have been replaced after 1994 by cables which can endure relatively high temperature. The cables of motor operated valves (MOV) in the upper area of the drywell in Kuosheng NPP have been replaced by cables able to endure higher temperature.
- Each TPC's NPP site has now two offsite power sources (either "345kV and 69kV" or "345kV and 161kV") for startup. Originally there was only one 345kV startup transformer in each site. Due to the aging failure incident in Chinshan NPP in 2007, the TPC decided to add one more backup startup transformer in each site.
- After the third 10-year ISA, the Chinshan NPP installed a seismic monitoring system in accordance with the AEC's review results of the ISA:
  - Both Units 1 and 2 have installed the seismic monitoring systems which conformed to the requirements of RG 1.12 Rev.2:"Nuclear Power Plant Instrumentation for Earthquakes."
  - The actuation of weak-motion seismometer (WMS) recorders will also actuate the strong-motion seismometer (SMS) recorders now. The actuation set-point was reduced from 0.0025g to 0.0015g.
- After the second 10-year ISA of Kuosheng NPP, the AEC issued the following requirements to the TPC regarding to the operational experience feedback (OEF) about plant aging:
  - The TPC should submit an evaluation report for the plant aging management strategy of Kuosheng NPP to AEC with the contents consisting of at least: (1) identifying the SSCs important to operational safety and license renewal by integrated assessment and screening; (2) establishing the aging management system structure and plan including aging mechanism, aging management, record documentation, method for aging trend analysis, etc. The TPC submitted the evaluation report in December 2001 and was approved by AEC.
  - The TPC should intensively study the impact of overlay welding on the overall stress of the piping, continuously collect information about test and maintenance technology of the recirculation piping, and submit a report for the aging management of the recirculation piping. The TPC has submitted the report and was approved by AEC.
  - The TPC should improve the seismic monitoring system of Kuosheng NPP. This measure is currently in progress.
- After the second 10-year ISA of Maanshan NPP, the AEC issued the following

requirement to the TPC:

- The TPC should improve the seismic monitoring system of Maanshan NPP. This measure is currently in progress.

- After the third 10-year ISA, the Maanshan NPP Made the following improvement in accordance with review comments of the AEC's ISA review report (NRD-SER-101-11):

  - The seismic margin assessment (SMA) of Maanshan NPP was conducted in August 2011 by a consultant company to evaluate the seismic resistance of the structures, systems and components (SSCs). After this SMA evaluation, 23 electrical and/or mechanical equipments were identified to require enhancement on their seismic resistance. The enhancement measures were completed on June 18, 2014 and the plant can now safely shutdown the reactor even if an earthquake with a magnitude of the review level earthquake (RLE)(≈ 0.72g) happens.

After the Fukushima Daiichi accident, the AEC required that the 10-year ISARs or PSR reports of the operating NPPs had to additionally include a dedicated chapter to discuss "lessons learned from the Fukushima Daiichi nuclear accident."

6.2.3.5 Nuclear Safety Reassessments after Fukushima Accident

After the Japanese Fukushima Daiichi accident in March 2011, the government of this nation initiated a national "Programs for Safety Re-assessment" to re-assess the safety of all existing NPPs in Taiwan on April 19, 2011. Later in August 2011, the nuclear stress tests (ST) program for the NPPs was started.

A final report for the Safety Re-assessment Program, entitled "Comprehensive Safety Reassessment report for NPPs in Taiwan in response to the lessons learned from Fukushima Daiichi accident" (written in Chinese), was published by AEC in August 2012, while the final ST National Report by AEC was issued on May 31,2013. More information about the safety re-assessment and ST programs is given in Subsection 6.4.1.

6.2.3.6 International Peer Reviews

The operation of the NPPs in Taiwan has been continuously reviewed by various international expert groups. For example, the World Association of Nuclear Operators-Tokyo Center (WANO-TC) conducts a peer review of each nuclear power plant every four years. The reports of these review visits are very valuable. However, according to the agreement with WANO-TC, these reports cannot be released to any third parties. Nevertheless, none of the previous review groups uncovered any problems that were deemed serious enough to warrant shut down of any of the reactor units, even temporarily, in Taiwan.

As an example, on June 11, 2015, the WANO-TC organized a team of 25 experts to spend a total of 15-days to visit the Chinshan NPP for carrying out a peer review on full eleven areas, including (1) the organization and administration, (2) operation, (3) maintenance, (4) engineering support, (5) radiation protection, (6) operating experience, (7) chemistry, (8) fire protection, (9) emergency preparedness, (10) training and qualification, and (11) lessons learned from WANO Significant Operating Experience Report (SOER). The evaluation results and recommendations from these reviews were quite beneficial to the

TPC. More information about international peer reviews conducted on Taiwan's operating NPPs by the Institute of Nuclear Power Operations (INPO) and WANO is given in Subsection 10.1.5.1.

In addition, the AEC's draft national report of the ST programs (for 3 operating NPPs only) as well as the licensee's ST reports of the 3 operating NPPs were peer reviewed by experts from the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD/NEA). The OECD/NEA independent peer review team visited Taiwan during the period of March 4 to 20, 2013, including a 2-day site visit to the Kuosheng NPP. Later, the AEC's formal ST National Report and the licensee's ST reports of all 4 existing NPPs were peer reviewed by experts from the European Nuclear Safety Regulators Group of the European Commission (EC/ENSREG). The EC/ENSREG expert team visited Taiwan during the period of September 23 to October 23, 2013, including 2 site visits to Maanshan and Lungmen NPPs.

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 NPP in Taiwan during the operating period must be approved by the regulatory body in advance before its implementation if it involves any 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 minimal increase in either occurrence of a malfunction or the malfunction consequence of the structure, system and component (SSC) important to safety which was 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 major earthquake with magnitude (M_w) 6.9 on the moment magnitude scale (MMS) (a successor to the Richter scale) struck the Osaka-Kobe area of Japan and resulted in 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) and as a 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, on September 21, 1999 at 01:47 a.m., a devastating major earthquake with magnitude 7.3 (M_w) 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 seismic trip system (ASTS) on strong earthquake is set at the design value of the operating basis earthquake (OBE) which is normally set at a value equal to half of the safe shutdown earthquake (SSE). Therefore, a protection of the nuclear power reactor from large earthquake by the ASTS was 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 originally-installed seismic monitoring 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 be compared 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 Final Safety Analysis Report

For a newly commercialized operating NPP, the first update of its final safety analysis report (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.

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.

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 1990s, 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 TS to the improved TS (ITS).

During the progress of the above-mentioned TS conversion, the AEC agreed to the TPC's proposal to convert the CTS of Chinshan NPP directly to the ITS 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, the standard TSs for both 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 10.2(8) 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. TPC is a member of both the World Association of Nuclear Operators (WANO) and the Institute of Nuclear Power Operations (INPO).

Based on the agreement between TPC and the above mentioned organizations, TPC participated in several mutual cooperation programs every year, including but not limited to the following:

- Sharing operating experience and good practice information among members,
- Participating in-plant, corporate and pre-startup peer reviews,
- Applying technical support missions (TSM) and exchange visits, and

- Gaining professional and technical development by attending workshops, seminars, leadership courses and expert meetings.

On the other hand, the AEC joined the USNRC's Cooperative Severe Accident Research Program (CSARP), Exchange of Technical Information and Cooperation in Nuclear Regulatory and Safety Research Matters, Thermal-Hydraulic Code Applications and Maintenance Research (CAMP), and RAdiation protection code analysis and Maintenance Program (RAMP), as well as the OECD/NEA's Cooperative Program on Decommissioning (CPD) and Component Operational Experience, Degradation and Ageing Program (CODAP). A Memorandum of Understanding (MOU) on Nuclear Cooperation between the AEC and the State Office for Nuclear Safety **(**SUJB**)** of the Czech Republic was also signed.

For thirty years (up to 2014), the nuclear communities in Taiwan and Japan took turns in hosting the annual Taiwan-Japan Nuclear Safety Seminar, which was usually held in November. In these seminars, members from both parties presented papers and discussed topics which were mutually interested in. This seminar served successfully as a forum for discussing nuclear safety issues and exchanging operating experience from both sides, and has proved mutually beneficial to both communities.

After the Fukushima Daiichi accident, countries around the world involved in nuclear power took boosted nuclear safety measures and emphasized the necessity of international cooperation and information sharing. A Memorandum between the Association of East Asian Relations of Taiwan and the Interchange Association of Japan for the Mutual Cooperation in the Field of Nuclear and Radiation Safety Regulation in the Peaceful Use of Nuclear Energy of Japan was signed on November 20, 2014, marking a step towards nuclear energy information exchange between these two countries. Through the first AEC-NRA Nuclear Regulatory Information Exchange Meeting held in July, 2015 in Japan, the AEC and the Nuclear Regulation Authority (NRA) communicated face to face, shared the nuclear regulatory experience and discussed the items and issues to be cooperated on in the future.

"Agreement for Cooperation between the Government of the Republic of China and the Government of the United States of America Concerning Civil Uses of Atomic Energy" has always been an important foundation of mutual nuclear energy cooperation and exchanges in this nation with the US. The termination of validity of this agreement was on June 22, 2014. After years of positive communicating and negotiating for contents of a new agreement, a new "Agreement for Cooperation between the American Institute in Taiwan (AIT) and the Taipei Economic and Cultural Representative Office in the United States (TECRO) Concerning Peaceful Uses of Nuclear Energy" was signed on December 20, 2013 in Washington D. C., United States, by authorized representatives from both sides and entered into force on June 22, 2014.

The TECRO-AIT Joint Standing Committee meeting on civil nuclear cooperation has been held in turn in Taiwan and USA annually since 1984. There are 4 groups including: (1) reactor regulation and regulatory research, (2) waste management and environmental restoration, (3) nuclear science, technology and safeguards, and (4) emergency management. This annual meeting gave a good opportunity for Taiwan to exchange the experience on nuclear regulations and operations with those of the USA.

6.3.5.2 Seismic Study

Earthquake is one of the most important safety concerns in the design of a NPP, 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 NPP 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 including 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 a 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 (PRA) 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 NPPs. 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.

More information about the application of PRA in Taiwan can be found in Subsections 10.1.4.3 and 14.1.2(4).

6.3.7 Corrective Action Program

In each of the 3 operating NPPs of the TPC 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, and analysis, as well as resources integration. The purpose of the integrated CAP system is to enhance or improve the analyses of root cause, common-cause, and problem trends, and the evaluation of the effectiveness of the corrective actions and the plant health indicators. In the Maanshan NPP, the CAP is also named the excellent management system.

Each operating NPP regularly held a review meeting of the CAP to review the deteriorative trend of the plant’s SSCs and identify the plant system weakness. In the meantime, the NPP will report to the nuclear managers of superior levels via the “safety culture implementation meeting” held quarterly by the Department of Nuclear Generation (DONG) in the TPC Headquarters to review the weakness of the nuclear systems. The DONG then statistically documents the CAPs related to human behavior and management in the past 18 months, and prioritizes these CAPs according to the degree of severity in order to supervise the operation of the plant as well as to find out and correct the weakness in time.

More information about CAP is given in Subsection 10.2(6).

6.3.8 Research and Development Programs in Nuclear Safety

The Institute of Nuclear Energy Research (INER) has been established for 47 years (up to the year 2015) and is the sole national research and development (R&D) institute in the field of civil applications of the nuclear energy in Taiwan. The major tasks of INER are to support the nuclear regulation, apply nuclear safety technology in nuclear power generation, and develop the decommissioning and radwaste treatment technologies.

The main nuclear R&D areas of the INER in recent years included the evaluation of license renewal for a NPP, the medium and small scale power uprates study, level 2 probabilistic risk assessments (PRA) of the operating NPPs, source term evaluation, seismic risk re-assessment of a NPP, high efficient solidification technology (HEST) study for the low level waste (LLW), nuclear facility decommissioning and radioactive waste management, radiobiological medicines, 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 there is a request.

In the year of 2014, the main R&D projects of INER in the area of nuclear safety included: (1) the development of key technology for improving safety in nuclear power plant management and operation, (2) the development and application of radwaste management technologies for nuclear power system lifecycle, and (3) cleanup of retired nuclear facilities by law.

The project of development of key technology for improving safety in nuclear power plant management and operation consisted of the following subtasks:

- Research in Safety Maintenance of NPP,
- Severe Accident and Compound Disaster Prevention, and
- Investigation for the Radiation Protection and the Emergency Response for the Nuclear Accident.

The project of development and application of radwaste management technologies for nuclear power system lifecycle in INER included the following subtasks:

- Study of Volume-Reduction Technologies for Decommissioning Radwaste,
- Study of the Treatment Technologies for Special Liquid Radwaste, and
- Development and Application of Radwaste Final Disposal Technologies.

The subtasks included in the INER's project of cleanup of retired nuclear facilities by law were as follows:

- Cleanup and Improvement of the Retired Nuclear Reactor and Its Affiliated Facilities, and
- Reduction Treatment and Safety Storage of Radwaste.

In addition to the R&D programs in INER, the AEC together with the Ministry of Science and Technology (MOST), which was previously named "National Science Council," provided a mutual fund annually to assist domestic universities and research institutes in conducting studies on the policy, application and fundamental research of the nuclear science and technology. The major research topics in recent years included technology development on nuclear safety, radioactive materials safety, radiation protection, radioactive medicine, and manpower cultivation as well as communication on risk assessment.

Nuclear power and safety technology was one of the major areas for the National Science Technology Project on Energy (NSTPE), phase-I, from 2009 to 2013 to promote nuclear power in Taiwan. INER and universities, such as National Tsing Hua University (NTHU), were working together on nuclear safety, performance enhancement of existing nuclear power plants, next generation nuclear power plants, waste disposal technology, and nuclear communication. After the Fukushima Dai-ichi accident in Japan in 2011, the focus of the project was shifted to mainly nuclear safety research. Due to the antinuclear movement in Taiwan after the Fukushima Dai-ichi accident, nuclear safety and nuclear backend research was excluded in the NSTPE, phase-II, and part of the budget and sponsorship was moved to AEC with NTHU being the primary organization conducting the research. This project is of significant importance for the nuclear safety regulation as well as for the incubation of talents in nuclear safety. Moreover, MOST also sponsors projects on nuclear power and radiation applications within the energy program.

6.4 Enhancement Actions after Lessons Learned from Fukushima

6.4.1 Programs for Safety Reassessments and Stress Tests for NPPs in Taiwan

After the Japanese Fukushima Daiichi nuclear accident caused by the Great Tohoku earthquake (also called the Great East Japan earthquake) of magnitude 9.0 on March 11, 2011 and the ensuing tsunami, the government of Taiwan initiated a national "Programs for Safety Re-assessment" to re-assess the safety of all existing NPPs in Taiwan on April 19, 2011.

The Safety Re-assessment Program which required the TPC to reassess the safety of its NPPs comprised of two parts: (1) Nuclear Safety Assurance and (2) Radiation Protection and Emergency Response & Preparedness. The reassessments were implemented in two stages: near-term (by June 2011) (the 1st-stage) and mid-term (by December 2011) (the 2nd-stage).

In the part of Nuclear Safety Assurance, the license holder was required to reassess the safety of its NPPs in at least the following areas:

(1) Prolonged station blackout,

(2) Protection against tsunami hazards,

(3) Spent fuel pool (SFP) cooling,

(4) Hydrogen detection and explosion prevention,

(5) Severe accident management,

(6) Prevention against seismic hazards,

(7) Infrastructure resilience, and

(8) Safety culture.

After reviewing the TPC's first-stage and second-stage reassessment submittals, the corresponding AEC's near-term and mid-term safety evaluation reports (SER) (written in Chinese) were approved by the Executive Yuan in October 2011 and August 2012, respectively. The AEC's mid-term SER entitled "Comprehensive Safety Reassessment Report for NPPs in Taiwan in Response to the Lessons Learned from Fukushima Daiichi Accident" was the final report covered the reassessment results of both stages for "Programs for Safety Re-assessment" and is available on the AEC's website.

Two issues related to the current licensing basis (CLB) of the NPPs were identified in the review of TPC's near-term reassessment reports: (1) the elevation measurement did not comply with FSAR in Chinshan plant; (2) the design for tsunami protection was not adequate at the emergency circulating water (ECW) pump room in Kuosheng. That these issues were resolved was later confirmed upon site inspection in June 2011.

During the re-assessment period, the European Union (EU) “stress tests” (ST) specifications were issued by the European Nuclear Safety Regulators Group (ENSREG) on May 25, 2011 and endorsed by the European Commission (EC). Therefore, to well utilize the EU experiences, the nuclear stress tests for the NPPs in Taiwan were conducted on August 5, 2011, subsequent to the Safety Re-assessment Program. The Taiwanese ST program was to conduct a comprehensive and transparent risk and safety assessment for each NPP, based on the “stress tests” specifications issued by ENSREG.

After reviewing the licensee’s ST reports of NPPs, a draft national report of the ST (written in English) was completed by the AEC in September 2012 and then the formal ST national report entitled “Taiwan Stress Test National Report for NPPs” was issued by the AEC on May 31, 2013.

The AEC’s draft national ST report as well as the licensee’s ST reports of 3 operating NPPs were peer reviewed by experts of an independent peer review team indentified by OECD/NEA. The OECD/NEA independent peer review team visited Taiwan during the period of March 4 to 15, 2013, including a 2-day site visit to the Kuosheng NPP. Furthermore, the AEC’s formal ST National Report and the licensee’s ST reports of all existing NPPs were peer reviewed by the EC/ENSREG expert team which visited Taiwan during the period of September 23 to October 3, 2013, including 2 site visits to Maanshan and Lungmen NPPs.

As a result of the safety re-assessment and stress tests programs as well as the OECD/NEA and EC/ENSREG peer review recommendations together with the lessons learned in light of the Fukushima Daiichi accident, various areas which need enhancement to improve the robustness of the NPPs in Taiwan were identified. Thus a variety of enhancement actions by both the regulatory body (AEC) and the licensee (TPC) have been planned or implemented. In the following, major efforts and enhancement actions related to nuclear power safety improvement in Taiwan are addressed. Table 6.7 lists the milestones of major activities in the Safety Re-assessment and Stress Tests Programs in Taiwan. More information about the safety re-assessment and stress test programs can be found in Subsections 6.2.3.5, 10.1.5.2, 14.3.1 and 14.3.2.

Table 6.7 Milestones in Nuclear Safety Re-assessment and Stress Tests for Taiwanese NPPs after the Fukushima Daiichi Accident

Date	Milestone
2011/03/11	Great Tohoku earthquake of magnitude 9.0 (M_w) (or Great East Japan earthquake) happened which, together with the ensuing tsunami, caused the Fukushima Daiichi nuclear accident.
2011/04/19	“Programs for Safety Re-assessment” approved by the Executive Yuan.
2011/04 ~ 05	Preliminary reports of safety re-assessment for 4 existing NPPs in Taiwan (written in Chinese) submitted by TPC to AEC.

2011/05/31	AEC's preliminary safety evaluation report (PSER) (written in Chinese) approved by the Executive Yuan.
2011/06 ~ 10	PSER reviewed by the Executive Yuan's Expert Review Team (ERT).
2011/06, 2011/10, and 2012/01	Walkdown inspections for 3 operating NPPs by AEC.
2011/08/05	Taiwanese stress tests for the 4 existing NPPs started.
2011/10/07	AEC's near-term safety evaluation report "The Near-Term Overall Safety Assessment Report for NPPs in Taiwan in Response to the Lessons Learned from the Fukushima Daiichi Accident" (written in Chinese), with the requirements of EU stress tests specifications included, approved by the Executive Yuan.
2011/11/01	AEC invited OECD/NEA to perform an independent peer review of the European Union (EU) Stress Tests for the 3 operating NPPs in Taiwan.
2011/11 ~ 2012/01	AEC conducted five regulatory meetings on stress test progress with TPC
2012/02	Draft version of "Comprehensive Safety Reassessment Report for NPPs in Taiwan in Response to the Lessons Learned from Fukushima Daiichi Accident" (in Chinese) completed
2012/03	Utility's stress test (ST) reports for the three operating NPPs (Chinshan, Kuosheng and Maanshan) (written in Chinese) submitted by TPC to AEC.
2012/04	Utility's stress test report for the newly built NPP (Lungmen, the 4th one) (written in Chinese) submitted by TPC to AEC.
2012/08	AEC's National Report for safety re-assessment: "Comprehensive Safety Reassessment Report for NPPs in Taiwan in Response to the Lessons Learned from Fukushima Daiichi Accident" (written in Chinese) approved by Executive Yuan.
2012/09	Draft ST National Report for the three operating NPPs as well as the newly built NPP completed by the AEC (in English).
2012/11/05	AEC issued the 1st batch regulatory orders based on the safety re-assessment and EU stress test.
2013/01	Final version of ST National Report for three operating NPPs (in English) completed
2013/02	Stress Tests National Report and Utility's Stress Tests Reports for the three operating NPPs (in English) provided to the independent peer review team organized by OECD/NEA.
2013/03/04 ~ 03/15	OECD/NEA independent peer review team visited Taiwan for reviewing the Stress Tests National Report and the ST reports of the 3 operating NPPs (including a 2-day site visit to the Kuosheng NPP).

2013/03/15	Joint press conference by AEC and the OECD/NEA peer review team
2013/04/23	Final report of OECD/NEA independent peer review completed.
2013/04/30	AEC invited EC to perform an independent peer review of the Stress Tests for the 4 existing NPPs in Taiwan.
2013/05/28	Final ST National Report for the 4 existing NPPs: “Taiwan Stress Test National Report for Nuclear Power Plants” published by AEC (in English)
2013/05/31	Final ST National Report and Utility’s Stress Tests Reports for the 4 NPPs (in English) sent to the EC/ENSREG review team
2013/05/31	“Desktop review” of the Taiwan ST National Report by European Union Peer Review Team (EU-PRT) started
2013/06/06	AEC issued the 2nd batch regulatory orders based on the recommendations and technical observations of OECD/NEA independent peer review
2013/07/09	Preparatory one-day meeting in Brussels, Belgium between AEC and EU-PRT representatives
2013/09/23 ~ 10/03	EU-PRT visited Taiwan to review the Taiwanese ST National Report and the ST reports of the 4 existing NPPs (including 2 site visits to Maanshan and Lungmen NPPs)
2013/11/07	Final version of the EU-PR report and EC summary report submitted by the EC to AEC, and published by EC and AEC on both official websites simultaneously
2013/11/08	AEC published the Chinese version of EC summary report on its website
2013/12/12	AEC published the Chinese version of EU-PR report on its website
2014/03/06	AEC issued the 3rd-batch regulatory orders based on recommendations of the EC/ENSREG peer review team

6.4.2 Regulatory Requirements for Safety Enhancement

Building on the results of the safety re-assessment and stress tests for NPPs in Taiwan and the OECD/NEA and EC/ENSREG peer review recommendations as well as insights from the actions being taken by other countries, the AEC established clear requirements to implement enhancements on nuclear safety. These requirements were embodied in regulatory orders issued by AEC to TPC in 3 batches on November 5, 2012, June 6, 2013 and March 6, 2014, respectively. TPC may propose alternatives subject to AEC approval. Details of the regulatory orders issued are given in Annex 2.

Among these regulatory orders, order Nos.10101, 10103, 10105, 10119, 10121, 101101, 10104(AECDNT), 1013003, 10201, 10202, and HQ-10201 are for seismic protection, while Nos.10102, 10103, 10104, 10105, 10110, 10118, 10202, 10203, and 10204 are for flood/tsunami protection. On the other hand, orders for power supply enhancement are

Nos.10106, 10107, 10108, 10109, 10110, and 10120. Order No.10111 requires TPC to evaluate the installation of a second UHS, No.10115 requires installation of spent fuel pool (SFP) instrumentation, No.10121 is about the improvement of seismic resistance of raw water reservoirs at the NPPs, and No. 10105 is for safety enhancement by seismic, flood and other external events walkdown inspections. In order XX-JLD-10203 (issued on 6/6/2013), AEC requires TPC to systematically assess the combination of events in the areas of flooding and extreme natural events.

In addition, AEC identified the following requirements with respect to "extreme weather" to improve the resilience of the NPPs:

- Confirm the protection of the buildings containing safety related equipment against projectiles of beyond design typhoons with wind speeds exceeding 70.2 m/s.
- Review the appropriateness of the design basis (DB) probable maximum precipitation (PMP) in the context of climate change and envisage associated improvements of the drainage systems. These should be reviewed through the process of periodic safety assessment.
- Clarify the watertight capability of fire-fighting doors and pipeline penetrations seals.
- Perform a systematic evaluation of combinations of extreme natural events.

Other requirements by the AEC as a result of the nuclear safety re-assessment and stress tests programs as well as recommendations from the relevant OECD and EU peer reviews include:

Requirement from the Fuel Cycle and Materials Administration (FCMA):

- Re-evaluate the safety of the independent spent fuel storage installations (ISFSI) in response to the Fukushima-like accidents as well as the beyond design basis earthquake (BDBE). (Request from FCMA)

After Fukushima, by referring to USNRC SRM of SECY-93-087, a review level earthquake (RLE) of 1.67 times safe shutdown earthquake (SSE) is required for the systems which can damage the mitigation systems.

Regarding the SBO and the loss of ultimate heat sink (UHS) specifically, several necessary measures were identified by AEC and accordingly TPC was required to address the following issues:

- Analyze the consequences of accidents involving several units on one site in more detail
- Analyze the shutdown conditions in more detail, in particular during the mid-loop operation for PWRs
- Evaluate in more detail the impact on the SFP of extreme external hazards resulting in SBO and complete loss of UHS.
- Analyze the backup capability of the swing EDG when one unit is in operation while the other unit is in shutdown condition
  - There are 2 EDGs per unit in all operating NPPs. Currently, if one unit is in

normal operation and the other unit is shut down with one EDG under inspection, the swing EDG will be assigned to the latter unit according to a new requirement issued as the operational experience feedback (OEF) after the SBO event of the Japan Higashidori NPP in April 7, 2011. Therefore, the capability of the swing EDG to back up the unit in normal operation is restricted. Envisaged measures to resolve this issue are required. (Refer to Subsection 19.8.3.)

- Perform an evaluation regarding to the installation of an additional air-cooled EDG at the NPPs.
- Perform an evaluation regarding to the establishment of an additional alternate heat sink such as the water fed by groundwater wells.
- Increase robustness of the NPPs in dealing with the primary coolant pump seal LOCA issue (MSNPP) during SBO.
- Enhance the off-site power system and increase its reliability, referring to the emergency management and requirements in Japan
- Strengthen SBO mitigation capability at all NPPs for design basis (DB) and beyond design basis (BDB) external events as recommended in item 4 of the USNRC Near-Term Task Force (NTTF) report
- Conduct a systematic review of non-conventional means, focusing on the availability and appropriate operation of NPP equipment in the relevant circumstances
  - The operability of the non-conventional means should be justified on the basis of technical data. The facilities where the mobile equipment is stored should be evaluated and reinforced if necessary so that they are secure even in the event of general devastation caused by BDB events.
- Justify the URG based on rigorous systematic review and thorough accident analysis.
- Justify the hydrogen and containment pressure control strategies in the URG by taking into account various accident scenarios
- Study the feasibility of adding a mobile heat exchanger (HX) to remove the heat from the containment and/or the reactor

Besides, following improvements and/or upgrading of operating procedures and guides are also required by AEC:

- Integration of the additional equipment and operations into the procedures or guidelines,
- Re-evaluation of the feasibility of emergency operating procedures (EOP) and severe accident management guidelines (SAMG) taking into account the new procedures/guidelines,
- Ensuring that the SAMGs are appropriate for multi-unit events and promoting them from guidelines to procedures if required,
- Strengthening SBO mitigation at operating reactors for DB and BDB external events,
- Improving emergency preparedness (EP) staffing and communications according

to the USNRC NTTF report, and

- Identification of the URGs and their implementation timing, the subsequent measures and monitoring strategy after implementing URG, including the monitoring of radioactive releases, backup ability of present systems and equipment.

Additional requirements by AEC include:

- Estimation of the duration of independent response capability for various severe accident scenarios, beyond design basis accidents (BDBA) and multi-event accidents
  - The required materials and equipment in contracting the off-site supports should also be identified.
- Improvement of main control room (MCR) in case of a BDBA in order to:
  - Ensure the capability of DC power for instrumentation and control (I&C) systems of MCR, TSC, backup TSC, etc.,
  - Improve the seismic level of MCR, TSC, backup TSC, and their equipment, and
  - Assess the adequacy of MCR human arrangement in case of multi-unit events.
- Design of reliable containment hardened vent with filters for BWR units and filtered containment venting systems for PWR units
  - Following recommendation 5.1 of the USNRC NTTF report, AEC required to add a robust and reliable hardened vent and filtered containment system in Chinshan NPP (Mark-I), Kuosheng NPP (Mark-III) and Lungmen NPP (ABWR). The drywell or wetwell hardened vent system of one unit should not be shared with the other units, and be able to be operated either with electrical power or manual operation.
  - Also, installation of filtered containment venting systems (FCVS) is required for Maanshan NPP.
- Improvements and integration of EOPs, SAMGs, EDMGs (extensive damage mitigation guidelines) and URGs
  - Each NPP should prepare the response action before revising the current SAMG. Appropriate training and qualification should also be applicable to the decision-making personnel.
- Enhancement of the SFP instrumentation per USNRC NTTF report.

6.4.3 Licensee's Enhancement Measures

6.4.3.1 General

Per regulatory requirements/orders mentioned in the previous section, the responsive enhancement actions conducted by TPC to address both prevention and mitigation of severe accidents can be divided into three stages: (1) studying the requirements/orders, (2) planning and implementing associated enhancement measures, and (3) completion of the enhancement measure. Most of the actions are still in the first 2 stages, while some are completed. In the following, significant TPC's enhancement measures will be discussed.

After Fukushima, following the AEC requirements (e.g., order XX-JLD-10203), TPC proceeded a reassessment in the following areas related to severe accident management (SAM):

- Capability for loss of AC power (SBO);
- Cooling of SFP;
- Capability of heat removal and UHS;
- Re-evaluation of EOPs;
- Implementing URGs;
- Support between different units;
- Mitigating beyond design basis accidents (BDBAs); and
- Preparedness and backup equipment.

In each of these areas, TPC has thus implemented various measures to address both prevention and mitigation of severe accidents including such measures as addition of alternative mobile diesel generator (MDG) power trucks, procurement of portable generators, use of fire trucks or portable pumps for cooling water supplies, provision of alternative water sources including sea water, provision of alternative portable air and nitrogen supplies for air-operated valves (AOVs) and safety relief valves (SRVs), among other measures to enhance safety.

The NPPs in Taiwan have upgraded EOPs and SAMGs in place which provide adequate instructions for the staff under such conditions.

The TPC also developed the ultimate response guidelines (URGs), which is similar to the concept of the US NEI 12-06 "Diverse and Flexible Coping Strategies (FLEX)" Implementation Guide, and can be viewed as a defense-in-depth supplement to the existing EOPs with the purpose of preventing a nuclear event from deteriorating to become a severe accident. More information about URG can be found in Subsections 6.4.3.5, 10.2(11), 18.4(C)(5) and 19.4.2.

In addition, extensive usage of mobile equipments in case of a severe accident has been implemented in all NPPs.

Furthermore, the TPC has committed to make the following safety enhancements for severe accident management:

- Establishing an organization structure to manage severe accidents

  This includes organizing the accident management teams (AMT), onsite TSCs, and off-site TSCs with support from the National Nuclear Emergency Organization (NNEO) at a national level and the emergency organization at the company level at TPC headquarters.

- Developing and implementing URG

  The goal of URG is to eliminate core damage event sequences when URG is implemented in conjunction with EOP and SAMGs.

- Developing and implementing alternative power supply systems, cooling water injection systems, and supporting system for water and air supplies
- Developing and implementing alternative cooling measures to prevent damage of spent fuel and to mitigate the release of radioactive materials from the SFP in case of SFP events

The TPC also put down a list of measures that were envisaged in order to increase the robustness of the NPPs. Among them, the following ones are highlighted:

- Enhance emergency DC power supply.
- Enhance water-tight capabilities for the fire doors of essential electrical equipment rooms.
- Improve seismic capacity of raw water hill reservoirs
  - e.g., improve the robustness of the piping of the uphill raw water reservoir by putting the fire piping above ground and connecting flexible piping at some selected positions for Kuosheng NPP.
- Install GTGs at the Lungmen site in a seismically isolated building.
- Water-tighten the existing air-cooled swing EDGs for the Chinshan, Kuosheng and Maanshan NPPs.
- Upgrade SFP additional cooling system (SFPACS) cooling tower and related piping to seismic category I at Chinshan NPP.
- Build tsunami walls at three operating NPPs sites at a height of 6 m above the current licensing basis (CLB) to increase the protection capability against the tsunami threat (e.g., to build tsunami walls of heights 17, 17 and 19 m above MSL for CSNPP, KSNPP and MSNPP, respectively).

Besides, the TPC conducted the following enhancement measures.

6.4.3.2 Electrical Power Supply Enhancement

Based on lessons learned from the Fukushima Daiichi accident, a variety of electrical power supply enhancement measures have been planned and implemented by TPC for its NPPs. For detailed information, please refer to previous Subsection 6.1.4.5 "Enhancement of Power Supply Systems."

6.4.3.3 Cooling Water Supply and Ultimate Heat Sink Enhancement

For each NPP in Taiwan, TPC built a large raw water reservoir on top of the hill to provide a water inventory sufficient for long-term cooling in case of loss of UHS. This large amount of raw water can be injected by gravity. To improve the defense-in-depth capability for situations such as a beyond design basis (BDB) SBO and loss of UHS caused by earthquake, a reinforcement of the raw water reservoir on top of the hill and the related piping will be alternatively and properly done in all 3 operating NPPs.

The normal and emergency UHSs for each site take suction from the sea. For example, in the Chinshan NPP the normal and emergency UHSs are the circulating water system (CWS) and the emergency service water (ESW) system, respectively. If normal UHS is not available, the safety grade emergency UHS is designed to remove decay heat loads for the purpose of maintaining the reactors in a safe shutdown condition and maintaining the SFPs in a stable and cooled condition. Based on the lessons learned from the accident at Fukushima Daiichi NPP, the following enhancements to provide alternative sources of cooling water for the operating plants have been implemented:

- Developed transportation and injection procedures for all water resources available, both onsite and offsite.
- Verified sufficient redundancy of fire engine resources and portable fire pumps.
- Developed schemes of alternative reactor water injection and SFP water injection using various injection paths.
- Developed schemes for alternate heat sink and recovery of UHS.
- Procured portable (or mobile) air compressors and spare nitrogen bottles for SRVs and AOVs for all the operating sites.

6.4.3.4 Seismic and Tsunami/Flooding Safety Enhancement

Investigations related to seismic safety improvements of the three operating NPPs in Taiwan started already before the Fukushima Daiichi NPP accident happened. After two faults in the vicinity of NPPs were identified and/or re-characterized, the work related to seismic hazard re-evaluation started at all three operating plants. One of these two faults, the Shanchiao Fault in the north of the Taiwan island, passes between the two plants Chinshan and Kuosheng NPPs. The other one, Hengchun Fault in the south, approaches within one kilometer of the Maanshan NPP. A significant amount of geological and geophysical work has been done both onshore and offshore. In addition, the probabilistic seismic hazard analysis (PSHA) work was conducted. However, the seismic hazard values used in the PSHA need to be updated by a Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 process. The comprehensive PSHA at the TPC’s NPPs will be finalized once the re-evaluated seismic hazard values are developed from the results of the SSHAC Level 3 study.

In light of the risks from seismic hazards, TPC has performed the seismic margin assessment (SMA) using a review level earthquake (RLE) to identify weak points, and then improved the seismic resistance ability of all three operating NPPs. For each unit, the seismic resistance ability of two success paths to shutdown the reactor and maintain it in a safe shutdown condition have been enhanced. As a result, the seismic resistance capabilities of Chinshan, Kuosheng and Maanshan reach 0.51g, 0.67g and 0.72g, respectively.

Seismic walkdowns that complement the use of these methods to assess the risk of cliff edge effects are implemented by TPC. Also, permanent seismic monitoring and alarm systems such as the ASTS systems have been installed in NPPs to improve the seismic protection. TPC also conducts periodic walkdowns to verify the capacity of equipment and materials needed to mitigate external and internal flooding, including mobile means. With respect to extreme weather condition, TPC identified some enhancement measures that consist in the dedicated walkdowns to verify trench performance, to implement mobile means (pumps, diesels) to support systems (internal drainage, electrical systems, etc.), to implement flood barriers, etc.

Besides, some other complementary measures against flooding have been planned and implemented by the NPPs including, for example, setting-up automatic closing gates, preventing large debris inlet clogging, installing flood barriers, installing water pipes above ground, installing water tight doors, etc.

6.4.3.5 Development of Ultimate Response Guidelines (URG)

During the design and construction stage of a NPP unit, all possible anticipated operational occurrences (AOOs) or transients as well as unexpected but possible accidents had been taken into consideration. Thus in case any of these design basis (DB) events occurs, the plant has in place the necessary corresponding procedures and rescue equipment for emergency response to ensure the safety of the plant. However, in spite of all the efforts done in the design, construction and operation, a beyond design basis (BDB) accident still might happen which could lead to the loss of all onsite and offsite permanently installed AC power or reactor make-up water systems and threaten the safety of the nuclear unit.

Thus, to response to a BDB accident (BDBA) like the Fukushima Daiichi compound accident, it is believed that the development of a more effective procedure to make the decisive action in time to get all possible cooling water sources (including raw water or even sea water) ready in a short time for injecting into the reactor is a necessity to prevent the core melt. Based on this understanding, the TPC developed the ultimate response guidelines (URG).

The so-called “ultimate response guidelines” may simply mean:

(1) Getting ready for injecting cooling water to the reactor from all available water sources (including raw water and sea water) when a BDB compound accident happened, resulting in a prolonged large-scale impact on the plant which might cause the loss of all onsite and offsite permanently installed AC power or reactor make-up water systems; and

(2) Based on the safety first principle, taking a decisive action to inject cooling water to the reactor from all available water sources to prevent the core melt and the release of radioactive materials outside even if the possibility of future electricity generation from the affected NPP may be sacrificed when it is judged that the reactor core cooling capability is to be lost. (see also Subsection 18.4(B)(5).)

URG can be viewed as a defense-in-depth supplement to the current emergency operating procedures (EOP) to prevent an accident from becoming a severe (core-melt) accident. The initiation of URG and related URG actions are based on the plant-specific (or site-specific) features as compared to the symptom-based abnormal operating procedures (AOPs) and EOP. Current AOPs and EOP are symptom-based. They are suitable for handling internal events. When there is a large-scale severe compound external event, like the Fukushima accident, whose effects are on the entire plant site, the response will be quite urgent and thus the symptom-based procedures may be required to enhance. Therefore, after the Fukushima Daiichi nuclear accident, each NPP of the TPC has developed an URG with respect to the plant-specific (or site-specific) features.

In the international nuclear community, some emergency response procedures and guidelines similar to the TPC’s URG have been developed recently in order to manage the beyond design basis accidents (BDBA) and severe accidents, for example, by the Boiling Water Reactor Owners Group (BWROG), Pressurized Water Reactor Owners Group

(PWROG) and the Nuclear Energy Institute (NEI). The URG is similar to the concept adopted in NEI 12-06: "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide."

The operability of the plant-specific features in the URG has been justified on the basis of rigorous systematic review and thorough accident analysis and the AEC will keep close look on the development of the URG which is subject to the review of AEC. Besides, AEC requests the TPC's NPPs to comply with the requirements of FLEX in NEI 12-06 (per USNRC NTTF Report Tier 1 recommendation 4.2) and to strengthen and integrate the EOPs, SAMGs and EDMGs with the URG consistent with the USNRC NTTF Report Tier 1 recommendation 8. More information about the TPC's URG is given in Subsections 10.2(11), 18.4(C)(5) and19.4.2.

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 related 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 and amended on March 28, 2012.

(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, 23 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.

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
23	Annual Detection Items of Radioactive Material and Equipment or Facilities Capable of Producing Ionizing Radiation

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, Notification and Response
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 Operation Permit of 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

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.

7.4 Amendment of Regulations

Nuclear regulations were constantly reviewed in this nation, especially when there was a significant nuclear-related event happened. Accordingly, a regulation would be subject to amendment if it deemed necessary. In the aftermath of the Fukushima Daiichi NPP accident in March 2011, some nuclear regulations of the AEC were amended to reflect the lessons learned. In the following, typical regulatory amendments of the AEC are to be presented.

7.4.1 Regulations Related to the Nuclear Reactor Facilities Regulation Act

Amendment of Regulations on the Scope of Inspection and on the Certification of Authorized Inspection Agencies for Nuclear Reactor Facilities:

"Regulations on the Scope of Inspection and on the Certification of Authorized Inspection Agencies for Nuclear Reactor Facilities" was promulgated on June 23, 2004 and lastly amended on January 13, 2014.

The first priority of operating a nuclear reactor facility is safety. To ensure the safety of the nuclear reactor facility related structures, systems and components during the construction of facilities, the requirements of systems and equipment design, installation, inspection, testing and other operations are to be incorporated by referencing the American Society of Mechanical Engineers (ASME) the provisions of Section III of boilers and pressure vessels of class 1 regulations: supervision and investigation job execution. Also during its operation, references the provisions of section 11 of the Boiler and Pressure Vessel Regulations ASME to perform inspection and testing, and should be based on the operator's requirements specification, during the operation provide for detecting the above structures, systems and components and test plan, submitted to the competent authority for review and approval.

7.4.2 Regulations Related to the Ionizing Radiation Protection Act

Amendment of Administrative Regulations for Radiation Protection Personnel:

Persons who already had certificates of senior, middle or primary class of radiation protection granted by the AEC before the implementation of the "Administrative Regulations for Radiation Protection Personnel" in 2003, might directly apply for the licenses of radiation protection personnel after the promulgation of the Ionizing Radiation Protection Act on January 30, 2002. This Regulation was promulgated on December 11, 2002 and lastly amended on August 31, 2011. Persons who had certificates of senior or middle class of radiation protection might submit application for licenses of senior radiation protection personnel to the AEC, while the ones who had certificates of primary class of radiation protection might apply for licenses of radiation protection personnel. The certificates of senior, middle or primary class of radiation protection personnel would be revoked two years after the promulgation of the Ionizing Radiation Protection Act.

7.4.3 Regulations Related to the Nuclear Emergency Response Act

Amendment of Directions on the Operations of the National Nuclear Emergency Response Center:

Based on lessons learned from the Fukushima Daiichi accident induced by the Great East Japan earthquake (or Great Tohoku earthquake) of magnitude 9.0 (M_w) and the ensuing tsunami, and that natural disasters occurred worldwide due to the abnormal extreme weather, plus the fact that the "Emergency Response Basic Plan" was amended on September 24, 2014, Articles 3, 4 and 5 of the "Directions on the Operations of the National Nuclear Emergency Response Center" were amended on November 4, 2014 as follows:

(1) In compliance with the amended "Emergency Response Basic Plan," directions for setting up organizations of various levels of the response center were added

in the Article 3. In addition, timing for the setting-up of organizations in the National Nuclear Emergency Response Center (NNERC), by the Disaster Prevention and Relief Office (DPRO) of the Executive Yuan, the Ministry of Education (MOE), the Ministry of Foreign Affairs (MOFA), the Ministry of Finance (MOF), and the Ministry of Science and Technology (MOST), as well as the structures of those organizations were added, too.

(2) Duties and names of the 5 organizations to be set-up by DPRO, MOE, MOFA, MOF, and MOST, respectively were added in the amended Article 4, by taking reference to the "Directions on the Operations of the National Nuclear Emergency Response Center," "Directions on the Management of foreign nuclear disaster," together with the corresponding tasks assigned to the organizations to be set-up according to Radiation Disaster Prevention and Relief Action Plan.

(3) Procedures for the prevention and relief of a nuclear accident with natural disasters such as earthquake, tsunami, etc. occurred concurrently were added in the Article 5.

7.4.4 Regulations Related to the Nuclear Materials and Radioactive Waste Management Act

Amendment of Regulations on Final Disposal of Low Level Waste and Safety Management of the Facilities:

"Regulations on Final Disposal of Low Level Waste and Safety Management of the Facilities," promulgated on September 10, 2003, was lastly amended on July 9, 2012 about Articles 2 and 6. The points of amendment are as follow:

(1) The specific test methods for uniformly-solidified body of the radioactive waste were made applicable for the general uniformly-solidified body by following the definition specified in item 4 of Article 2 and the method specified in Annexed Table 3 (Test Items, Methods, and Standards of Homogeneous Solidified Low level Waste).

(2) The test contents for resistance to bacteria and to radiation were revised in the amended Article 6.

Amendment of Regulations for the Operation Permit of Radioactive Waste:

"Regulations for the Permit of Import, Export, Transit, Transship, Transport, Discard, and Assignment of Low Level Radioactive Waste," promulgated on December 24, 2003 according to "Nuclear Materials and Radioactive Waste Management Act," was amended on July 31, 2012 and renamed as "Regulations for the Operation Permit of Radioactive Waste." The points of amendment are as follows:

In "Nuclear Materials and Radioactive Waste Management Act," the radioactive wastes are classified into low level radioactive waste (LLRW) and high level radioactive waste (HLRW), while "Regulations for the Permit of Import, Export, Transit, Transship, Transport, Discard, and Assignment of Low Level Radioactive Waste" regulated only LLRW. Thus this Regulation was amended to include the regulations for HLRW and renamed as "Regulations for the Operation Permit of Radioactive Waste" on September 19, 2014.

Amendment of Regulations on the Final Disposal of High Level Radioactive Waste and Safety Management of the Facilities:

"Regulations on the Final Disposal of High Level Radioactive Waste and Safety Management of the Facilities," promulgated on August 30, 2005, was amended on January 18, 2013. According to the original Article 6 of this Regulation, the applicant for the construction license of a final disposal facility for the HLRW was required to submit an environmental impact assessment (EIA) report approved by the Environmental Protection Administration (EPA) and a detailed site survey plan in order to obtain an approval for the authority to conduct a detailed investigation of the site. However, this EIA requirement for the application of the construction license (CL) of a final disposal facility for the radioactive waste is already mentioned in Article 3 of the "Regulations for the Review and Approval of Applications for Construction License of Radioactive Wastes Treatment, Storage and Final Disposal Facilities" as amended on April 13, 2009. Therefore, Article 6 of "Regulations on the Final Disposal of High Level Radioactive Waste and Safety Management of the Facilities" was amended to discard the requirement of EIA.

Amendment of Fees for Regulatory Services under the Nuclear Materials and Radioactive Waste Act:

"Fees on Regulatory Services under the Nuclear Materials and Radioactive Waste Act," promulgated on June 3, 2003, was amended on November 23, 2009 before the Fukushima accident. Since the occurrence of this accident, the safety requirements of all nuclear facilities became more rigorous and thus more resource and manpower had to be put into the regulatory inspection and review of the radioactive materials related activities. In addition, to comply with the upgrading requirements of protection against the compound disaster induced by severe external events such as earthquake and tsunami, a 10-year periodic safety review (PSR) was required for the storage facilities of LLRW. Therefore, the "Fees on Regulatory Services under the Nuclear Materials and Radioactive Waste Act" was re-amended on November 25, 2015.

Amendment of Regulations for the Review and Approval of Applications for Decommissioning Permit of Nuclear Reactor Facilities:

"Regulations for the Review and Approval of Applications for Decommissioning Permit of Nuclear Reactor Facilities," promulgated on July 14, 2004, was amended on July 9, 2012. According to the original Article 2 of this Regulation, the applicant for an approval of decommissioning a nuclear reactor facility was required to submit an EIA report approved by the EPA in addition to the decommissioning plan and financial assurance documents. However, although this EIA requirement is one of the prerequisites for issuing the approval of decommissioning, it is needed only when the regulatory body is going to make the final decision on the application. Therefore, Article 2 of this Regulation was amended on July 9, 2012 to discard the EIA requirement at the initial application stage, but require the EIA report to be submitted before the AEC is to make a final decision after the overall application review.

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 36 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 medical 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.

In order to develop sufficient trust of the public on the regulatory control of nuclear power operation in this nation, the AEC did its best in 2016 to make reports and information of its nuclear regulation open to the public and increased the frequency of holding press conference. The AEC also tried to visit and have dialogue with the environmental protection groups and nuclear-concerned organizations as frequently as possible, and had a communication meeting with them based on the needs.

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 as 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.

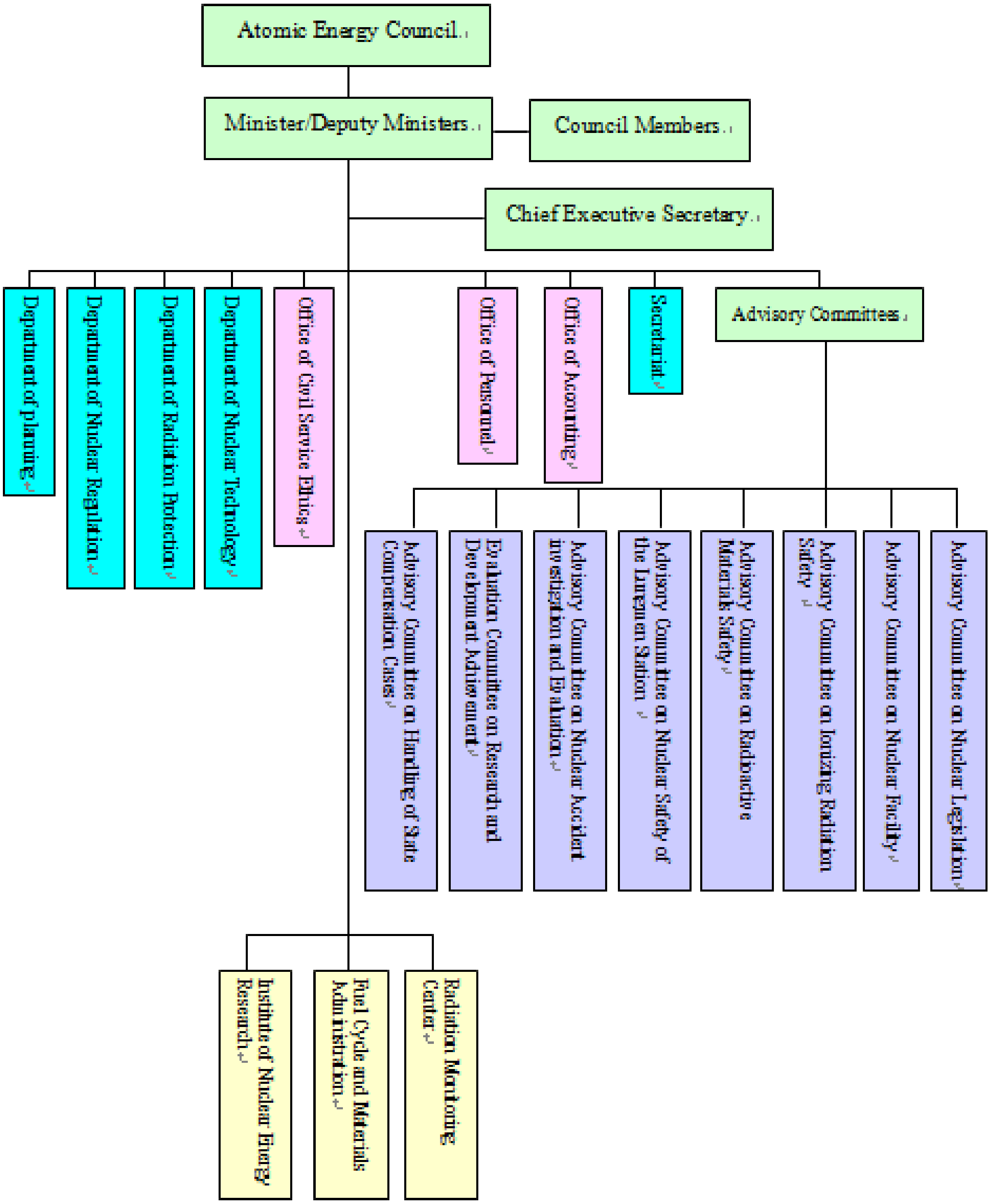


Figure 8.1 Organization Chart of the AEC

The AEC headquarters employs approximately 206 personnel with the FY2015 budget of

NT$593 millions (not including the budget of the three affiliated agencies) as shown in Table 8.1.

Table 8.1 Budget and Staffing by Appropriation

Appropriation	Budget (Million NT Dollars)			FY Staffing (Man-year)		
	FY2013	FY2014	FY2015	FY2013	FY2014	FY2015
AEC Headquarters	523	572	593	176	190	206
INER	2,310	2,210	2,200	813	793	837
FCMA	92	86	82	36	39	44
RMC	79	70	72	30	32	40
Total	2,998	2,935	2,947	1,055	1,054	1,127

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,
- Certification of radiation protection personnel,
- 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 AEC's Department of Nuclear Technology (DNT) is the government's primary competence for nuclear security and radiological emergency response. DNT maintains a high level of readiness for protecting the country through the development, implementation and coordination of programs and systems designed to serve as a last line of defense in the event of a nuclear incident or radiological accident. It is the DNT's most concern to minimize the impacts of emergencies on worker and public health and safety, and the environment. The major tasks of DNT 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 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 764 personnel including researchers, technicians, and supporting staff in 2015. The researchers are all professionals with graduate degrees including 115 with doctoral and about 181 with Master degree. The FY2015 budget of INER was NT$2,200 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 Radio-pharmaceutics, 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. This facility was designed to receive all the solidified low-level radioactive wastes generated in the country, especially those from the operation of NPPs of the 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 materials 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 advices from this Committee will be referred to by the AEC.

(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 related community. This Committee advises on radiation safety issues and gives expert opinions on the related uses of radiation and radioisotopes. It also advises the AEC management, as required, on matters of radiation safety 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, which is not an advisory type committee, is 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 neighboring public around the said NPP. The responsibility of this committee covers 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 2015 are 593, 2,200, 82 and 72 million NT Dollars for the AEC headquarters, INER, FCMA and RMC, respectively. The total annual budget for the AEC all together reaches 2,947 million NT Dollars in 2015. 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. A sum of 54 million NT Dollars is 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.

8.1.4.4 Human Resources

(1) Recruitment and Hiring Process

The number of staff in the AEC headquarters, INER, FCMA and RMC are 185, 764, 41, and 31, respectively in the fiscal year 2015. As all the staffs 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 the retirement or departure of current staff. The Civil Service Level 2 and 3 Senior "Examinations" 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's or doctor's 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 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 the certificates of 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 Ministry of Health and Welfare (MOHW), the Ministry of Labor (MOL), the Ministry of Foreign Affairs (MOFA), and the Directorate General of Budget, Accounting and Statistics (DGBAS).

(1) Ministry of Economic Affairs (MOEA)

The Ministry of Economic Affairs 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 more information, please refer to Subsection 8.2.

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 27,140 and its assets worth 1,928 billions NT Dollars in July 2015. As of December 2014, the total installed electric power capacity of the TPC reached 31,651 MWe (Nuclear: 5,144 MWe, 16.25%; Fossil: 21,560 MWe, 68.12%; Hydro: 4,644 MWe, 14.67%; Wind: 287 MWe, 0.91%; Solar: 16 MWe, 0.05%). Its main mission is to maintain the stable supply of electric power with good quality and reasonable price.

(2) Environmental Protection Administration (EPA)

The Environmental Protection Administration, 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 (NFA, MOI)

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 observation and evaluation of emergency drills at the nuclear power stations.

(4) Ministry of Health and Welfare (MOHW)

The Ministry of Health and Welfare 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 Ministry of Health and Welfare to issue licenses for hospital workers operating the X-ray units or accelerators, or handling the radioisotopes or radiopharmaceuticals that release radiation.

(5) Ministry of Labor (MOL)

The AEC closely monitors the legislations proposed by the Ministry of Labor, 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 (MOFA)

The AEC works with the Ministry of Foreign Affairs 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 (OECD/NEA), 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) Directorate-General of Budget, Accounting and Statistics (DGBAS)

The Directorate-General of Budget, Accounting and Statistics 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 the Ministry of National Defense and the 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 which they are interested in and considering participation of the county officials in the AEC inspection activities or the advisory committees. However, the counties are allowed to observe and assist AEC's inspections, but not to 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 with 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 the Ministry of Education, the Ministry of Culture, the National Palace Museum, the Government Information Office, the Academia Sinica and the 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

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 used to play the role for promoting various types of energy including nuclear power, while the AEC focused on the regulatory part of nuclear energy utilization. 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 various efforts including 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.

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 of the License Holder for the Safety of Nuclear Installations

According to Articles 5 and 7 of the Nuclear Reactor Facilities Regulation Act, the holder of the construction license (CL) of a nuclear reactor facility assumes the responsibility to safely construct a NPP as approved and with the conditions imposed by the regulatory body at the time when the CL 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 NPP was granted, the holder of the operating license (OL) shall assume the responsibility for safely operating the NPP as approved to ensure that the health and safety of the public are protected.

In other words, based on the nuclear regulation laws in Taiwan, the prime responsibility for the nuclear safety of the operating NPPs and the NPP under construction lies on the license holder, which in Taiwan is the TPC.

9.1.1 Organization of the Taiwan Power Company and Mechanism for the License Holder to Discharge Its Prime Responsibility for Safety

Figure 9.1 shows the latest organizational structure of the TPC after its reorganization on January 4, 2016. As shown in this figure, the TPC is composed of four main divisions (including the Distribution and Service Division, Transmission System Division, Nuclear Power Division, and Power Generation Division), one research institute (the Taiwan Power Research Institute), several offices, many departments and about seven committees responsible for driving cross-departmental issues.

On December 17, 2015, the TPC announced that its nuclear sector will be reorganized and later the Nuclear Power Division was established on January 4, 2016. As shown in Figure 9.2, the TPC Nuclear Power Division is composed of Department of Nuclear Generation (DONG), Department of Nuclear Safety (DNS), Department of Nuclear Engineering (DNE), Department of Nuclear Back-end Management (DNBM), Nuclear Safety Committee (NSC) and a Planning Office as well as the Nuclear Emergency Planning Executive Committee (NEPEC). The Chief Executive Officer (CEO) of this Nuclear Power Division is a Vice-President (in Nuclear) of the TPC, while the deputy-CEO is a Chief Engineer.

TPC has three NPPs in operation and one nearly-completed NPP in mothballs. Each NPP has two identical nuclear units. Figure 9.3 is the general organization chart of a TPC's NPP. In each of the three operating NPPs, there are one Plant General Manager (PGM) and three Deputy Plant General Managers (DPGM). Among these 3 DPGMs, one is in charge of the operation. DPGMs will act as assistants to the PGM and, in case the PGM is absent or not available, one of DPGMs will take the authorities to act as the PGM. Both the PGM and his deputy of operation must have valid SRO certificates issued by the TPC, although they are not required to have valid SRO certificates issued by the AEC.

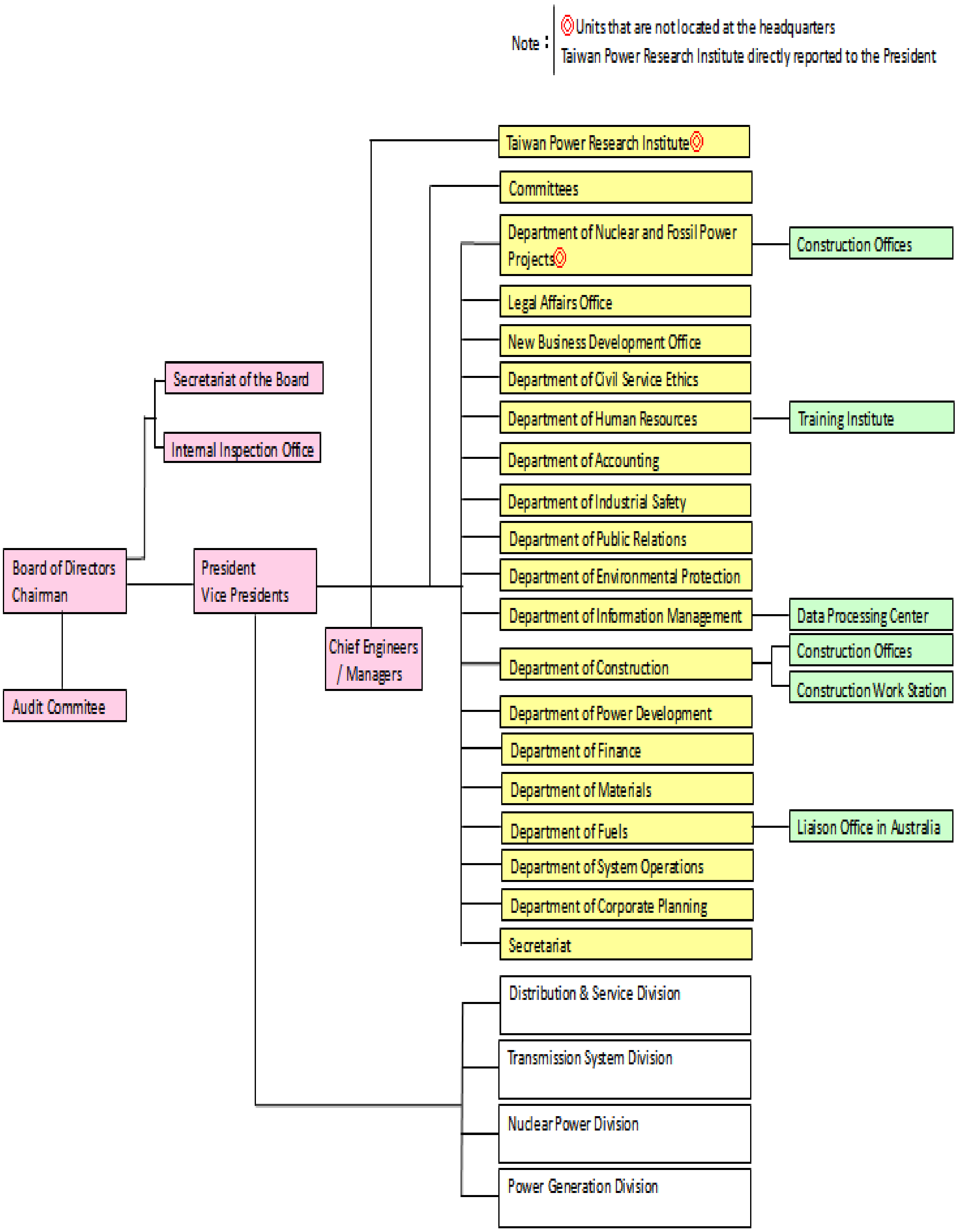


Figure 9.1 Organization Chart of Taiwan Power Company

As specified in the approved FSAR of each operating NPP, 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 PGM shall take the principal responsibility for all phases of operation and maintenance. He is responsible for the safe, orderly, and efficient operation of the NPP and for the compliance of operation with the requirements of the operating license and technical specifications (TS). 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 PGM reports to the Vice-President of Nuclear of the TPC via the Director of 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.

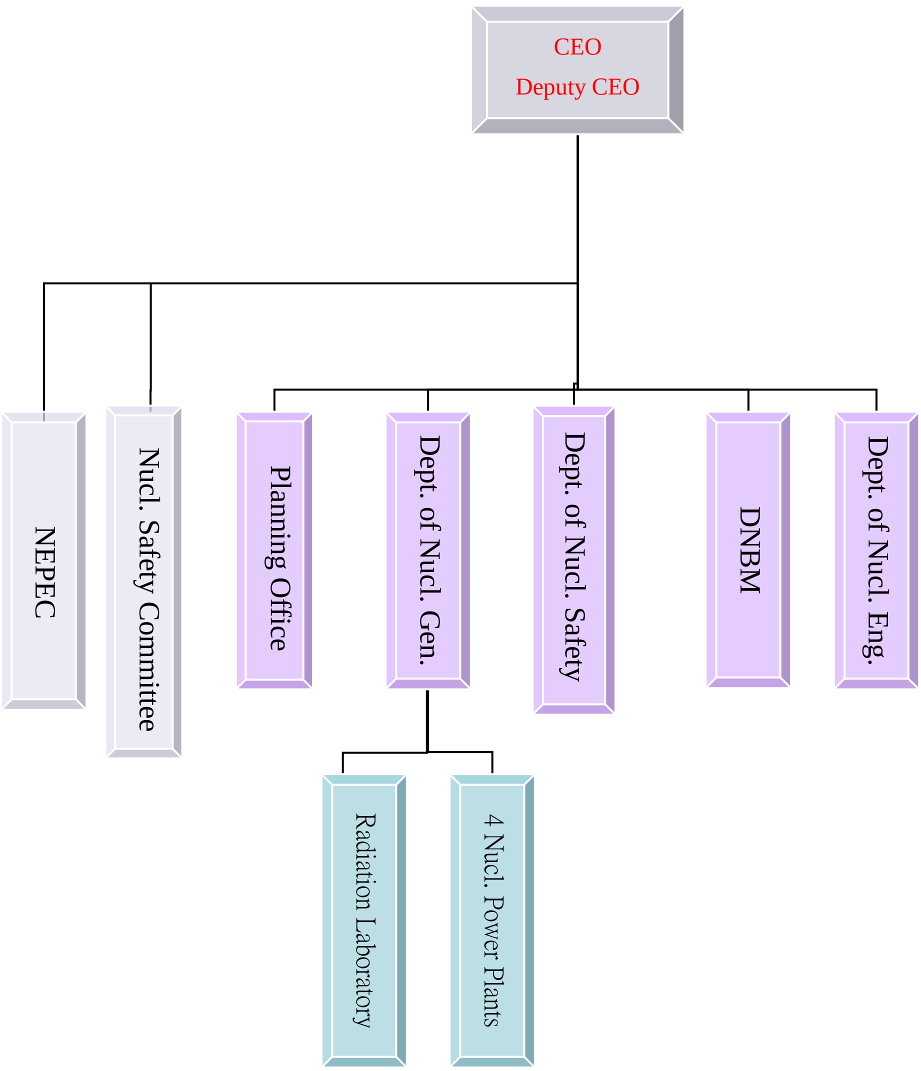


Figure 9.2 Organization Chart of the TPC Nuclear Power Division
(台電公司核能發電事業部)

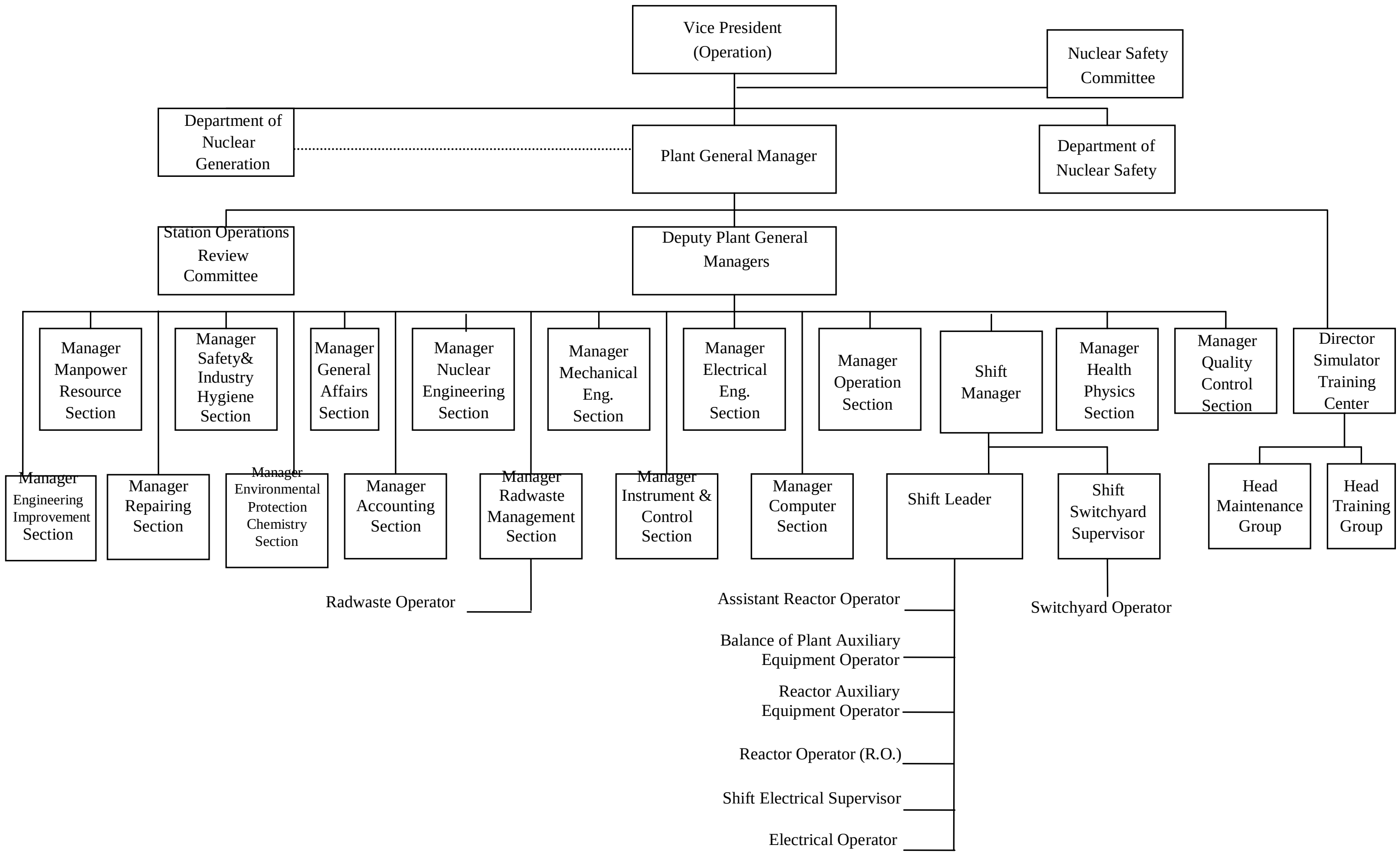


Figure 9.3 General Organization Chart of the TPC's Nuclear Power Plant

During the periods when the PGM and all three DPGMs are not available, the PGM will delegate his responsibility to the Operation Section Manager. In the time period after office hours or when the PGM, three DPGMs and the Operation Section Manager are all not onsite, the on-duty Shift Manager (SM) of the plant will be responsible for the operation of the plant and the compliance of operation with the requirements of the OL and TS.

The supervision of the plant operation and performance is under the direction of the Operation Section Manager who reports to the DPGM 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 SMs of the plant receive technical direction from the Operation Section Manager, but report directly to the DPGM in charge of operation. The SMs who must have valid SRO licenses issued by the AEC are in charge of the plant operation during their shifts and have the authority to shutdown the reactor if necessary under their judgment.

The on-duty Shift Leaders (SL) of the nuclear units 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 SM is not available, the on-duty SL of a unit has the authority and responsibility to act in place of the SM.

In each NPP, there is a Station Operation Review Committee (SORC) organized to advise the PGM on matters concerning nuclear safety. In an operating NPP, the SORC is responsible for reviewing all safety-related affairs and making recommendations to the PGM. As an illustration, major 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 TS,
- All proposed changes or corrections which may affect the nuclear safety systems or components,
- All TS violation events,
- All reportable events,
- Plant emergency plan,
- Etc.

When newly prepared or revised, a plant operating procedure will be reviewed by the SORC of the NPP, and then approved by the Plant General 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 TPC (the license holder).

9.1.2 Mechanism for the License Holder to Maintain Open and Transparent Communication with the Public

With sincere attitude and fair view-points, the TPC presents information about the company in six aspects to the public through the TPC's web site, including the information on management, power generation, demand & supply of electricity, customers, environment, and construction engineering. In the aspect of power generation, for example, one can obtain information from the TPC's web site about the current status and performance records of the fossil power and/or nuclear power management, renewable energy, electricity purchased by TPC from the independent power producers (IPP) (i.e., private utilities), and measures in response to the Fukushima Daiichi nuclear accident.

In addition, the TPC formed a "Nuclear Communication Team (NCT)" to act as a communication window about the nuclear affairs between the general public and the TPC. The ways this NCT communicated with the public at least include:

- Regularly holding international nuclear conferences to have foreign experts giving speeches about nuclear related knowledge to the public.
- Dispatching trained employees from nuclear sectors of the TPC to communicate face-to-face with the public, students and private companies.
- Installing a specific web site to discuss the fundamental nuclear knowledge and release information about specific nuclear-related topics and regularly renew the contents of this website.
- Issuing newsletters to respond actively to incorrect reports on the media.

More information about the licensee's transparent policy can be seen in Subsections 10.3 and 19.10.

9.1.3 Mechanism for the License Holder to Ensure Having Appropriate Resources (Technical, Human, Financial) for On-site Accident Management and Consequence Mitigation

As shown in Figure 9.4, there is a Nuclear Emergency Planning Executive Committee (NEPEC) under the TPC Nuclear Power Division in TPC Headquarters which is responsible for emergency response to nuclear accidents. In case a nuclear accident happens in a NPP, the NEPEC will be in charge of directing and manipulating the manpower, materials and finance support to the onsite emergency response actions in order to mitigate the consequence and recover from the accident. Whenever offsite supports are needed in emergency response to a nuclear accident onsite, the NEPEC with the approval of its Chairman may report to National Nuclear Emergency Response Center (NNERC), which is under the Central Disaster Response Center (CDRC) (refer to subsection 16.1.1.1), to ask for help.

The Chairman of NEPEC is also the Vice-President in Nuclear of TPC. As shown in Figure 9.4, the NEPEC is composed of 10 groups: Dose Assessment, Accident Assessment, Environmental Monitoring, Maintenance Support, Regulatory Planning, Exercise Planning, Operational Support, Public Relation, Logistic Support, and Finance & Accounting. Under the supervision of NEPEC, there is also an Emergency Operation Facility (EOF) organized by technical people dispatched from groups of Operational Support, Dose Assessment, Accident Assessment, and Environmental Monitoring of NEPEC.

The EOF is a TPC (the licensee) controlled and operated offsite support center. It has facilities for:

- Management of overall licensee (TPC) emergency response,
- Coordination of radiological and environmental assessment,
- Determination of recommended public protective actions, and
- Coordination of emergency response activities with central and local government organizations.

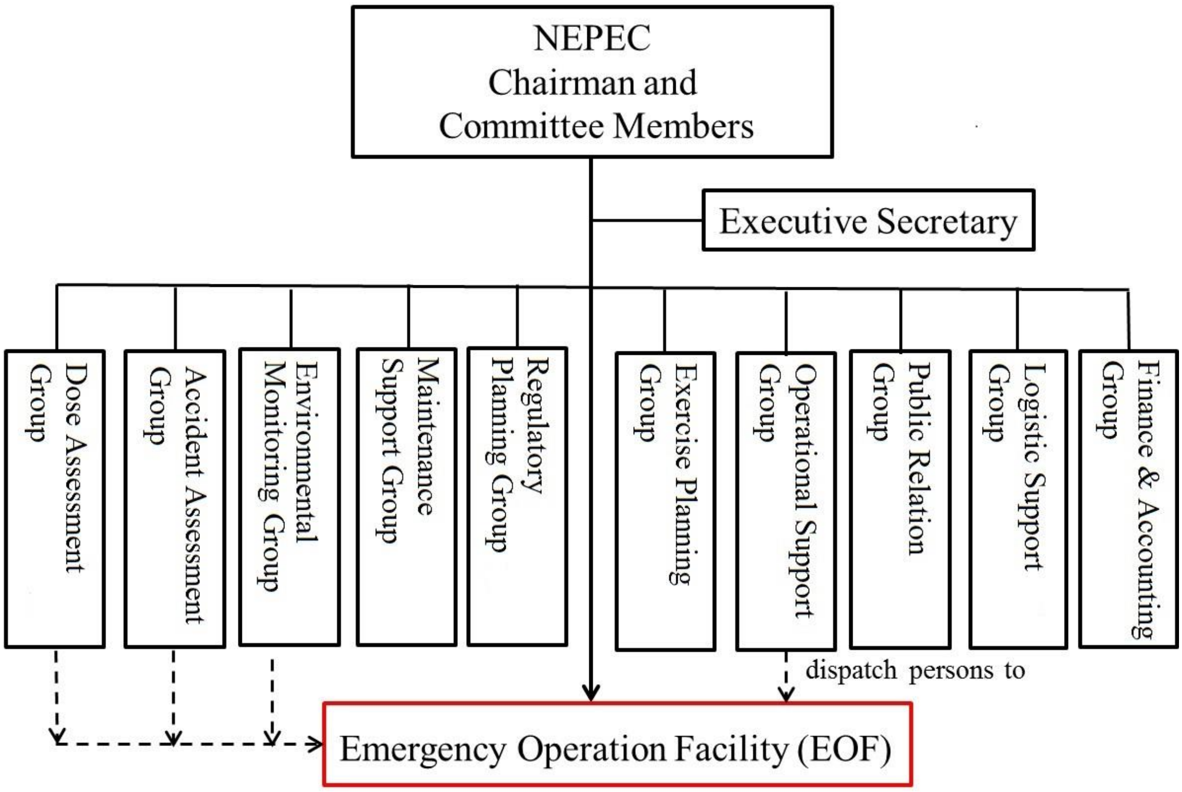


Figure 9.4 Organization Chart of the Nuclear Emergency Planning Executive Committee (NEPEC)

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 NPP 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 NPP shall receive pre-operational inspections from the AEC to verify that the NPP 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 initial loading of the nuclear fuel. Then, with the completion of all the 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 NPPs.

The operation of a NPP 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 NPP.

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 NPP. For example, if the operator of a NPP 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 NPP 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. More information about the penalty for violations of regulations can be found in Subsection 19.7.

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 Overview of the Arrangements and Requirements to Prioritize Safety

10.1.1 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 AEC holds the responsibilities over 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 implements the laws for nuclear safety control, radiation protection, environmental radiation 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 exert more efforts in enhancing the transparency of nuclear safety information to the public.

10.1.2 Commitment to Safety

The 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 its 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. In other words, 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.

10.1.3 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 as previously described in Subsection 6.2.1.

10.1.3.1 Safety Culture Implementation Plan

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.

10.1.3.2 Safety Culture Reinforcement Plan

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

(1) Number of incidents caused by human errors,

(2) Number of incidents violating the regulation of the 4^{th} degree and above,

(3) Number of unplanned auto scram in 7000 hours of critical condition (number per unit),

(4) Performance of safety systems (% of total time in service),

(5) Number of industrial safety (number of incidents per 200 thousand man-hours per plant), and

(6) Percentage of unplanned capacity loss.

Category 2: Quantitative Indicators for the Processes

(1) Number of walking-management (man-times per quarter),

(2) Number of supervision of tool-box meeting (man-times per quarter),

(3) Number of supervision of self-assessment (man-times per quarter),

(4) Number of safety condition improvement (man-times per quarter),

(5) Percentage of completion of request for equipment repair (%), and

(6) 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.

10.1.3.3 Safety Culture Advanced Plan

Based on lessons learned from the Fukushima NPP accident, an advanced safety culture program has been developed by TPC since June 2011 to foster a high level of nuclear safety for protecting people's life and property safety. This advanced SC program includes 4 major areas: management effectiveness, contractor management, risk management and personnel performance as follows:

(1) Management Effectiveness (4 items)

- Effective communication and implementation of organizational goals,
- Enhanced efficiency of site management,
- Verification and correction of management efficiency, and
- Lesson learned from exemplary behavior.

(2) Contractor Management (6 items)

- Establishing mechanism for the evaluation of contractors' personnel work effectiveness,
- Complying with the contractors' personnel working rules.
- Upgrading contractors' technical capabilities,
- Strengthening contractors' awareness of safety culture,
- Emphasizing the feedback of contractors' experience, and
- Training on the occurrence of reactor trip or scrum during maintenance or after maintenance.

(3) Risk Management (5 items)

- Maintaining good risk management during outage,
- Maintaining good risk management during operation,
  - Emphasizing the availability of the risk-significant systems for ensuring the safety margin of NPP operation,
  - Using the tool of maintenance integrated risk utilities (MIRU) to handle the routine maintenance schedules, and
  - Implementing a control of potential trips to reduce the maintenance risk.
- Maintaining good management of core reactivity,
- Maintaining emergency preparedness to response to compound disaster, and
- Conducting careful assessments if the off-site power distribution of a NPP is changed.

(4) Personnel Performance (4 items)

- Continuing the promotion of training to prevent human errors,
- Carrying out the application of error-prevention tools on the job,
- Conducting the post-job meeting, and
- Strengthening the completeness of various procedures by using "progressive verification, paragraph confirmation, and integration".

10.1.4 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 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. The most important tasks of the AEC have been gradually 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

10.1.4.1 Licensing

Typical and of utmost importance in licensing is for the construction and operation of nuclear power plants (NPPs). As mentioned previously in Subsection 6.2.3.1, a two-step licensing review process is followed for the issuance of construction, initial fuel loading, and start-up test permits, before an operating license (OL) is issued. An operating license will be given after a detailed review on the FSAR and start-up test results by checking the fulfillment of licensing requirements. As plant operators play a key role in their dynamic responses to normal operations and anomalies, their qualification and ability are crucial for nuclear safety. Plant reactor operators (ROs) and senior reactor operators (SROs) are required to pass stringent tests, including written examination, plant walk-through and simulator operation, before they are allowed to work at the main control room (MCR) of the plant.

Similar to most countries, the Nuclear Reactor Facilities Regulation Act of Taiwan requires the licensee to conduct Periodic Safety Reviews (PSRs) every 10 years for the operating NPPs in agreement with IAEA safety standards. The 10-year integrated safety assessment report (ISAR) includes chapters on safety performance of radiation, radwaste management, major plant modifications, aging management of SSCs, seismic re-evaluation, lessons learned from significant events, and feedback from domestic and foreign experiences and research results. More detailed descriptions of the PSRs are provided in Subsections 6.2.3.4 and 14.1.2(2) of this report.

10.1.4.2 Inspections and Enforcement

While safety evaluations aim at compliance between licensing documents and safety requirements, the regulatory inspections concentrate on whether the work and performance on the scene meet the design requirements set forth in the licensing documents. Resident inspections, regular inspections, outage inspections, expert team inspections, and special team inspections as well as the unannounced inspections afford stringent and independent enforcement measures upon licensee of NPPs. Resident inspectors are assigned to check daily operation and selected surveillance tests in operating power plants. Periodic outage inspections are performed to assure the quality of maintenance work during each unit outage. Team inspections are typically conducted for special tasks with corresponding teams of experts. In addition, unannounced night-shift inspections are conducted semi-annually at each plant. More information about regulatory inspection is provided in Subsection 6.2.3.2.

10.1.4.3 Reactor Oversight Process

To provide the public an easy way to understand the safety levels of the operating NPPs, the AEC has referred to the reactor oversight process (ROP) of the USNRC and developed the domestic ROP at the end of 2004, as mentioned previously in Subsection 6.2.3.3. The performance indicators (PI) 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.

The AEC has posted the performance indicators (PI) of nuclear power plants and significance determination process (SDP) evaluation results of inspection findings on its website since 2004 and 2006, respectively. Inspectors conduct inspection and initial screening of the findings.

The PRA is used to consider the probability, progression and consequences of equipment failures or transient conditions to derive numerical estimates that provide a consistent measure of the safety of reactor facility. The AEC developed and used PRA as a tool to enhance the evaluation of the safety of NPPs since 1982 as described in Subsection 6.3.6. The "PRA Model Based Risk Significance Evaluation Tool" (PRiSE) developed by the INER is used by the AEC for risk significance estimation and justification in SDP. The TPC started to submit to AEC the PRA results from data updates every 3 years since late 1990. The TPC is committed to follow the PRA standards to build-up, update and maintain the PRA models. The TPC will update the PRA Low Power and Shutdown (LP&SD), level 2 (L2) and level 3 (L3) models based on the dedicated ASME PRA standards. More detailed descriptions of the PRA are provided in Subsections 6.3.6 and 14.1.2(4) of this report.

10.1.5 Independent Safety Assessment

10.1.5.1 INPO and WANO Safety Review

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 reviews and discussions. For example, during the last few years, TPC had the following relevant activities:

(a) During July 28-August 1, 2014, WANO-TC sent two experts to the Linkou Nuclear Training Center for peer review of the standard training.

(b) During August 18-28, 2014, a team of twelve experts organized by WANO- TC visited TPC Headquarters for the WANO Corporate Peer Review (CPR).

(c) During November 13-25, 2014, a team of twenty-five experts organized by WANO- TC visited Kuosheng NPP for the WANO peer review.

(d) During December 1-5, 2014, WANO-TC sent two experts to Lungmen for "Preparation and Implementation for the Nuclear Power Plant Layup" technical support missions (TSM).

(e) During February 2-6, 2015, WANO-TC sent five experts to Kuosheng for "Radiation Dose Reduction Countermeasures-ALARA" TSM.

(f) During March 9-13, 2015, WANO-TC sent four experts to Maanshan for "Improve the Reliability of Emergency Diesel Generators" TSM.

(g) During August 3-7, 2015, WANO-TC sent two experts to Linkou Nuclear Training Center for peer review of the standard training.

(h) During August 31-September 4, 2015, WANO-TC sent five experts to Headquarter for "Approach and Conduct in Corporate Oversight and Monitoring" TSM.

(i) During June 11-25, 2015, a team of twenty-five experts organized by WANO-TC visited Chinshan NPP for the WANO peer review.

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 guidelines 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.

10.1.5.2 Independent Review on Stress Test Reports

After the nuclear accident at Fukushima, a stress test program (in accordance with the EU specification developed by EC/ENSREG) was performed on each NPP in Taiwan to verify the robustness of design and recognize the cliff-edge effect and hidden weakness. The stress tests focused on three principle areas:

(1) Extreme external event initiators such as earthquakes, flooding and other extreme natural events

(2) Loss of safety functions and systems due to the loss of power, ultimate heat sink (UHS), or the combination of both

(3) Accident management

The stress tests were completed in 2012 and reports (originally written in Chinese) were submitted to AEC for review in March – April 2012. In March and September 2013, the OECD/NEA and EC/ENSREG expert teams had arrived Taiwan to conduct independent peer reviews of the stress tests for the Taiwanese NPPs, respectively, under the invitation of the AEC. The review results are briefly illustrated as follows:

(1) Major Findings of the NEA Peer Review:

  (a) Need to perform fault displacement hazard analysis

  (b) Need to deploy a local seismic network near NPPs to capture small earthquakes in order to understand whether or not the pattern of the epicentre indicates correlation with osculated tectonic features.

  (c) Need to provide an interface between the post-earthquake and post-tsunami operating procedures

  (d) Need to systematically assess the combinations of events in the areas of flooding and extreme natural events

  (e) Need to check the probable maximum precipitation with regional topographical maps

(2) Major Findings of EU Peer Review:

  (a) Need to conduct site-specific basis assessment for mud flows and mass movements

  (b) Need to conduct post-seismic inspection for SSC with a low seismic classification

  (c) Need to develop strategies to minimize the quantities of contaminated water produced under accident conditions and to evaluate possible options to create closed cooling circuits

  (d) Need to improve the availability of RPV depressurization for BWRs

(e) Need to improve the habitability in the local shutdown panel areas under accident conditions

(f) Need to conduct the training of SAMGs for multi-unit events

(g) Need to improve the infrastructure and ensure the availability of heavy road-clearing equipment, as roads/bridges representing the weak points to access sites in case of a strong earthquake

Based on the results of stress tests as well as the recommendations of the OECD/NEA and EC/ENSREG independent peer reviews, the AEC issued three batches of regulatory orders on 2012/11/05, 2013/06/06 and 2014/03/06, respectively. The milestones of major activities in the Taiwanese Safety Re-assessment and Stress Tests Programs are given in Table 6.7. More detailed descriptions of the safety assessment, stress tests and independent inspection, and follow-up actions are provided in Subsections 6.4.1, 14.3.1 and 14.3.2 of this report.

10.2 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 Operating 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 (SAMGs) 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.

After the Japan's Fukushima Daiichi Accident, AMT has integrated the information of NEI 14-01 revision 0 emergency response procedures and guidelines for beyond design basis events and severe accidents, Boiling Water Reactor Owners Group (BWROG), Pressurized Water Reactor Owners Group (PWROG) and Nuclear Energy Institute (NEI) to develop abnormal operating procedures (AOPs), emergency operating procedures (EOPs), FLEX (diverse and flexible coping strategies (NEI 12-06)) support guidelines (FSGs), extensive damage mitigation guidelines (EDMGs), SAMG and ultimate response guidelines (URG). It is believed that these procedures and guidelines of the three existing NPPs can enhance the severe accident management capability of the plant staff. More detailed descriptions of the severe accident management for safety enhancement are provided in Subsections 6.4.2.

(6) Corrective Action Program

With reference to the WANO's guidelines, the Maanshan 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. More information about CAP is given in Subsection 6.3.7.

(7) Performance of the Maintenance Rule

In order to regulate the effectiveness of the NPP maintenance, USNRC promulgated "Maintenance Rule" (MR) 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 MR. 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 MR 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 MR. While for the Kuosheng NPP, there were 202 system functions being included in the scope of MR in the same year. As for the Maanshan NPP, the Maintenance Integrated Risk Utilities (MIRU) computer program was established in 2007 and applied for the maintenance risk management. The numbers of items belonged to the scopes of MR in the Maanshan NPP are 242 in 2008. In conclusion, the advantages of implementing the Maintenance Rule in the existing NPPs in Taiwan include:

(a) Implementation of MR 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 MR, and hence, the reliability of equipment can be effectively improved.

(c) The effectiveness of performing the MR can be continuously improved by periodical performance evaluation.

(8) Improvement of the Technical Specifications

Originally, the Technical Specification (TS) 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. More information about the implementation of the ITS in the domestic NPPs can be found in Subsection 6.3.4.

(9) Investigation of Near-Miss Events

The near-miss events are those events that have some component failures or malfunctions, but the severity of which does not reach the level of an abnormal event. 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.

(10) Installation of Automatic Seismic Trip System

As mentioned in Subsection 6.3.2, the disastrous Chi-Chi earthquake (M=7.3) occurred on September 21, 1999 prompted the AEC to request the installation of an automatic seismic trip system (ASTS) at each of the TPC’s six operating nuclear units to further ensure plant safety. Installations and tests were completed in November 2007, and the systems have been put in service since then.

(11) Development of Ultimate Response Guidelines

EOP and SAMG are symptom-based procedures to cope with severe accidents, depending on real-time operational parameters to mitigate the event consequence. For the compound severe accidents, such as the Fukushima nuclear accident, its impact on NPP will relatively spread rather than focus on one system or one area. With regard to this fact, the TPC therefore developed the ultimate response guidelines (URG). The URG is specifically designed to cut off event evolution and make immediate actions to prevent core damage. URG will be integrated with EOPs, SAMGs, and EDMGs similar to the Recommendation 8 of the USNRC Near-Term Task Force (NTTF 8).

The URG guides the plant operators’ conduct of reactor depressurization, core cooling water injection, and containment venting to prevent BWR, PWR and ABWR from encountering core damage by beyond design basis accidents (BDBA) such as: (1) events severer than SSE earthquake and tsunami, (2) SBO, and (3) loss of UHS. Figures 10.1 and 10.2 show the schematic and the flow chart of URG, respectively. The R&D experience of the TPC’s URG was sent to discuss in the international organization BWR Owners Group (BWROG) committee. The draft and official edition of EPG/SAG Rev.4 will be released in 2017 and 2019, respectively. More information about the TPC’s URG is given in Subsections 6.4.3.5, 18.4(C)(5) and 19.4.2.

10.3 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.

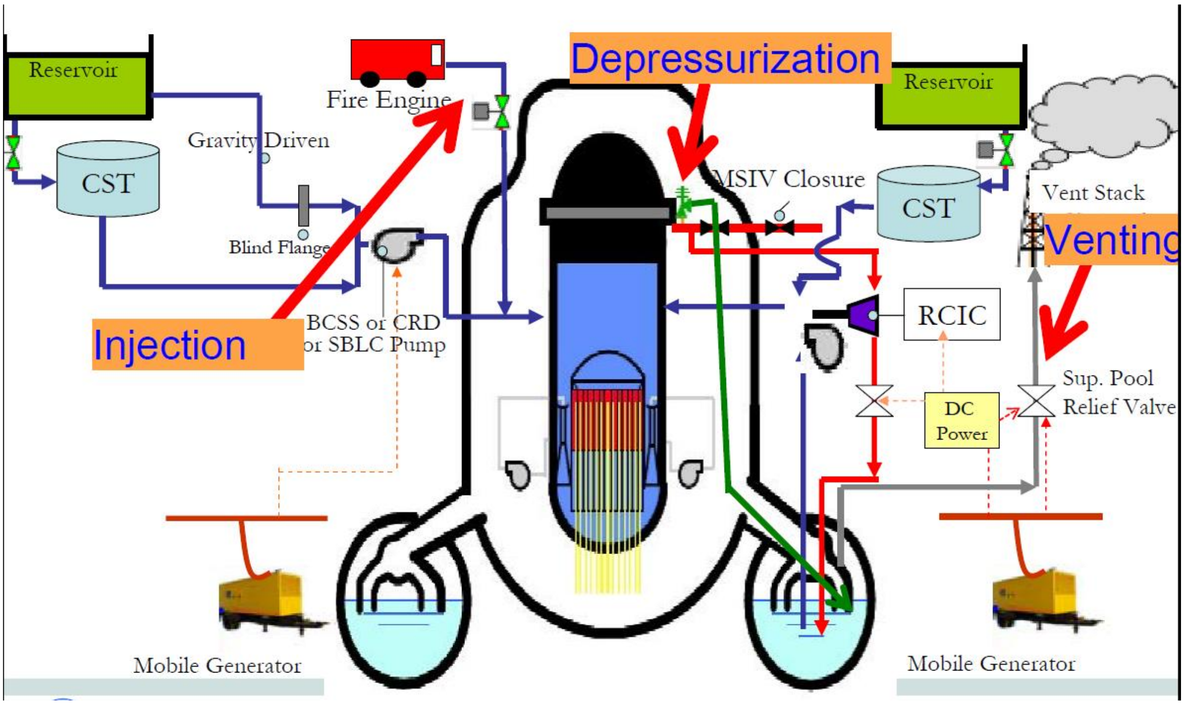


Figure 10.1 The Schematic of BWR URG

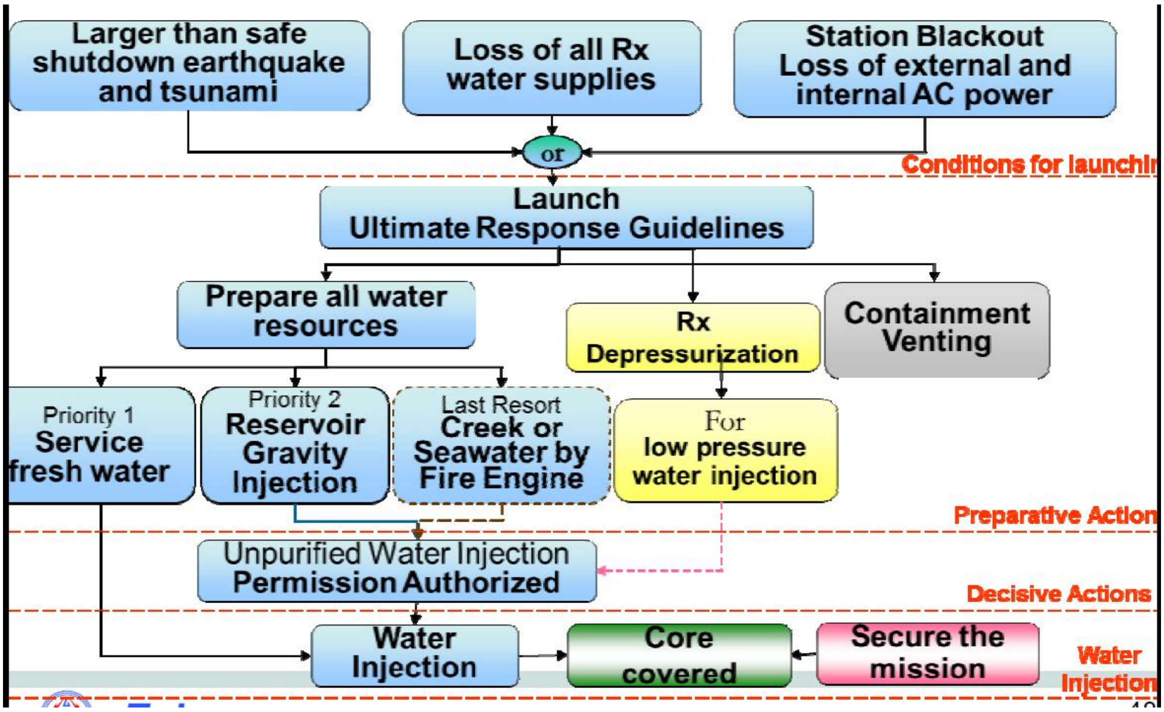


Figure 10.2 The Flow Chart of URG

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 (NSDC), 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. First of all, the real-time color-coded data of selected parameters from the Safety Parameter Display System (SPDS) 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 5 minutes at site boundaries of all NPPs. In addition, the area gamma radiation updated every 5 minutes for 46 Environmental Radiation Monitoring (ERM) stations located in the entire Taiwan area are also available. (See also Subsection 15.2.4.)

The TPC also offers adequate information on nuclear power (such as real-time information on NPP operations, environmental radiation monitor and so forth) so that the general public can learn more about the achievements and status of nuclear power safety. This would enable the public to cultivate a deeper understanding of the subject during their supervision of the operations.

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.

The AEC also takes part in some of the cooperative activities and training seminars sponsored by the OECD's Nuclear Energy Agency (OECD/NEA) 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 (NPT) 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.

The TPC organized 4 international symposiums in 2013 and invited renowned international figures to help the general public learn more about the facts of the Fukushima incident and relevant nuclear power development experiences around the world. There were also symposium activities in 2014 and 2015, and even in 2016.

There have been also quite the number public activities on nuclear safety education held by nuclear-related NGOs in recent years. The delivery of such information has helped to foster the public's confidence in nuclear power safety. More information about the transparency measures is given in Subsections 9.1.2 and 19.10.

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 Financial Requirements

According to the Nuclear Reactor Facilities Regulation Act of 2003 (Articles 5 and 6), the Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities of 2004 (Article 3), the Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities of 2004 as amended in 2005 (Article 14), and the Enforcement Rules of the Atomic Energy Act of 1976 as amended in 1996 (Article 15), the license applicant of a NPP 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 license (CL) (which was termed construction permit (CP) previously), an operating license, or the nuclear fuel license.

The Nuclear Damage Compensation Law of 1971 (as amended in 1977 and 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. According to this Law, the compensation liability is limited to a total amount of 4.2 billion New Taiwan dollars (NT$). After the Fukushima accident, amendment of the Nuclear Damage Compensation Law is now undergoing, which will increase the limit of compensation liability from the current 4.2 billion NT$ to an amount of 15 billion NT$.

11.1.2 Financial Resources of the Licensee

According to 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 CL of a nuclear reactor facility must be a legal company with assets more than 100 billion NT$.

The TPC, a state-owned public utility company, is the sole owner of NPPs 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. At the end of December 2015, the total TPC's assets were worth about 1,935 billion NT$ (~ 59 billion US$ at the exchange rate of NT$32.6 : US$1.0). Figure 11.1 shows the total TPC's assets in the past 5 years up to 2015.

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 of NPPs to protect the public health and safety. Thus, the Nuclear Reactor Facilities

Total Assets: NT$ (million)

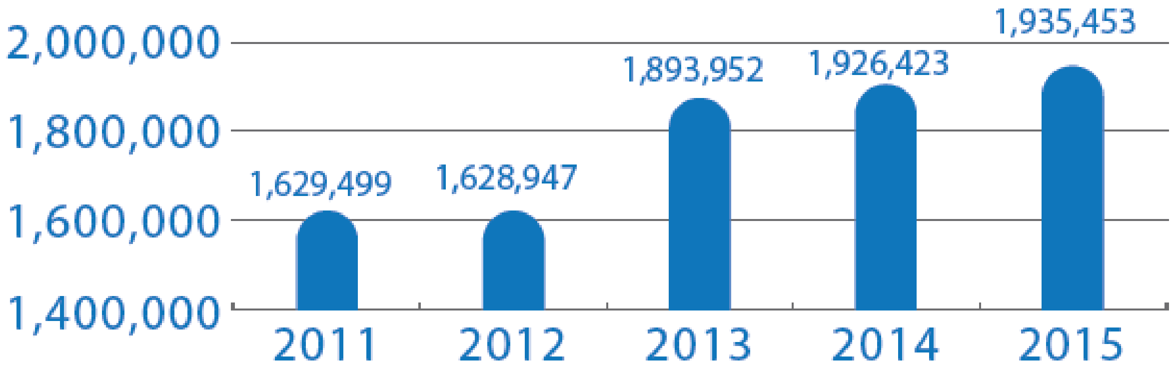


Figure 11.1 Total Assets of the Taiwan Power Company

Regulation Act requires the licensee to have sufficient financial resources to properly construct and safely operate the NPPs. Although there does not appear a direct 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 plant 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 NPP before sufficient funds have been accumulated could potentially hinder the implementation of the decommissioning, spent nuclear fuel (SNF) and the radwaste disposal programs of the plant.

11.1.3 Financing of Safety Improvements

The TPC has established a betterment plan for the safe operation and reliability improvement of each NPP and planned to secure the required research and development (R&D) 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 (WRNM) system, automatic voltage regulator (AVR) and power system stabilizer, hydrogen water chemistry (HWC) system, ultrasonic flow measurement system, emergency circulating water system, spent fuel pool (SFP) re-racking, and automatic seismic trip system (ASTS) for both BWR and PWR units. Thus, a significant investment has been made in the betterment plan of the nuclear units. In addition, implementation of Post-Fukushima issues as listed in the latter Table 14.3 will significantly help TPC's NPPs improve their safety.

Table 11.1 gives the total number of design change requests (DCR) resulted from the betterment programs of the three operating nuclear power stations in recent years. The TPC's budgets for R&D were about 165 million NT$ in 2011, 333 million NT$ in 2012, 239 million NT$ in 2013, and 295 million NT$ in 2014.

The AEC also performed necessary regulatory R&D as part of its Mid-Term and Long-Term Nuclear Energy Research and Development Programs for updating the regulatory actions to maintain the safe operation of NPPs and revising regulations for taking into consideration the state-of-the-art nuclear technology and the ever-increasing environmental requirements. To this end, the Atomic Energy Act of 1968 as amended in 1971 stipulates that the AEC should be responsible for funding the R&D programs to promote the nuclear science and technology. In recent years, the AEC's fiscal budgets for R&D were about 847 million NT$ in 2014, 846 million NT$ in 2015 and 856 million NT$ in 2016, representing about 28.9%, 28.7% and 29.3%, respectively, of its total fiscal budget.

Table 11.1 Number of Design Change Requests Completed in 2011~ 2014

	CS0	CS1	CS2	KS0	KS1	KS2	MS0	MS1	MS2
2011	12	43	40	33	12	57	19	28	70
2012	9	11	46	27	54	11	27	66	64
2013	10	37	9	22	45	54	21	58	20
2014	8	28	39	22	24	49	22	38	81

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

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 NPP shall establish a nuclear backend fund for the decommissioning of the nuclear installation and the treatment and final disposal of the SNFs and low-level radioactive wastes (LLRW). The TPC estimated 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 (CPI) and the international experiences. This fund has been collected on the basis of the amount of electricity generated by the NPPs since 1987. In the first two years, the sharing rate of electricity for the fund was set at NT$0.14 per kilowatt-hour (kWh) of electricity generated by the nuclear power and was gradually raised to NT$0.17/kWh in 1993 and then to NT$0.18/kWh in 1998. In 1999, "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 thus based on this Rule. The collection rate for the Nuclear Backend Fund was between NT$0.14 and NT$0.18 per kWh in the past. Currently the rate is NT$0.17/kWh or about 5.2 US mills/kWh (at an exchange rate of NT$32.6 vs US$1.0). At the end of August 2016, the net amount of the backend fund reached 281.38 billion NT$ (compared to 205.7 billion NT$ at the end of August 2010). The estimated total cost of the TPC's nuclear backend management programs is subject to a re-evaluation every 5 years or whenever necessary to be commensurate with the development status of the TPC's nuclear power programs, industrial technologies and government regulations. Under the scenario of operating each of the existing six nuclear power units for 40 years, the total nuclear backend cost was estimated to be about NT$335.3 billion.

An ad hoc committee, established under the Ministry of Economic Affairs (MOEA), 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.

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 (DNBM) at the TPC headquarters is responsible for the planning and implementation of the radioactive waste disposal programs and future decommissioning of the TPC's NPPs. The Radwaste Management Section of each NPP 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 NPPs.

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 as amended in 1997 requires that all NPP licensees purchase specified amounts of liability insurance at a maximum level of 4.2 billion NT$ 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 4.2 billion NT$. Should the amount received from the liability insurance or financial guarantee not suffice to cover the liability, the government shall loan the balance to the licensee to cover its complete liability, but only to the limited amount of 4.2 billion NT$. 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.

If a nuclear accident does occur, one important feature of the current Nuclear Damage Compensation Law as amended in 1997 is that 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 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.

However, after the Fukushima nuclear accident happened in March 2011, the Nuclear Damage Compensation Law as amended in 1997 is subject to amendment again in order to reflect the lessons learned from this accident and the Three Miles Island accident and thus provide better protection of the public. The followings are the major amendments proposed by the AEC for the Nuclear Damage Compensation Law to be amended:

- Limit of the total liability claims for one nuclear accident is to be raised from current 4.2 billion NT$ to 15 billion NT$ which is equivalent to about 0.3 billion Special Drawing Rights (SDR).
- Expiration limit required for a claimant to submit a compensation claim is extended from 3 years to 10 years since the date when he knew there was damage to him and who was the licensee responsible for the accident of the nuclear facility. Also, the expiration limit required to submit a compensation claim is extended from 10 years to 30 years since the date when the nuclear accident occurred
- Furthermore, the immunity of the licensee’s liability on nuclear accident caused by severe natural disasters is abandoned.

The above AEC proposed amendments on the Nuclear Damage Compensation Law was approved by the Executive Yuan on September 29, 2011 and submitted to the Legislative Yuan (Congress) for review on October 4, 2011. However, because of the policy of nuclear power phase-out by 2025 of the new administration since May 20, 2016, amendments on the Nuclear Damage Compensation Law are no more urgent. Thus, under the agreement of the Legislative Yuan, the proposal to amend this law was withdrawn on July 27, 2016.

11.2 Human Resources

11.2.1 Requirements for Personnel Qualification, Training and Retraining

Item 4 of Article 23 and item 3 of Article 26 of the Atomic Energy Act of 1968 (as amended in 1971) 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 2000s, the Nuclear Reactor Facilities Regulation Act (2003), the Ionizing Radiation Protection Act (IRPA) (2002), and the Nuclear Materials and Radioactive Waste Management Act (2002) as well as their Enforcement Rules were promulgated.

Article 5 of the Nuclear Reactor Facilities Regulation Act of 2003 stipulates that without a construction license (CL) 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 (OL) has been issued, can the facility be formally operated (Article 6). Furthermore, to operate a nuclear power reactor, all the control room operators must have licenses of reactor operator (RO) or senior reactor operator (SRO) issued by the AEC in advance (Article 11). The qualification requirements for ROs are stipulated in Articles 4 and 5 as well as Appendix 2 of the Regulations on Nuclear Reactor Operators' Licenses of 2004 as amended in December 2009.

Article 29 of 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). Article 7 of this Act further requires the licensee of these radiation related business must set up a radiation protection management organization or have the licensed radiation protection personnel in order to implement the radiation protection practice. Articles 30 and 31 of this Act requires that to handle radioactive materials or to manipulate 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. Article 14 stipulates that to do or participate in radiation-related activities, one must be at an age of 18 or over. For a pregnant woman employee working in the radiation control area, the licensee must re-evaluate her working condition and adjust her job arrangement if necessary. Requirements for setting up a radiation related facility or business are given in the Standards for Establishment of Radiation Protection Management Organizations and Radiation Protection Personnel of 2002. On the other hand, the qualification requirements for radiation protection personnel are stipulated in the Administrative Regulations for Radiation Protection Personnel of 2002 as amended in 2011, while those for operators are stipulated in the Administrative Regulations for Operators of Radioactive Material or Equipment Capable of Producing Ionizing Radiation of 2002 as amended in 2009 and Administrative Regulations for the Operators of Production Facilities of Radioactive Material of 2003.

Article 27 of the Nuclear Materials and Radioactive Waste Management Act of 2002 requires that to manipulate a radioactive waste (RW) treatment facility, the RW operator must be qualified. The qualification requirements for an operator to manipulate the radioactive waste treatment facility are stipulated in the Regulations on Radioactive Waste Operators of 2009.

11.2.1.1 Training Requirements for Regulatory Staff

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 NPP. 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 NPP inspectors and reviewers with the TPC's design change requests (DCRs) which had been implemented or were planned as the classroom discussion topics.

There are four kinds of the AEC's inspection personnel, including those for nuclear power plants (NPP), radiation protection, nuclear safeguards and emergency response, and radioactive materials management, respectively. Each kind of the AEC's inspection personnel has two categories of inspector licenses: Inspector and Senior Inspector. There is also the RO/SRO examiner responsible for giving the qualification examinations for the candidates of nuclear reactor operators.

As an illustration, the requirements for being a NPP inspector, senior inspector, or RO/SRO examiner are given in the following.

In May 2005, the AEC internally required its NPP inspectors and RO/SRO examiners to be licensed in order to strengthen their technical ability and further enhance the quality of inspections and reactor operator examinations. Qualification requirements (as amended in 2015) for being a candidate of the NPP 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 6 days, and
- Self-reading the assigned on-the-job training courses for an inspector.

On the other hand, qualification requirements (as amended in 2015) for one to become a candidate of the NPP 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 (QA) for 6 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.

The licenses of the inspectors and senior inspectors are effective for 6 years. 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 in order to be qualified for renewing their lisenses by 3 months before the lisenses fall due.

Qualification requirements 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.

11.2.1.2 Regulatory Requirements for Reactor Operators

11.2.1.2.1 Requirements for the Number of ROs On-Duty

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 NPP with twin units, which are also specified in the technical specifications (TS) 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 Subsection 9.1.1 of this report, both the SM and SL are required to have valid SRO licenses.

11.2.1.2.2 Qualification Requirements for the ROs

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

(1) Experience Requirements:

- At least 2-year working experience in a power station and among them at least 1 year in a NPP, and
- At least 6-month working experience in the NPP which he is applying for the RO license and among them at least 3 months on duty for operation.

(2) Training Requirements:

- At least 1-year training including no less than 3 months of simulator training, and
- Completion of the required RO training courses which must be no less than 300 hours.

(3) Physical Condition Requirements:

- Passing the physical health check-up required.

The educational pre-requirement 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:

- At least 2-year working experience in a NPP, and
- At least 6-month working experience in the NPP 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:

- At least 1-year training including no less than 3 months of simulator training, and
- Completion of the required reactor operator training courses which must be no less than 300 hours.

(3) Physical condition

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

An RO can also apply for an SRO license. In this case, he must have: (1) an RO license of the same NPP 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 an RO license, the candidate must pass the relevant regulatory examinations which will be held in two stages as shown in the following. Application for these regulatory examinations for the operator license must be submitted through the licensee of the NPP.

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 the subsequent on-the-job training for at least three months, an RO license can then be granted.

In summary, licenses of RO and SRO are issued to applicants who have engaged in the relevant fields with sufficient experience and successfully passed the examination administered by the AEC.

11.2.1.3 Regulatory Requirements for Radiation Protection Personnel

According to Article 2 of the Administrative Regulations for Radiation Protection Personnel of 2002 as amended in August 2011, the radiation protection personnel can be classified into two categories: the radiation protectors and the senior radiation protectors. Article 3 of that Regulation points out that to apply for the license of radiation protector, an applicant must fulfill at least one of the following qualification requirements:

- University or graduate school graduate from the radiation protection related departments, having passed the radiation protector examination and the subsequent on-the-job training for at least 3 months on radiation protection,
- Graduate from College of Science, Engineering, Agriculture or Medicine, having taken at least 6 credits of radiation protection related courses or passed the radiation protection personnel training for at least 108 hours, and then passed the radiation protector examination as well as the subsequent on-the-job training for at least 6 months on radiation protection, or
- High school graduate, having passed the radiation protection personnel training for at least 108 hours, and then passed the radiation protector examination as well as the subsequent on-the-job training for at least 9 months on radiation protection.

To apply for the license of senior radiation protector, an applicant must fulfill at least one of the following qualification requirements:

- Graduate from University of Science, Engineering, Medicine or Agriculture, having taken at least 8 credits of radiation protection related courses or passed the radiation protection personnel and advanced training for at least 144 hours, and then passed the senior radiation protector examination as well as the subsequent on-the-job training for at least 3 months on radiation protection,
- Graduate from College of Science, Engineering, Medicine or Agriculture, having taken at least 8 credits of radiation protection related courses or passed the radiation protection personnel and advanced training for at least 144 hours, and then passed the senior radiation protector examination as well as the subsequent on-the-job training for at least 6 months on radiation protection,
- Qualified radiation protector, having taken at least 8 credits of radiation protection related courses or passed the radiation protection personnel and advanced training for at least 144 hours, and then passed the senior radiation protector examination,
- Licensed radiation protector, during the period of validity of his license, having taken at least 2 credits of radiation protection related courses or passed the radiation protection personnel advanced training for at least 36 hours, and then passed the senior radiation protector examination, or
- Persons who passed one of the national examinations as listed in Article 3 of the Administrative Regulations for Radiation Protection Personnel as amended on August 31, 2011 and then passed the subsequent on-the-job training for at least 3 months on radiation protection.

11.2.1.4 Regulatory Requirements for Radioactive Waste Operators

According to Article 2 of the Regulations on Radioactive Waste Operators of 2009, operators of the radioactive waste (RW) treatment facility can be classified into two categories: the RW operators and the senior RW operators. To apply for the license of RW operator, an applicant must fulfill the following qualification requirements:

- High school graduate or the equivalent,
- Having passed the required RW operator training, and
- Having passed the RW operator examination by the regulatory body.

To apply for the license of senior RW operator, an applicant must fulfill the following qualification requirements:

- College graduate (or the equivalent), or qualified RW operator for at least 3 years,
- Having passed the required senior RW operator training, and
- Having passed the senior RW operator examination by the regulatory body.

11.2.1.5 Licensee's Training and Retraining 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, while 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. On the other hand, the Nuclear Materials and Radioactive Waste Management Act of 2002 requires that to manipulate a radioactive waste (RW) treatment facility, the RW operator must be qualified.

The nuclear related licenses and certificates as required above are summarized as follows:

- License for senior reactor operator (SRO),
- License for reactor operator (RO),
- Certificate for radiation protection personnel,
- Radiation safety certificate,
- Certificate for senior RW operator, and
- Certificate for RW operator.

The total number of nuclear related license (or certificate) holders (including RW operators) employed by the TPC to work in the NPPs as of August 2015 is 600, 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.

Table 11.2 Number of Nuclear-Related License (or Certificate) Holders Employed by the TPC (As of August 2015)

Type of License/Certificate	Chinshan	Kuosheng	Maanshan
Senior Reactor Operator	27	30	30
Reactor Operator	18	24	24

Radiation Protection Personnel	34	39	31
Radiation Safety	72	71	68
Senior RW Operator	10	16	7
RW Operator	35	30	34
Subtotal	196	210	194
Total	600		

The Nuclear Reactor Facilities Regulation Act and its Enforcement Rules stipulate that the licensee of a NPP shall provide its employees with the educational and training opportunities. Accordingly, the TPC provides its employees with the opportunity of refreshing the professional knowledge and up-to-date technology by the annual educational programs.

In general, the personnel technical training programs in the NPP 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.

11.2.1.5.1 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 RO 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 RO candidates for applying the licenses. Then, these candidates have to pass the AEC's RO license examinations to get their operator licenses. In accordance with the AEC regulations, a retraining program for the licensed ROs shall be developed and conducted by the NPP to assure that the licensed operators receive adequate, appropriate, and required training. To renew an RO license, the RO license holder must fulfill the regulatory requirements of RO retraining.

The RO/SRO 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 systems,
- Waste treatment systems, and
- Overall plant operations and accidental transient response.

(2) Simulator Operation Training

- 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 MCR panels,
- Responsibilities of the RO and SRO,
- Fuel management and control,

- Remote shutdown operation, and
- Administration control processes.

11.2.1.5.2 Licensed Reactor Operators Retraining Program

According to Article 10 of the Regulations on Nuclear Reactor Operators' Licenses as amended in December 2009, the licensee of a NPP should submit an RO retraining program to the AEC for approval and all its reactor operators must be retrained in accordance with the 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 as mentioned in the Appendix 4 of this Regulation 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 re-qualification tests including written and oral examinations in order to be legally qualified to continue their operation of the reactor.

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.

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 NPP has to submit the following documents about its reactor 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 contents such as classroom lectures and operation on simulators and on the plant site as described in the following. 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 Simulator and Plant Site

Training on simulator must be no less than 30 hours every year. The scope of the retraining courses about operations on simulator and plant site is as follows:

(a) Exercise on normal plant operations,

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

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

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

Details of the above retraining courses are up to each NPP to adjust. 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.

11.2.1.5.3 Re-qualification of Licensed Reactor Operators

As mentioned in the previous Subsection 11.2.1.5.2, the licensee of an NPP must submit a reactor operators' retraining program, based on the Appendix 4 of the Regulations on Nuclear Reactor Operators' Licenses as amended in December 2009, to the AEC for approval and all its reactor operators must be retrained in accordance with this approved retraining program and pass, at the end of the retraining, 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.

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, or
- 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.

11.2.1.5.4 Training for Non-Licensed Plant Technical Staff

Non-licensed plant staff includes:

- Non-licensed on-shift operators of plant system/equipment radwaste control and process system/equipment switchyard, pump house, gas turbine, etc.,
- Each category of maintenance and engineering support engineers such as mechanical maintenance engineers, electrical maintenance engineers, I&C engineers, nuclear engineers, chemistry and/or radiochemistry engineers, health physics engineers, quality assurance (QA) and quality control (QC) engineers, and computer engineers, and
- Each category of maintenance and engineering support technicians for maintenance and engineering support, such as mechanical maintenance technicians, electrical maintenance technicians, I&C technicians, chemistry/radiochemistry technicians, health physics technicians, and QA and QC technicians.

The training for non-licensed operators, engineers, and 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, QA requirements as well as industry operating experiences, and personnel applicable to their area of responsibility.

(3) BWR/PWR/ABWR System Technology Training

Depending on what type of reactor they are working in, 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 developed by the staffs who have completed the manufacturer's system technology training will be in Chinese.

11.2.1.5.5 Training for TPC's General Employees and Contractors' Personnel

All persons employed in the NPP and the people hired by the TPC contractors who need to access the NPP to do their job 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

11.2.1.5.6 Plant Simulator Training Center

In each TPC's NPP, there is a simulator training center mainly for training the reactor operators (RO). Each training center is equipped with a full-scope simulator and small-scale mock-ups. The simulator simulates the MCR of a NPP and provides a manipulation environment complying with the requirements for RO training and the simulator regulation ANSI/ANS-3.5: "Nuclear Power Plant Simulators for Use in Operator Training and Examination."

The simulator is able to simulate a variety of normal, abnormal and emergency situations in the nuclear unit operation and is used to support:

- Duplicating the control room panel,
- On-the-job training for operational personnel,
- RO (and SRO) licensing training,
- RO (and SRO) licensing test by AEC, and
- RO (and SRO) retraining.

11.2.1.5.7 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.

11.2.2 Human Resources of the Licensee

11.2.2.1 Manpower of the Taiwan Power Company

As mentioned in Subsection 9.1.1 (Figures 9.1 and 9.2), the TPC is composed of many departments of general management, operations and business and many other offices as well as four NPPs (three in operation and one in mothballs).

Up to mid-September 2015, the total number of employees in the TPC was 27,059. Among them, 2,794 were working in the nuclear sectors, including 573, 669, 598 and 478 employees working in CS, KS, MS and LM NPPS, respectively.

As shown in Figures 9.1 and 9.2, the nuclear-related organizations in the TPC headquarters include the Department of Nuclear Generation (DONG) (having a staff of 193), Department of Nuclear Safety (DNS) (staff of 99), Department of Nuclear Engineering (DNE) (94) and Department of Nuclear Back-end Management (DNBM) (85). 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 headquarters and is responsible for the training of the TPC employees.

The Nuclear Safety Committee (NSC) 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 NPP, there is a Station Operation Review Committee (SORC) organized to advise the Plant General Manager (PGM) on matters concerning nuclear safety.

In each NPP, as shown in Figure 9.3, 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, and one Simulator Training Center.

As shown in Table 11.2, the number of reactor operator license holders in each of the three operating NPPs is 45, 54, and 54 for CS, KS, and MS, respectively in August 2014.

11.2.2.2 Supports from Contractors in Emergency

To become qualified to work in a NPP, a contractor's employee must have the maintenance certificate of related equipments, pass the physical examination and be trained in advance on courses of security, industry safety, radiation protection, emergency planning, environmental protection, waste management, quality control (QC), etc. as mentioned in Subsection 11.2.1.5.5 for at least 3 hours.

The mechanical, electrical, I&C, and repairing maintenance groups of the TPC's NPP regularly invite the contractors to dispatch their qualified technical persons to participate in the regular plant maintenance or temporary emergent repair jobs. In case a nuclear accident occurs and if the Operation Support Center (OSC) does not have enough manpower to deal with the emergency, the contractors' technical persons will enter the plant to assist in resolving the case.

In addition to the above-mentioned regular O&M contractors, other contractors including, for example, the reactor vendors (GE and W), A/E consultants (e.g., Pacific Engineers & Constructors, Ltd. (PECL), Bechtel Group, E&C Engineering Corp. (E&C), Ebasco Services, Inc., etc), and the international nuclear organizations (INPO, WANO, etc.) may also provide technical supports for severe accident management if required.

11.2.2.3 On-site and Off-site Manpower Supports for Severe Accident Management

As to be described in Subsection 16.1.1.3, according to the possible severity of their impact the nuclear accidents can be divided into three categories, namely:

(1) alert event,

(2) site area emergency accident (SAEA), and

(3) general emergency accident (GEA).

When an accident of SAEA or GEA occurs, all onsite emergency organizations will be mobilized immediately and all emergency operation centers will be set up and operate at once in order to carry out the plant rescue mission. The emergency operation centers include the technical support center (TSC), operation support center (OSC), health physics center (HPC) and emergency public information center (EPIC).

In case the accident deteriorates and the plant staff itself is unable to handle the situation, off-site support will be needed for severe accident management. For each TPC's NPP, off-site manpower supports may come from the following:

- Nearby NPP,
- TPC headquarters,
- Army – Chemical Units of the Army for radiation decontamination,
- Local governments,
- Contractors,
- Etc.

11.2.3 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) is an important technical supporting manpower pool to assure the operational safety of the NPPs. At the end of 2014, INER has 764 formal employees, more than 300 contracted technical employees and more than 200 R&D Substitute Service (RDSS) military draftees. RDSS is one of the special skill alternative services (SSAS) in the Taiwanese conscription system of the military service.

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 undergraduate education of nuclear engineering is under the Department of Engineering and System Science and the graduate program of nuclear engineering is under the Institute of Nuclear Engineering and Science. The Department of Engineering and System Science, originally named the Department of Nuclear Engineering, was founded in 1964. The Department of Nuclear Engineering offered the Bachelor, Master, and Doctoral degrees in the engineering field. This 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 was renamed to its current name (Department of Engineering and System Science) since 1997. In response to the renaissance of nuclear power around the world in 2000s, the University established the Institute of Nuclear Engineering and Science in 2007 and the Interdisciplinary Program of Nuclear Science under the College of Nuclear Science in 2009. In the meantime, TPC has been offering scholarship to designated numbers of students of the College of Nuclear Science for years to attract more involvement of the younger generations in this field.

In addition to the NTHU, a non-profit association in the private sector, the Nuclear Science and Technology Association (NuSTA) of which the manpower comes mainly from the retired specialists of domestic nuclear organizations such as AEC, INER and TPC, has a nuclear technology training program to give courses and lectures of nuclear technologies for the public and industries that are interested in understanding nuclear energy or obtaining more about the up-to-date information of nuclear technology.

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 Overview of Human Factors and Organizational Issues for the Safety of NPPs

The interface between a mechanical system and its operators is usually known as the human factor. The human performance of plant staff depends on the persons' capabilities, limitations and attitudes, as well as on the quality of instructions and training provided. The goal of the human factor studies is to minimize the potential for human error by addressing factors that may adversely influence human performance.

Since the beginning of the project to construct a NPP, human performance is a very important factor in all phases of the plant lifecycle including design, commissioning, operation, maintenance, surveillance, modification, decommissioning and dismantling. More detailed descriptions of the consideration of human factors and man-machine interface (MMI) are provided in Subsection 18.3 of this report.

Quality management of plant staff is also highly important, because the way in which the work is organized, staffed, manned, supervised, evaluated and rewarded will determine the effectiveness, productivity and safety of the facility. The accidents at Three Mile Island (TMI), Chernobyl, and Fukushima NPPs were, in part, caused by human errors. One key issue is to strengthen the human and organizational aspects of nuclear safety in operating and regulatory organizations. In order to ensure a strict nuclear safety with the quality control of human performance in Taiwan, a nuclear safety management system was established to link the top-down processes of AEC, TPC headquarters, and NPPs as shown in Figure 12.1.

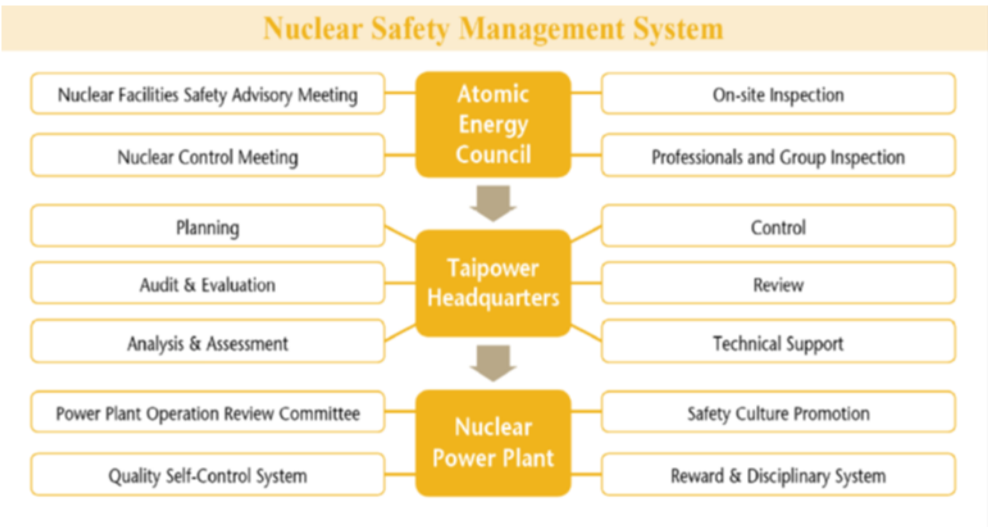


Figure 12.1 A Nuclear Safety Management System in Taiwan

12.2 Human Factors in the Design of NPPs and Subsequent Modifications

To prevent the occurrence of abnormal events due to human errors in the Lungmen NPPs, the AEC requires that human factors and man-machine interface (MMI) have to be taken into account in Chapter 18 : Human Factors Engineering (HFE) of the PSAR. At the construction stage of the Lungmen project, the AEC examines whether the human–system interface (HSI) meet the requirements for the “Human Factors Engineering Program Review Model,” in the NUREG-0711 (Rev. 0, 1994). The AEC also requires that the evaluation of the difference between NUREG-0711 (Rev.0) and NUREG-0711 (Rev.2, 2004) has to be illustrated in the FSAR. More information about the HSI can be found in Subsections 12.3(8). To operate a NPP, one must have a reactor operator (RO) license, and to become a licensed operator of a NPP, the trainee has to pass the AEC’s written examination and operating test for the RO. During plant operation, the AEC oversees the human performance through site inspections, safety reviews, and regulatory meetings. For those licensed operators, regular re-training courses are arranged to maintain their capabilities of dealing with normal and abnormal operating conditions.

To minimize the misjudgment of and the erroneous operation by reactor operators, 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 station has its own simulator). After passing all the examinations associated with these training courses, the trainee will become a candidate of the RO and be assigned to an operating shift of the related NPP for the on-the-job training under the guidance and supervision of a senior reactor operator (SRO).

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 Kuosheng and Maanshan NPPs 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.

12.3 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 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.

(2) The operating experiences, worldwide or domestic, are regarded as ones being

able to actively prevent the occurrence of repeated events in local 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.

(3) For the purpose of reducing human errors, ten preventive measures as shown below are reiterated in the TPC's safety culture enhancement program:

- 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 12.2.

(4) 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.

(5) 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.

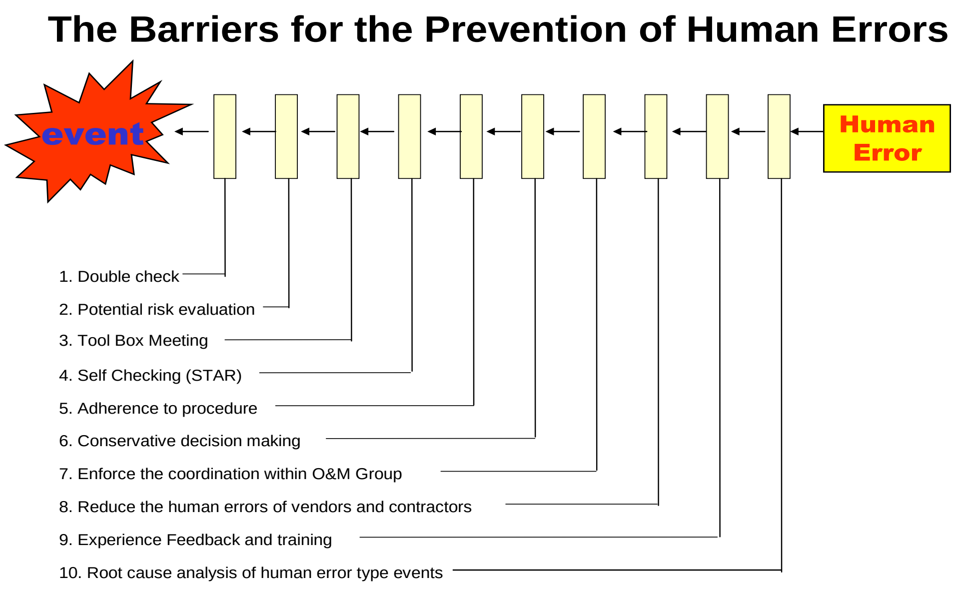


Figure 12.2 Barriers for the Prevention of Human Errors

(6) 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

(7) 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 2^{nd} time, his license will be terminated.

(8) Human–System Interface (HSI) design was adopted in the Lungmen Project. The Lungmen Project is the 4^{th} 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.

12.4 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.

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 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.

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.

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 learning 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.

12.5 Role of the Regulatory Body and the Facility Operator

12.5.1 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.

As mentioned in Subsection 11.2.1.1, the AEC also provides its staff with systematic training to maintain their professional capability up to date for meeting ever-increasing regulatory challenges. For example, advanced technology training courses are followed to enhance the regulatory capability of the inspectors. In addition, selected staff members are dispatched to countries with advanced nuclear technology for on-the-job training (OJT).

12.5.2 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 12.3 and 12.4.

12.6 Fukushima Lessons Learned

Lessons learned from the accidents at Fukushima showed that to some extent the accident was caused by human errors. In order to improve the synergy among technology, human factors, and organizational factors, the TPC has implemented several measures which will be discussed in the following.

12.6.1 Strengthening Nuclear Safety Organization and Culture

The nuclear safety organization and culture are strengthened by the following measures:

- Fortify employees' attitudes towards safety and cultivate good working habits among them to reduce operational negligence.

- Raise personnel training performance and operation skills.

  Training can help people acquire the skills, knowledge, and attitudes to make them competent in the safety aspects of their works and assigned tasks to diagnose plant upsets.

- Follow a rigorous nuclear quality-guarantee project to formulate implementation procedures and standards for each operation.

- Establish a strict safety and QC system and a safety management organization to ensure safety at every level.

More information about strengthening nuclear safety organization and culture is described in Subsections 6.2.1 and 10.1.3.

12.6.2 Routine Training in Response to Emergency Situations

Though viable safety measures have been considered in the design of the NPPs based on the risk management consideration, the TPC has established a "TPC Nuclear Reactors Emergency Response Plan" in accordance with the "Nuclear Accident Emergency Response Act" and related regulatory requirements. The implementations of the TPC's operating exercises and drills in response to emergency situations are illustrated as follows:

(1) Operating Exercises

In the operating exercise, the routine training in response to emergency situations was conducted, including:

- General training in the emergency plan: once every other year.
- Professional training in the emergency plan of special duties: once a year.

All NPPs shall provide training for the personnel assigned to the duties of responding to emergencies (Emergency Response Team) including initial training and annual training.

(2) Operating Drills

Every NPP has to conduct a drill once a year internally. On the other hand, the TPC, central and local governments, and the military, police, and medical units are all mobilized to participate in an annual national nuclear safety exercise that is held by turns at one of the three operating NPPs. In addition to the supervising agencies, TPC also invites professionals and scholars to form an evaluation group to assess exercises on each response measure to make the emergency response plan more effective.

More information about the NPP emergency drills and national nuclear emergency exercises is described in Subsections 16.2.2 and 19.8.4.

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 14 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, NQA-1a, NQA-2 and NQA-2a 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 Subsections 6.2.3.2 and 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 (CM) 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 CM plans. The contents of the CM plan include:

(1) Purpose,

(2) Scope,

(3) Framework of CM,

(4) Design bases,

(5) Design and design change control,

(6) Design document control,

(7) Evaluation of the CM process,

(8) Interfaces and integration,

(9) Working procedures for CM plan, and

(10) Computer codes for design and CM.

A CM 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 process 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 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 of the PSAR. Similarly, the "Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities" describe the required contents of 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 (LMNPP), 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 in accordance with the requirements of the AEC. These added contents are:

- Chapter 18 : Human Factors Engineering (HFE),
- 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 LMNPP 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 LMNPP. 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 LMNPP 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 the nuclear power station decommissioning; the transfer, the interim storage and the final disposal of spent fuels; 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

According to Article 15 of Regulations for the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities, licensees are required to update their final safety analysis reports periodically to incorporate the revised information and analyses. The update of FSAR submitted as part of the application for the license, to assure that the information included in the report contains the latest information developed. The description of the FSAR update is provided in Article 6 of this report. 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. Subsequent revisions must be filed 6 months after each refueling outage. If one FSAR is shared by multiple units, the reference date will be set by the second unit.

(2) Periodic Safety Review

According to Article 9 of the Nuclear Reactor Facilities Regulation Act and its enforcement rules, the owner of a nuclear reactor facility shall submit an "Integrated Safety Assessment Report" (ISAR) to AEC for review at 6 months before the end of every 10 years of operation. 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.

(3) 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. License renewal application shall be submitted at least five years and as early as 15 years before the expiration of its current license according to article 16 of Regulations for the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities. 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)
- Aging management program (AMP)
- Updated final safety analysis report (UFSAR)

● Update technical specifications, and

● Provide environment information such as impacts of license renewal on surface water, underground water, land use, ecology, etc.

Since the operation licenses of Units 1 and 2 of Chinshan NPP will expire on December 5, 2018 and July 15, 2019, respectively, the TPC has performed the necessary safety assessment and applied for license renewal in July 2009 for this NPP. However, TPC withdrew the application of license renewal for the Chinshan NPP on July 7, 2016. As a result, the safety evaluation programs regarding to license renewal were also stopped.

(4) PRA Update

The technology and application of PRA have been developed in this country for more than thirty 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 brief history of PRA development in Taiwan is shown in Table 14.1.

A technically acceptable PRA complied with the American Society of Mechanical Engineers (ASME) PRA standard will be performed by the requirement of USNRC RG 1.200 in the peer review process. 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. Table 14.2 lists a time table of the TPC’s development of PRA models based on ASME standards.

(5) 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

Table 14.1 A Brief History of PRA Development in Taiwan

Major PRA Projects (Main Sponsor)	Periods	Scope*							Application	Task Force Man-year
		L1-P	L1-SM	L1-TY	L1-FR	L1-FL	L1-SD	L2		
Kuosheng (AEC)	1983 • 1985	✓	✓	✓	✓	✓		✓	Base PRA model development	37 (4.5)#
Maanshan (AEC)	1985 • 1987	✓	✓	✓	✓	✓		✓ (1992)	Base PRA model development	27.5 (2.0)#
Chinshan (AEC)	1988 • 1990	✓	✓	✓	✓	✓		✓	Base PRA model development	34.5 (1.0)#
1st-3 (Taipower)	1994 • 1997	✓	✓	✓			✓		Few cases of justification of continued operation	52
2nd-3 (Taipower)	1997 • 2000	DCR & experience updates (~1999)			✓	✓	✓	CSET CPET	⋄TIRM (risk monitor) ⋄Fire models upgrade ⋄PRAM (PRA Maintenance) ⋄OLM	66
3rd-3 (Taipower)	2000 • 2003	DCR & experience updates (~2002)						LERF	⋄TIRM-2 ⋄FT Engine developed ⋄NEI-00-02 peer review ⋄Chinshan RIFA (Risk-informed Fire Wrapping Alternatives Analysis) ⋄RI-ISI pilot	66
4th-3 (Taipower)	2004 • 2007	DCR & experience updates (~2005)							⋄Lungmen base PRA model ⋄SDP tool (PRiSE) developed ⋄Follow-on NEI Peer Review ⋄Maintenance Rule Implementation	66
5th-3 (Taipower)	2008 • now	PRA model upgrade							⋄Upgrade to meet ASME/ANS PRA standards and peer review ⋄NFPA 805 transition (potential) ⋄Typhoon model enhancement	24

*P: Internal at-power; SM: Seismic; TY: Typhoon; FR: Internal Fire; FL: Internal Flood; SD: Shutdown; L1: Level-1; L2: Level-2
#:from US Consultant

Table 14.2 A Time Table of the TPC's Development of PRA Models Based on ASME Standards

Part	Requirements	Timetable	NPPs
2	Internal events at power PRA	2008.12-2011.12	CS, KS, MS
3	Internal flood at power PRA	2008.12-2011.12	CS, KS, MS
4	Fires at power PRA	2009.12-2012.11 2013.10-2017.10	CS KS, MS
5	Seismic events at power PRA	2012.07-2014.06	CS, KS, MS
6	Screening and conservative analysis of other external hazards at power	2012.12-2015.11	CS, KS, MS
7	High wind events at power PRA		
8	External flood events at power PRA		
9	Others external hazards at power PRA		
10	Seismic margin assessment at power	2011.07-2013.12	CS, KS, MS

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.

As mentioned previously in Subsection 6.1.2 of this report, the TPC has submitted the MUR application for the 3 operating NPPs during 2006 to 2008. The AEC reviewed these applications with reference to RIS 2002-03 and approved them during 2007 to 2009.

Furthermore, the TPC launched SPU projects for its Chinshan and Kuosheng NPPs after the MUR power uprate. The SPU application covers 3% original licensed thermal power (OLTP) uprate. The TPC propose a two-step power uprate. The first step is the 2% OLTP uprate, while the second step is the 1% OLTP additive uprate if steam dryer vibration monitoring (SDVM) is installed and used to monitor acoustic effect on steam dryer. Under the condition of the same reactor operating pressure, there are no major changes or modification to equipment. Chinshan NPP has increased its thermal power with 2% OLPT in 2012. Kuosheng NPP has also increased since 2014.

14.1.3 Design Changes

According to Article 13 of Reactor Facilities Regulation Act and Article 8 of Enforcement Rules for the Implementation of Nuclear Reactor Facilities Regulation, the following design or equipment change requires prior AEC's approval：

- Technical specifications change,
- 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 malfunction of SSCs 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.

For changes requiring prior AEC's approval, an assessment report should be submitted to the AEC. Then, the AEC will review the report and the changes 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 performance of changes fully meet the design requirements.

14.1.4 Phase-out Plan of Nuclear Power Plants

The TPC has established the nuclear power plant backend program. The work scope of this backend program includes:

(1) Decommissioning of the Nuclear Power Station

According to the Nuclear Reactor Facilities Regulation Act, the phase out of a NPP shall be completed in 25 years since its operation has been fully terminated, including the demolition of the facilities. Three years before the permanent shut-down of nuclear reactor facilities, operators must propose a decommissioning plan. In accordance with this law, TPC has begun planning for the decommissioning of Nuclear Power Plant No. 1, Chinshan NPP Units 1 and 2, since the operation of Unit 1 is to be terminated in 2018. The decommissioning plan of Chinshan NPP was submitted to the AEC in November 2015.

The TPC has mapped out a preliminary plan for decommissioning a NPP, which will perform the tasks in five stages:

(i) Preliminary operations before decommissioning, including

- Preliminary investigation on the history and specific features of the site, decommissioning strategies and operations research,
- Preparation of the work plans and decommissioning plans (including the environmental impact assessment report),
- Presentation of the topics and documents for approval.

(ii) Transitional period for the shutdown of machines

(iii) Demolition of the plant

(iv) Radiation detection in the perimeters

(v) Site recovery

The phase out of the Chinshan NPP Units 1 and 2 is complicated and the time frame is quite tight, since the operation of Unit 1 is to be terminated in 2018 while the TPC has no prior experience in phasing out a NPP and a comprehensive legal framework for the task has yet to be established in Taiwan. As such, the TPC has established a task force for the mission and at the preliminary stage, cross-function operations will be adopted to carry out the duties. Presently, the TPC has taken part in the phasing out project headed by the EPRI, which will provide relevant technical services (including visiting the plant for phasing out) and consultations. Besides, the TPC has also participated in the Co-operative Program on Decommissioning (CPD) under the Organization for Economic Cooperation and Development (OECD) as an observing member and is working to become an official member. In the future, the organization shall be subject to adjustment as needed.

(2) Transfer, Interim Storage and Final Disposal of Spent Fuels as well as Final Disposal of Low Level Waste from Plant Operation and Decommissioning

The TPC applies a 3-stage strategy for the management of spent nuclear fuel (SNF) that is applied internationally, namely, the pool storage, dry cask storage and final disposal. All the spent fuels from the three operating NPPs are currently stored in the spent fuel pools (SFP) 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 SFPs of the 1st and 2nd NPPs in Taiwan will be used up in the near future.

As the spent nuclear fuel pool storage facilities in the 1st and 2nd NPPs cannot accommodate the SNF produced over a 40-year period of operations by each reactor, the TPC is currently planning to construct dry storage facilities to enable each power plant to have sufficient storage facilities before the SNF (highly radioactive wastes) is sent to final disposal sites.

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 1st 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 with 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. More information about the ISFSI in Taiwan can be found in Subsection 19.9.2.

Taking reference of widely adopted international measures, the TPC will adopt deep geological disposal methods for its SNF final disposal. Currently, the TPC is undertaking the tasks of investigating and evaluating the characteristics of the potential host rocks.

14.2 Verification by Analysis, Surveillance, Testing and Inspection

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 NPP:

- 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 team inspection to assure the quality of maintenance works.
- Unannounced inspections are performed without pre-notice to test the alertness of plant operators.
- Taskforce inspection is conducted for items that require special professional technology to perform the inspection, in which experts outside the AEC are invited to join the inspection team. The scope of this inspection usually includes all special 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, A/E Company, and the Lungmen site office are inspected with reference to 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 (NDE), 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 (HSI) 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 (HFE) 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 I&C 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 of the TPC 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 QA 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 found 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 also 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 Plants.

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 ranges 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.

14.3 Fukushima Lessons Learned

14.3.1 Comprehensive Safety Assessment of Nuclear Power Generation

As mentioned in Subsection 6.4.1, after the Japan's Fukushima Daiichi Accident, the AEC thoroughly reviewed the lessons learned from Fukushima accident and proposed the "Programs for Safety Reassessment" which was approved by the Executive Yuan on April 19, 2011. The purpose of this safety reassessment was to substantially improve the safety margin of the NPPs by a complete planning and review of the plants' resistance to earthquakes and tsunamis and by improving their rescue capacity in terms of power sources, cooling water sources, the SFP cooling and the integration of resources and preparation.

The AEC has requested the TPC to reevaluate its capability to cope with extreme natural disasters including earthquakes, tsunamis, extreme rainfalls and mudslides, and to take possible countermeasures. The reassessment program comprised of three parts: (1) nuclear safety assurance, (2) radiation protection, and (3) emergency response and preparedness, which were implemented in two stages: near-term (by June 2011) and mid-term (by December 2011). The AEC completed its review of the TPC's safety reassessment reports and issued the following:

- "Preliminary Assessment Report of Nuclear Safety" in May 2011.
- "The Near-Term Overall Safety Assessment Report for Nuclear Power Plants in Taiwan in Response to the Lessons Learned from Fukushima Daiichi Accident" in October 2011.
- "The Overall Safety Assessment Report for Nuclear Power Plants in Taiwan in Response to the Lessons Learned from Fukushima Daiichi Accident" in August 2012, which included the near-term and mid-term reassessment results.
- Regulatory orders in November 2012.

Based on the requirements issued by the AEC after the safety reassessment, the TPC's three operating NPPs have proposed 96 enhancement plans. At the end of 2014, 95 of these plans had been completed except the seismic hazard reevaluation, including the geological survey and the SSHAC level 3 process, which is still in progress. These enhancement plans could be classified into the following 4 areas:

(1) Enhancement of earthquake-resistant capabilities

(2) Enhancement of tsunami/flooding-protection capabilities

(3) Enhancement of event mitigation capabilities including:

  - Backup power supply
  - Water resources and injection
  - SFP cooling
  - Resources preparedness

(4) Establishment of ultimate response guidelines (URG)

14.3.2 Stress Tests and Independent Peer Review

The AEC requested the TPC to perform the stress test to identify the "cliff-edge" effects based on the safety enhancement requirement of comparable NPPs in EU. As mentioned in Subsections 6.4.1 and 10.1.5.2, the scope of the stress test of Taiwan's NPPs to be carried out was in accordance with the ENSREG stress test specifications. The TPC completed a series of stress tests for the operating NPPs to confirm that the general assessment has helped to upgrade the capacity to protect the plant against hazard and to minimize the damage. The licensee's final stress tests reports of the 3 operating NPPs were submitted by the TPC to AEC in March 2012. The stress test report of the LMNPP was completed on April 27, 2012. The milestones associated with the safety reevaluation and stress tests for NPPs performed in Taiwan after the Fukushima accident are given in Table 6.7.

The AEC invited Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD) to identify experts who could conduct an independent peer review of the Taiwan's stress tests for the 3 operating NPPs. The NEA's independent peer review team visited Taiwan between March 4 and 20, 2013 and paid a site visit to Kuosheng NPP.

The AEC also invited the EU to set up an independent peer review team for the stress tests on TPC's 4 existing NPPs. The peer review team visited Taiwan between September 23 and October 3, 2013, including two site visits to Maanshan and Lungmen NPPs.

14.3.3 Issuance of Regulatory Orders

After the post-Fukushima safety reevaluation, stress tests, and independent peer review, the AEC issued the following regulatory orders/requirements to TPC to enhance the capability of NPPs to cope with extreme natural disasters, while the TPC may submit alternative plans, subject to AEC approval, to provide the equivalent function to comply with the requirements of regulatory orders:

(1) Adopt the conclusions of the following USNRC NTTF Report Tier 1 Recommendations:

- 2.1 Seismic and flood hazard reevaluations
- 2.3 Seismic and flood walkdowns
- 4.1 Station blackout (SBO) regulatory action
- 4.2 Equipment covered under 10 CFR 50.54(hh)(2)
- 5.1 Reliable hardened vents for Mark I and Mark II containments
- 7.1 SFP instrumentation
- 8 Strengthening and integration of EOPs, SAMGs, and EDMGs (& URGs)
- 9.3 Emergency preparedness regulatory actions (staffing and communications)

(2) Follow-up the Tier 2 & Tier 3 Recommendations by USNRC NTTF

(3) Follow-up the ENSREG's action plans from EU stress tests

(4) Perform special countermeasures for issues related to the seismic, tsunami, and SBO by referring to the good international practices as follows:

- To conduct survey on the newly found faults of Sanchiao and Hengchun near NPPs in October 2013.
- To install additional seismic instrumentation for monitoring and system identification

- To re-evaluate the hazard by state-of-the-art methodology and incorporate the new findings
- To simulate the mechanism of seismic and tsunami hazards and the resulting risks
- To enhance the water tightness of buildings (or build seawall or tidal barrier) to the level 6 meters above current licensing bases (CLB)
- To enhance the structure of non-seismic qualified TSC
- To build a seismic isolation TSC building

(5) Give additional considerations for the SBO rule as follows:

- To include seismic, tsunami, salt fog and landslides damage
- To consider specific natural events with high hazard
- To evaluate the capability to recover offsite power in 2 hours
- To study the initiating event frequency of LOOP
- To consider the fact that Taiwan is a north-south elongated island surrounded by the sea with isolated grid and without backup

(6) Perform countermeasures for SBO as follows:

- To enhance emergency DC power supply -- Extend the SBO coping time to at least 24 hours after the unnecessary loads are isolated.
- To install an additional seismic qualified gas-cooled EDG (the 6th EDG)
- To install an alternate UHS

(7) Install passive autocatalytic recombiners (PAR) to prevent hydrogen explosions

(8) Enhance the water-tight capabilities for the fire doors of essential electrical equipment rooms

(9) Enhance the seismic resistance for the fire brigade buildings to cope with BDBE

(10) Improve the seismic resistance of raw water reservoir and consider the installation of the impermeable liner

(11) Improve the reliability of offsite power supplies

(12) Consider the RCP seal LOCA issue of PWR plant

More information about the post-Fukushima regulatory requirements/orders is given in Subsections 6.4.2 and Annex 2.

14.3.4 Status of Implementation

The implementation status of Post-Fukushima orders in Taiwan’s NPPs was listed in Table 14.3.

After reviewing the Fukushima accident, TPC has assessed its NPPs and determined the following specific protections (as shown in Figure 14.1) in order to offer adequate resistance against complex disasters like earthquakes and tsunamis:

(1) Enclose the crucial sea water pumps in buildings,

(2) Locate the emergency diesel power generator at surface elevation,

(3) Make pneumatic cooling type diesel power generator supplying the backup power,

(4) Make pneumatic cooling type turbine power generator offering the backup power,

(5) Use raw water reservoirs relying on gravity for injecting water into the RPV,

(6) Add flood barriers to the Chinshan , Kuosheng, Maanshan and Lungmen NPPs and add tsunami embankments to the Chinshan , Kuosheng, Maanshan and Lungmen NPPs, and

(7) Execute the URG action for injecting water into the RPV.

Table 14.3 Implementation Status of Post-Fukushima Orders in Taiwan's NPPs

#	Name	Keywords	Current Status		
			CS[1]	KS	MS
1	XX-JLD-10101	seismic hazard (NTTF 2.1)	in progress	in progress	in progress
2	XX-JLD-10102	flood hazard (NTTF 2.1)	in progress	in progress	in progress
3	XX-JLD-10103	seismic and flooding risk	completed	completed	completed
4	XX-JLD-10104	DBT + 6m	in progress	in progress	in progress
5	XX-JLD-10105	EE walkdowns (NTTF 2.3)	completed	completed	completed
6	XX-JLD-10106	SBO rule (NTTF 4.1)	Tracking NRC rulemaking actions	Tracking NRC rulemaking actions	Tracking NRC rulemaking actions
7	XX-JLD-10107	2 EDGs always	completed	completed	completed
8	XX-JLD-10108	DC 8h, 24h load shedding	completed	completed AEC's RAIs[2]	completed AEC's RAIs
9	XX-JLD-10109	SBO coping 24h	completed AEC's RAIs	completed AEC's RAIs	completed AEC's RAIs
10	XX-JLD-10110	6th (8th) EDG/GT	completed	completed AEC's RAIs	completed AEC's RAIs
11	XX-JLD-10111	Alternate UHS	in progress	in progress	in progress
12	XX-JLD-10112	B.5.b	completed AEC's RAIs	completed	completed
13	XX-JLD-10113	equipment SBO (NTTF 4.2)	completed AEC's RAIs	completed AEC's RAIs	completed AEC's RAIs

14	XX-JLD-10114	hardened vent + filtered venting (NTTF 5.1)	in progress	in progress	in progress
15	XX-JLD-10115	SFP instrumentation (NTTF 7.1)	completed AEC's RAIs	completed AEC's RAIs	completed AEC's RAIs,
16	XX-JLD-10116	procedures incl. URG (NTTF 8)	Tracking NRC rulemaking actions	Tracking NRC rulemaking actions	Tracking NRC rulemaking actions
17	XX-JLD-10117	volcanic PRA	in progress	in progress	in progress
18	XX-JLD-10118	water-tight fire doors for essential equip.	in progress	in progress	in progress
19	XX-JLD-10119	BDBE resistance fire brigade bdg	Alternatives completed	Alternatives completed	Alternatives completed AEC's RAIs,
20	XX-JLD-10120	reliability offsite power	in progress	in progress	in progress
21	XX-JLD-10121	Seismic raw water res. +impermeable liner	in progress	in progress	in progress
22	XX-JLD-10122	PARs	in progress	in progress	in progress
23	CS-JLD-101101	equal to 0.4g	in progress	-	-
24	MS-JLD-101301	RCP seal LOCA	-	-	in progress
25	XX-JLD-10201	fault displacement analysis	in progress	in progress	in progress
26	XX-JLD-10202	interface post-earthquake and post-tsunami proc.	completed	completed	completed
27	XX-JLD-10203	event combination flooding and natural EE	completed	completed	completed
28	XX-JLD-10204	PMP with topo. maps	completed	completed	completed
29	XX-JLD-10301	landslides assessment	in progress	in progress	in progress
30	XX-JLD-10302	post-seismic inspection on non-seismic category I SSCs	completed AEC's RAIs	completed AEC's RAIs	completed AEC's RAIs
31	XX-JLD-10303	installation of	in progress	in progress	in progress

		closed cooling water loops			
32	XX-JLD-10304	BWR RPV depressurization	in progress	in progress	-
33	XX-JLD-10305	habitability MCR	in progress	in progress	in progress
34	XX-JLD-10306	multi-unit and multi-site accidents	completed	completed	completed
35	XX-JLD-10307	resistance of the plant site infrastructure	in progress	in progress	in progress
36	HQ-JLD-10201	local seismic network small earthquakes	completed AEC's RAIs	completed AEC's RAIs	completed AEC's RAIs
37	HQ-JLD-1013001	EPZ	completed	completed	completed
38	HQ-JLD-1013002	communication for EP (NTTF 9.3)	completed	completed	completed
39	XX-JLD-10104 (DNT)	seismic reinforced TSC	completed	completed	completed
40	XX-JLD-1013003	new seismically isolated TSC	in progress	in progress	in progress
41	XX-JLD-1013004	staffing for EP (NTTF 9.3)	in progress	in progress	in progress
42	RL-JLD-1012042	40 mobile detection equipment	completed	completed	completed
43	RL-JLD-1012043	13 radiation monitor, stations in EPZ	completed	completed	completed
44	RL-JLD- 1012044	vehicles for mobile detection equipment	completed	completed	completed

[1] The two reactors at CSNPP will be permanently shut down in 2018 and 2019, respectively. Therefore, TPC might request to rescind the AEC's order of safety enhancement.

[2] AEC's RAIs : TPC has completed safety enhancement, but some requests for additional information (RAIs) issued by AEC have not been resolved.

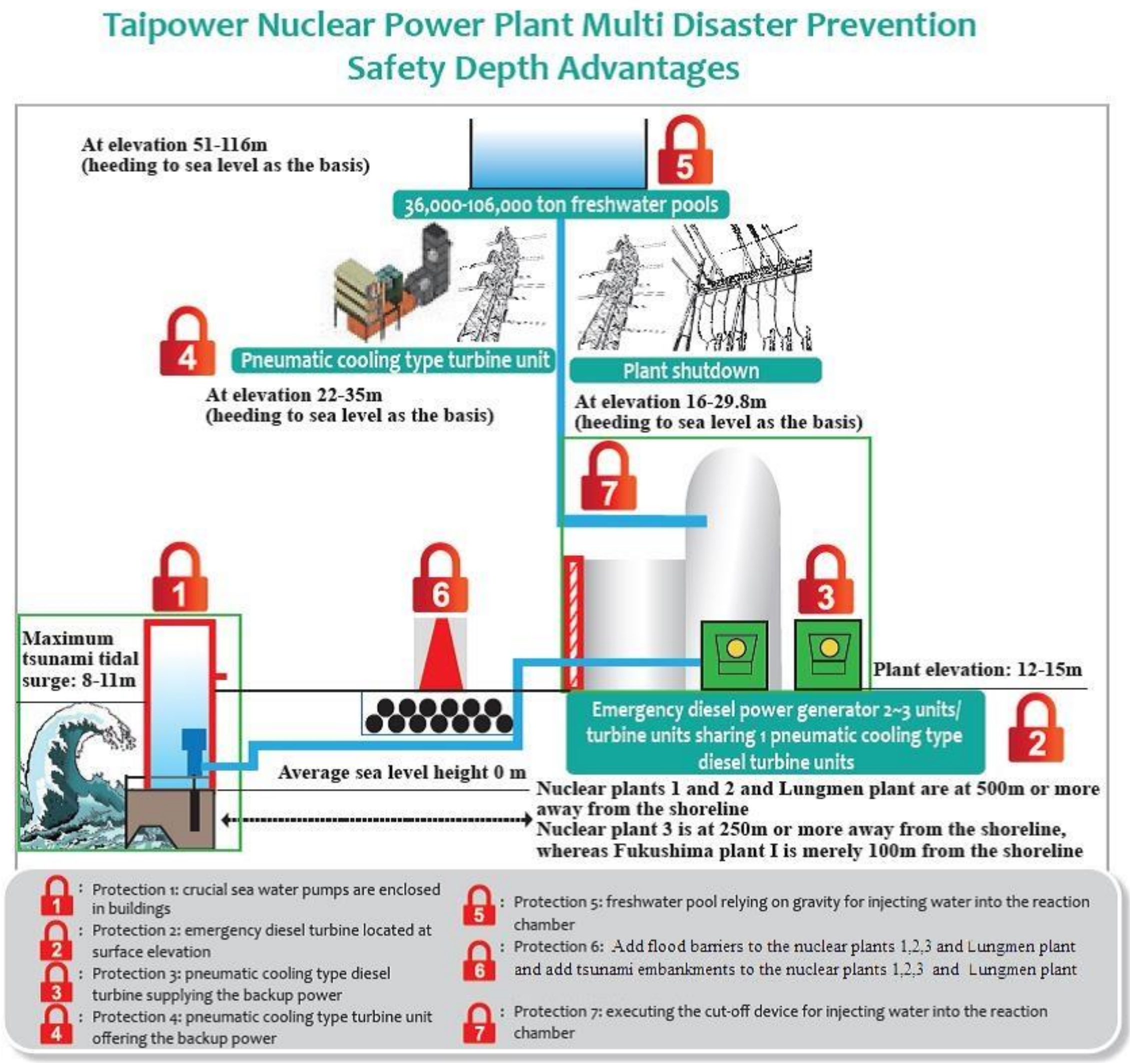


Figure 14.1 The Multi Disaster Prevention of TPC Nuclear Power Plants

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 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 Laws, 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. On December 22, 2010, an additional daughter regulation was promulgated, namely "Regulations for the Annual Detection Items of Radioactive Materials and Equipment or Facilities Capable of Producing Ionizing Radiation."

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 23 Regulations prescribing 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.

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 - Lens of the Eye - Skin, Hands, and Feet	 150 mSv 500 mSv	 15 mSv 50 mSv

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 collective dose distribution for different radiation work categories and the collective doses for the employees in three NPPs from 2009 to 2014 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. 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 50,000 workers associated with the occupational radiation exposure in Taiwan. The Atomic Energy Council has established 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 thermo-luminescent 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.

Table 15.2 Collective Dose in Different Radiation Work Category

year	2009	2010	2011	2012	2013	2014
(Unit: Man-Sievert)						
TPC	7.447	7.787	6.786	6.307	6.371	6.686
Medical	0.429	0.488	0.505	0.622	0.815	0.789
Industrial	0.768	0.767	0.684	0.665	0.615	0.815
NRM*	0.000	0.000	0.000	0.000	0.000	0.000
Others	0.203	0.152	0.177	0.183	0.192	0.302
Total	8.847	9.914	8.152	7.777	7.993	8.592
(Unit: Man-Year)						
Total Employee	42,966	44,607	46,545	48,225	48,621	50,438

* NRM — Natural Radioactive Material

Table 15.3 Collective Dose in Each NPP of the TPC

Unit: Man-Sievert

Year	2009	2010	2011	2012	2013	2014
Chinshan	2.076	1.904	2.885	1.966	1.937	3.187
Kuosheng	3.048	4.144	2.294	2.599	3.632	2.562
Maanshan	2.011	1.068	1.096	1.741	0.802	0.937
Total	7.135	7.116	6.275	6.306	6.371	6.686

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 production 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 (mSv) in five consecutive years defined by the AEC, in which the first five year cycle is year 2003 through 2007, the second five year cycle is 2008 through 2012, and the current cycle is 2013 through 2017. 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 1 mSv 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 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.

Based on "Regulations on the Design of the Environmental Radiation Dose of a Nuclear Power Plant" (promulgated on January 8, 1990), the dose constraints to members of the public for gaseous effluents are as follows:

(1) Radioactive Inert Gas

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

(2) Radioactive Iodine, Tritium and Particulates (half-life > 8 days)

- Equivalent dose in organ from radioactive iodine, tritium 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 including 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 2008 to 2014 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.

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]
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}
2010	1.97×10^{-6}	8.66×10^{-5}	1.43×10^{-4}	4.16×10^{-4}
2011	2.86×10^{-6}	3.10×10^{-5}	6.66×10^{-5}	1.61×10^{-4}
2012	1.12×10^{-6}	6.92×10^{-5}	1.17×10^{-4}	3.29×10^{-4}
2013	2.36×10^{-6}	7.59×10^{-5}	6.87×10^{-5}	1.61×10^{-4}
2014	2.48×10^{-6}	1.17×10^{-4}	3.45×10^{-5}	8.75×10^{-5}

Note: 1. From external exposure of noble gas (dose constraint 0.05 mSv/yr-unit).
2. Equivalent dose of organ from radioactive iodine and particulate (dose constraint 0.15 mSv/yr-unit).
3. Dose constraint 0.03 mSv/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]
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}
2010	7.28×10^{-4}	3.71×10^{-3}	3.90×10^{-5}	1.04×10^{-4}
2011	7.45×10^{-4}	3.61×10^{-3}	3.50×10^{-5}	9.50×10^{-5}
2012	8.81×10^{-4}	9.03×10^{-3}	5.82×10^{-5}	2.68×10^{-4}
2013	2.80×10^{-3}	1.20×10^{-2}	4.04×10^{-5}	4.30×10^{-4}
2014	4.92×10^{-3}	5.94×10^{-3}	7.54×10^{-5}	2.96×10^{-4}

Note: 1. From external exposure of noble gas (dose constraint 0.05 mSv/yr-unit).
2. Equivalent dose of organ from radioactive iodine and particulate (dose constraint 0.15 mSv/yr-unit).
3. Dose constraint 0.03 mSv/yr-unit.
4. Equivalent dose of organ (dose constraint 0.1 mSv/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]
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}
2010	2.67×10^{-4}	8.03×10^{-4}	2.24×10^{-5}	2.24×10^{-5}
2011	2.66×10^{-4}	7.42×10^{-4}	2.24×10^{-5}	2.24×10^{-5}
2012	2.19×10^{-4}	5.36×10^{-4}	1.02×10^{-5}	5.15×10^{-4}
2013	3.28×10^{-4}	2.34×10^{-4}	9.34×10^{-6}	9.34×10^{-6}
2014	2.14×10^{-4}	1.63×10^{-4}	1.03×10^{-5}	1.03×10^{-5}

Note: 1. From external exposure of noble gas (dose constraint 0.05 mSv/yr-unit).
2. Equivalent dose of organ from radioactive iodine and particulate (dose constraint 0.15 mSv/yr-unit).
3. Dose constraint 0.03 mSv/yr-unit.
4. Equivalent dose of organ (dose constraint 0.1 mSv/yr-unit).

* for two units.

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 52 environmental radiation monitors installed within the radius of 2.0 km of the four NPPs in Taiwan. Among these monitors, 14 are for Chinshan NPP, 14 for Kuosheng NPP, 12 for Maanshan NPP, and 14 for Lungmen NPP, with 2 of them being shared by both Chinshan and Kuosheng NPPs.

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.

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 Stations	45	36	32	Gamma Dose Rate/ Quarterly
HPIC Stations	7	7	5	Gamma Dose Rate /hr
Air:				
Particulates Stations	16	11	16	Gross β, γ Spectrum[1]/ Weekly, γ Spectrum/Quarterly, Sr-89,90[2]
Iodine Stations	16	11	16	I-131/Weekly
Fallout	1	1	1	γ Spectrum / Monthly
Water:				
Sea Water	9	9	10	γ Spectrum[3], H-3[3]/ Monthly, Sr-89,90[2]
Drinking Water	7	6	7	γ Spectrum, H-3/ Quarterly, Sr-89,90[2], I-131[4]
River Water	2	4	2	γ Spectrum, H-3/ Quarterly, Sr-89,90[2]
Pond Water	5	3	3	γ Spectrum, H-3/ Quarterly, Sr-89,90[2]
Ground Water	2	3	2	γ Spectrum, H-3/ Quarterly, Sr-89,90[2]

Precipitation I	2	2	3	γ Spectrum / Monthly, H-3/ Quarterly, Sr-89,90[2]
Precipitation II	2	2	3	γ Spectrum, H-3[7]
Agriculture & Marine Products:				
Milk: Cow/Goat	—	—	1	I-131, γ Spectrum / Quarterly, Sr-89,90[2]
Rice	2	3	3	γ Spectrum / Semiannually, Sr-89,90[2]
Vegetables	6	5	5	I-131, γ Spectrum / Semiannually, Sr-89,90[2]
Tea	5	—	—	γ Spectrum / Semiannually, Sr-89,90[2]
Fruits	2	2	1	γ Spectrum / Annually, Sr-89,90[2]
Vegetables (Root)	3	3	2	γ Spectrum / Annually, Sr-89,90[2]
Sweet Potato	1	1	—	γ Spectrum / Annually, Sr-89,90[2]
Vegetable Stem[5]	1	1	1	γ Spectrum / Annually, Sr-89,90[2]
Poultry	3	3	3	γ Spectrum / Semiannually, Sr-89,90[2]
Seaweed	2	2	2	I-131, γ Spectrum / Annually, Sr-89,90[2]
Sea Fish & Shellfish	5	5	6	γ Spectrum / Quarterly, Sr-89,90[2]
Index Organism:				
Acacia (Land)	1	1	1	γ Spectrum / Monthly
Algae (Sea)	1	1	1	I-131, γ Spectrum / Annually, Sr-89,90[2]
Land & Coast:				
Beach Sand	9	12	11	γ Spectrum / Quarterly
Soil	14	14	11	γ Spectrum / Semiannually
Sea Sediment	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.

To conform with the regulatory requirements and ensure the safety of the public and the environment, TPC conducted the radiological environmental monitoring programs based on the site characteristics. There are three exposure pathways (i.e., inhalation, ingestion, and direct radiation) that are routinely monitored. Samples of various environmental media are obtained to meet the objectives. Sampling locations were selected with consideration given to site meteorology, hydrology, demography, agricultural products, lifestyle and the land-use.

Sampling locations were divided into two classes, indicator and control. Control stations were at locations considered to be unaffected by the operations of NPPs. All others were considered as the indicator locations and may be potentially affected by the Nuclear Power Station. The monitoring area for Chinshan and Kuoshen includes New Taipei, Keelung, and Ilan. For Mananshan Nuclear Power Station, Pingtung is considered.

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. Up to September 2016, a total of 46 Environmental Radiation Monitoring (ERM) stations have been established in Taiwan, Kinmen island, Matzu island, Penghu island and Lan-yu island. The RMC has conducted annually national and international inter-laboratory comparisons on environmental radioactivity measurements for quality control. (See also Subsection 10.3.)

15.3 Fukushima Lessons Learned

15.3.1 Radiological Protection Guideline Following a Nuclear Accident

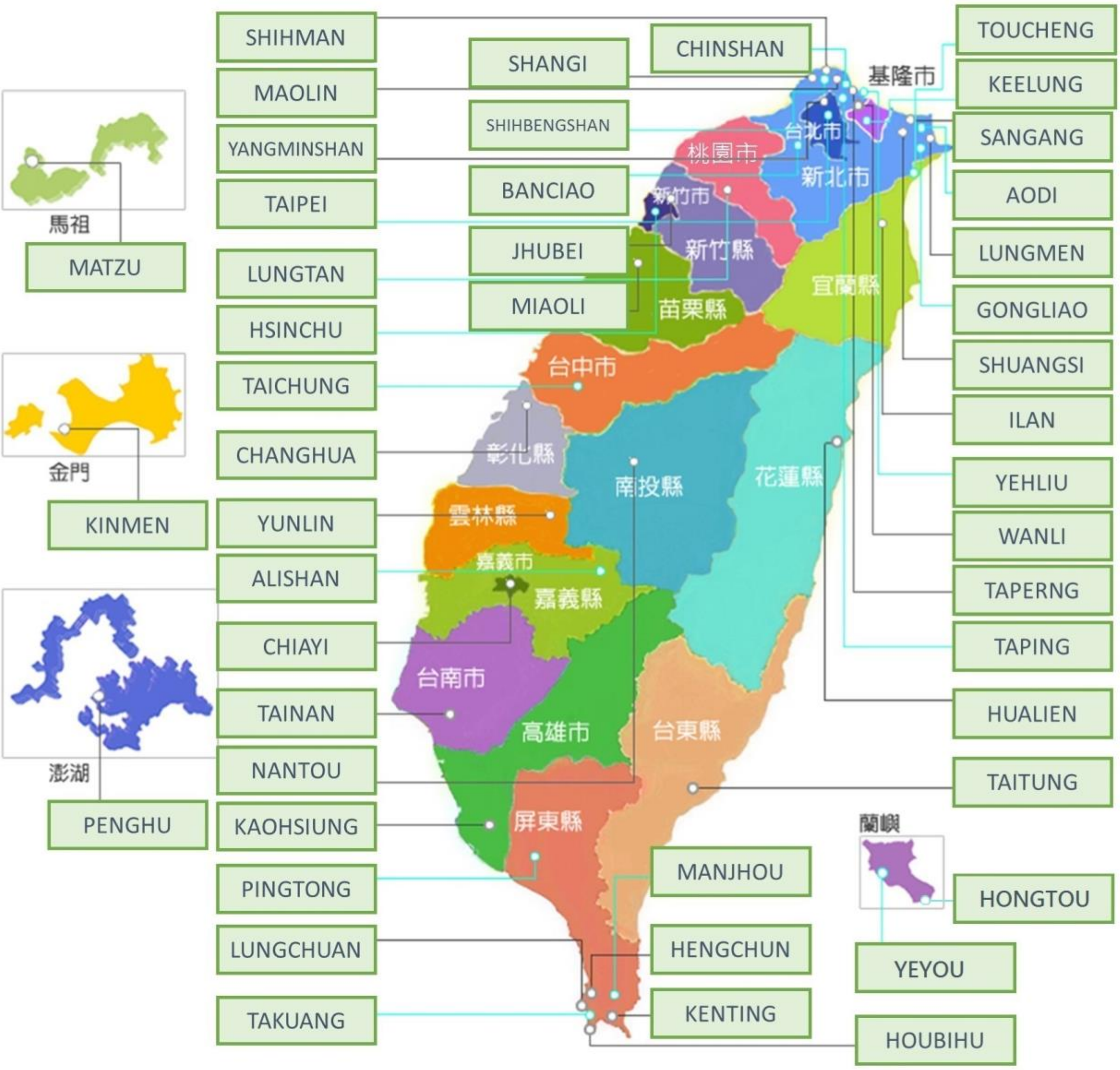
After the Fukushima accident, the Ministry of Health and Welfare (MOHW) has amended the Guidelines of permitted radioactive contamination in food on January 18, 2016 (website: http://goo.gl/o75U0n). In addition, the AEC has already promulgated an interim standard on the radioactivity control of imported consumer products and inbound passengers after Fukushima nuclear accident (0.2μSv/h).

15.3.2 Review the Rescue and Support Capability

For radiation monitoring and protection purpose, the manpower and equipment of relevant organizations are required to be mobilized and coordinated. Local government agencies should either establish their own laboratories or cooperate with universities or research institutes to enhance their radiation monitoring capability.

15.3.3 Establishment of Dose Reconstruction Capabilities for the Public and Rescuers

In order to provide the full-function bio-dosimeter services after an accidental exposure, the Institute of Nuclear Energy Research is required to establish the bio-dosimeter laboratory, purchase the automatic microscope, establish the in vitro dose calibration curve and participate in the bio-dose network for inter-comparison.



Note:
(1) There are 46 radiation monitoring stations in total (up to September 2016).
(2) Dose rate (in μSv/h) is updated every 5 minutes.

Figure 15.1 Environmental Radiation Monitoring Network in the ROC

15.3.4 Enhancing the Analysis Capability of Environmental Radiation Detection Laboratories

For the purpose of enhancing the analysis capability of the environmental radiation detection laboratories in the cities or counties around NPPs, it is necessary to setup the high efficiency pure germanium multi-channel analyzer system and the wide range pure germanium energy spectrum analysis system, establish the prompt accidental detection and analysis capability, upgrade the environmental detection technique and quality, and purchase the induction coupled plasma mass spectrum (ICP-MS) equipment. It is also necessary to establish a 3G wireless-transmission for the emergency response environmental radiation monitoring and display network, which can be installed at any chosen locations and is able to transmit the real-time radiation level and meteorological information.

15.3.5 Radiation Safety Measures Taken by AEC to Reduce the Impact of Fukushima Accident to the People in Taiwan

During the Fukushima Accident Period:

(1) Install the Door-type Radiation Detectors at Airports and Ports for Rapid Detection of Incoming Passengers

AEC set up the door-type radiation detectors in 4 airports and 2 ports, leading to a rapid screening to detect whether incoming passengers from Japan are contaminated or not. From March 15, 2011 to April 30 of the same year, a total of 200 thousand passengers had been screened and only 45 persons were detected to have a very slight contamination. They were allowed to leave the scene without any radiation safety concern.

(2) Plan the Inter-Ministerial Cooperation Mechanisms to Protect the Radiation Safety of Food, Water and Commodity for the People

In order to ensure the public health, the AEC invited relevant ministries including the Ministry of Health and Welfare (MOHW), the Council of Agriculture (COA), the Ministry of Finance (MOF), the Ministry of Economic Affairs (MOEA), etc. in charge of food, agriculture, fishery, animal husbandry, tobacco, alcohol and drinking water to hold several ad hoc meetings. The AEC also coordinated the environmental radiation laboratory of the country with analytical capability to do the effective detection and analysis services. For consumer products imported from Japan, the AEC promulgated an “Interim Standard on the Radioactivity Control of Imported Consumer Products and Inbound Passengers after Fukushima Nuclear Accident” as a guide for the Bureau of Standards, Metrology and Inspection (BSMI) of MOEA, the Customs Administration of MOF, and the Chunghwa Post Co. to detect imported products or mails from Japan by using the radiation detector.

(3) Integrate the National Radiation Monitoring Systems to Assess the Impact of Radioactive Fallout

To assess the degree of hazard increase caused by the fallout approaching Taiwan, the AEC integrated the monitoring systems of 6 accredited national environmental radiation laboratories and the wind forecast by the Central Weather Bureau of Ministry of Transportation and Communication (MOTC) to conduct the nationwide environmental radiation monitoring. The monitoring items included air, soil, plants, dust, sea water, etc. Results of these monitoring were published on the AEC’s website immediately in order to make people feel safe.

After the Fukushima Accident:

(1) Continue the Inter-Ministerial Cooperation in the Protection of the Public on Radiation Safety of Food

After the occurrence of the Fukushima accident, the AEC received commissions continually from the MOHW, COA, MOF and other government units, asking to conduct radiation detection and analysis on food, alcoholic drinks, tobacco, and fishery products imported from Japan. Until July 2015, a total of over 78,000 items had been examined and the relevant job is still ongoing. Besides, for control of the Japanese food imported to Taiwan, the MOHW adopted the same standard (cesium: 100 Bq/kg) as Japan and Korea, which is more stringent than the international standards of the Codex Alimentarius Commission (Codex) (cesium: 1000 Bq/kg). With the above control measures, the food radiation safety of the population can be assured.

(2) Strengthen the Domestic Radiation Warning Systems by Introducing the Aerial Measuring System

In view of the Japan's accident experience, the land, sea and airspace full range radiation detection information must be understood exactly. The AEC has completed operations of the sophisticated early warning systems and implemented the radiation monitoring project for the diffusion of fallout mutually with foreign agencies. This included the establishment of utilization of the Coast Guard ships and military helicopters to support the radiation monitoring in order to strengthen the monitoring capability in case of a nuclear accident. Furthermore, some US air and ground radiation detection equipment has been introduced to upgrade the domestic real-time detection capability during emergency response.

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 “National Nuclear Accident Emergency Response Plan (NNAERP)” 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 the following section. The nuclear reactor facility licensee shall set up a dedicated Nuclear Emergency Response Organization (NERO) in Taipower headquarters and the onsite 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 Nuclear Accident Emergency Response Basic Plan (NAERBP) (which was enacted in July 2005 and later amended in 2009 and 2014 by AEC to replace the previous NNAERP) 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

According to the Emergency Response Basic Plan (ERBP) as amended on September 24, 2014, the TPC is responsible for all the emergency response activities inside the plant in case of a nuclear accident, while the National Nuclear Emergency Response Center (NNERC), which is under the Central Disaster Response Center (CDRC) and activated by the AEC, assumes those activities outside the plant. The commander of CDRC is designated by the Premier of Executive Yuan, and the AEC Chairman will be the co-commander. The NNERC 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, Ministry of Health and Welfare, National Communications Commission, Environmental Protection Administration, Coast Guard Administration, Office of Disaster Management, Department of Information Service, Ministry of Foreign Affairs, Ministry of Finance, Ministry of Education, and Ministry of Science and Technology. Under the command of the NNERC, there are three temporary emergency centers, including the Nuclear Emergency Support Center (NESC), the Regional Nuclear Emergency Response Center (RNERC) and the Radiation Monitoring and Dose Assessment Center (RMDAC). Figure 16.1 shows the response to a complex nuclear disaster. For more information, please see Subsection 16.2.1(1).

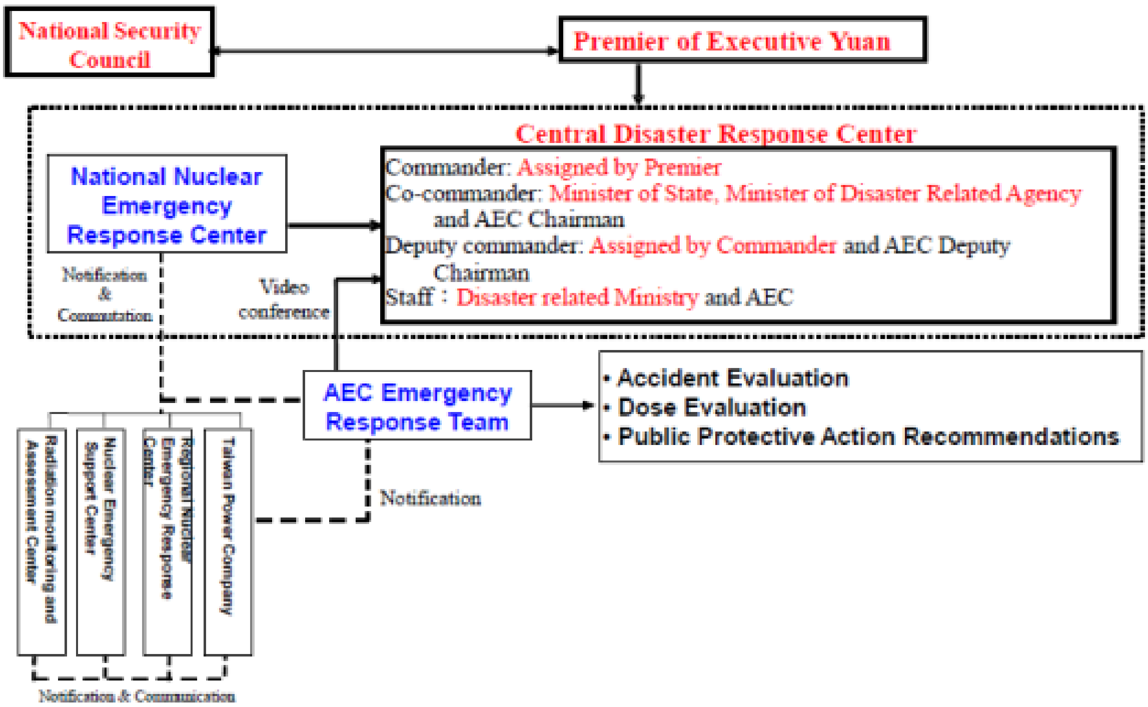


Figure 16.1 Response to a Complex Nuclear Disaster

Missions of the above-mentioned emergency response centers 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 the Regional Nuclear Emergency Response Center (RNERC) and the 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, the TPC, the Army Command Headquarters, and the Maritime Patrol Directorate General. Its major missions are information collection, siren broadcasting, accident consequence prediction, radiation detection and monitoring, public dose assessment, protection actions 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 notification of in-house sheltering and evacuation, distribution of iodine tablets, reception and accommodation of the public, temporary migration of the affected inhabitants, medical cares for the injured people, traffic control, alert and order maintenance, and assistance for announcing alarm and publishing news.

(4) Nuclear Emergency Support Center

The Nuclear Emergency Support Center (NESC) is composed of experts from the military corps. 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 Organization in TPC headquarters, and activate the Nuclear Emergency Response Organization within the facility consisting of Technical Support Center (TSC), Operation Support Center (OSC), Health Physics Center (HPC) and Emergency Public Information Center (EPIC), upon the occurrence or possible occurrence of a nuclear accident. The Nuclear Emergency Response Organization is responsible for support and coordination of response activities inside the facility. Also this Organization has to evaluate the accident situations and possible radiation dose influence, and cooperate with NNERC, RMDAC, NESC and RNERC for proceeding of the related emergency response measures. This Organization 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 onsite Nuclear Emergency Response Organization 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 offsite 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 plant 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 Organization of TPC headquarters will evaluate the possible impact of the accident and prepare for necessary response activities. AEC will establish an emergency team, and notify RMDAC, RNERC and NESC to standby, based on the accident situation and its possible impact. AEC may notify 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 the 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 of AEC.
- 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 of AEC 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 re-analysis of the accident dose and risk distribution for the current EPZ after the Fukushima accident in 2011 was made by applying the computer code MACCS2 with the following input information:

- Assumption of source terms resulted from simultaneous failure of multiple units induced by composite natural disasters,
- Updated meteorological data,
- Possibility of radiation release,
- Weather conditions,
- 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 after the Fukushima accident in 2011, the EPZs for the three operating NPPs were all set as a circle with the radius being extended from 5 to 8 kilo-meters from the center of the nuclear power station. The boundary of EPZ of each NPP 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 emergency plan implementing 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 and Training

16.1.2.1 Nuclear Accident Emergency Response Basic Plan and Nuclear Emergency

Public Protection Plan

Based on the Nuclear Emergency Response Act promulgated on December 24, 2003, the AEC shall consult all designated agencies to lay down the Nuclear Accident Emergency Response Basic Plan (NAERBP) and the Nuclear Emergency Public Protection Action Guides. The contents of the Nuclear Accident 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, the NNERC, the RMDAC, the NESC and the 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 Nuclear Accident Emergency Response Basic Plan (NAERBP) and the Nuclear Emergency Public Protection Action Guides (NEPPAG). 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 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 siren 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 of the public protective measures within the EPZ. 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, periodic training courses together with the equipment test and maintenance are held in each nuclear power station and the designated agencies. The scope of training includes emergency response implementing procedures, rescue of injured persons, and emergency repair of damaged equipment. Inspectors from 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 NNERC. Special trainings on 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 within the EPZ every time before an off-site drill is 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 (or USBs) about nuclear emergency response are also distributed in the EPZ area associated with each nuclear power station every year.

In order to ensure 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

A 24/7 (24 hours a day, 7 days a week) operation center, the Nuclear Safety Duty Center (NSDC) established in the AEC, provides the single point-of-contact for collecting, processing and disseminating emergency notifications whenever needed. It performs initial notifications, and coordinates management, logistics and mobilization actions during periods of national emergencies, natural and man-made disasters, or other extraordinary situations requiring centralized management notification and response. The main functions and capability of this NSDC center are illustrated as follows:

(1) 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.

(2) Environmental Radiation Monitoring System (ERMS): Before 2011, there were a total of 30 radiation monitoring stations around this country, providing the real time environmental radiation information nationwide. After the Fukushima accident in 2011, additional 15 stations were installed to broaden the monitored region.

(3) 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 AEC, TPC Headquarters and the operating NPPs will be performed.

(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) 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.

(6) 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.

(7) 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

In case an “Alert” event occurs, the licensee must mobilize all personnel of the Nuclear Emergency Response Organization and the Nuclear Emergency Response Unit of the nuclear facility to respond to the event and immediately notify the AEC and the local government. The AEC will then mobilize the Emergency Response Team which consists of senior directors and heads of the AEC as well as technical groups as the sub-teams, and accomplish a second level activation of the National Nuclear Emergency Response Center (NNERC). In the meantime, the AEC will notify related organizations to set up the Radiation Monitoring and Dose Assessment Center (RMDAC) as well as to accomplish a second level activation of the Regional Nuclear Emergency Response Center (RNERC) and the Nuclear Emergency Support Center (NESC). If the situation worsens to become a “Site Area Emergency” accident or even a “General Emergency” accident, besides conducting his emergency response actions continuously, the licensee must regularly report to the NNERC and the RNERC. After receiving the notification of a worsened accident, the NNERC must accomplish its first level activation and notify the local government and the Ministry of National Defense (the Army) to accomplish a first level activation of the RNERC and the NESC, respectively.

(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 nearby the NPP include radio, TV, broadcasting vehicles and emergency siren systems set up at police stations in the EPZ, etc.

(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 16.1. Medicines for radiation dose reduction (e.g. Iodine Tablet) are prepared for all the evacuees. Accommodation centers 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 centers. De-contamination processes will be executed wherever necessary. RNERC is responsible for providing the evacuees water, food, medicines and other necessary assistance and 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 plant, the decision on whether the offsite residents need to take shelter or to evacuate or other protective actions is based on the predicted radiation exposure as listed in Table 16.1, which forms the Protective Action Guides (PAGs) used by 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 16.2.

Table 16.1 Intervention Levels for Protective Actions

Protective Action	Intervention Level
All residents need to take sheltering inside the house	Avertable Dose of 10 mSv in 2 days
Residents to be evacuated from EPZ	Avertable Dose of 50 to 100 mSv in 7 days
Take Iodine Tablet	Avertable Thyroid Equivalent Dose of 100 mSv
Temporary relocation (To be terminated when Projected Dose below 10 mSv in 30 days)	Projected Dose of 30 mSv in 30 days
Permanent Relocation	Expected Lifetime Dose Greater than 1 Sv, or Temporary Relocation over 1 Year

Table 16.2 Food and Drinking Water Control Standards

Radionuclide	Action Level (kBq/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

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. Critics and recommendations from this group are documented for TPC's improvement.

For the off-site emergency response, a full scale exercise was held in this country every two years before 2001. However, the frequency has been changed to once a year since 2002 as required by the government. Currently the southern nuclear facility (i.e. Maanshan) and one of the northern nuclear facilities (i.e. Chinshan or Kuosheng) held the exercise by turns. The items of the off-site exercise in recent years 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, related recovery measures, preventive evacuation of students and vulnerable groups such as those in hospitals and elderly care centers, in-house sheltering and evacuation of populations in EPZ, setting up the aerial measuring system, terrestrial and marine radiation monitoring, setting up the accommodation sheltrs, medical care of contaminated persons, etc. The participating organizations include all Ministries involved in the NNERC, the RNERC, the RMDAC, the NESC, and the 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 evaluated 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 measures. The missions of the relevant organizations are as follows:

(1) Ministry of Interior (MOI)

The Ministry of Interior 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 (MOND)

The Ministry of National Defense 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 (MOF)

The Ministry of Finance 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 (MOEA)

The Ministry of Economic Affairs 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 (MOTC)

The Ministry of Transportation and Communication 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 (DGBAS)

The Directorate-General of Budget Accounting and Statistics provides the local government of the affected regions the financial support required to perform the recovery measures.

(7) Ministry of Health and Welfare (MOHW)

The Ministry of Health and Welfare 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.

(8) Environmental Protection Administration (EPA)

The Environmental Protection Administration 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.

(9) Financial Supervisory Commission (FSC)

The Financial Supervisory Commission 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.

(10) Atomic Energy Council (AEC)

The Atomic Energy Council 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.

(11) Council of Agriculture (COA)

The Council of Agriculture 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.

(12) National Communications Commission (NCC)

The National Communications Commission coordinates the communication organizations for the normal communication in the affected regions, and provides the emergency communication measures as needed.

(13) 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.

(14) Nuclear Reactor Licensee

The nuclear reactor licensee shall recover the damaged nuclear facility, perform radiation monitoring, dose assessment, 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 Current Nuclear Damage Compensation Law with the latest version promulgated on May 14, 1997 was 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.

After the Fukushima accident, the Nuclear Damage Compensation Law was reviewed and it resulted in some amendments proposed. It was proposed to abandon the immunity of the licensee’s liability on nuclear accident caused by severe natural disasters, to raise the limit of the total liability claims for one nuclear accident from current 4.2 billion NT$ to 15 billion NT$ which is equivalent to about 0.3 billion Special Drawing Rights (SDR), and to extend the expiration limit required to submit a compensation claim from 10 years to 30 years since the date when the nuclear accident occurred. However, these proposed amendments, which had been submitted to the Legislative Yuan (Congress) for review, were withdrawn from the Congress on July 27, 2016 because of the new government policy of nuclear power phase-out by 2025.

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) Regular communication tests between the AEC and the headquarters of the IAEA, the USNRC, and the USDOE/NNSA (National Nuclear Security Administration) have been performed for several years. And participation of the emergency drills held by the IAEA, and INEX (International Nuclear Emergency Exercises) International Exercises as well 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.

16.4 Fukushima Lessons Learned

16.4.1 Emergency Response Mechanism of Complex Disaster

After the Fukushima accident AEC had revised ERBP plan to cope with complex disaster which meant natural disaster induced nuclear accident. Once a natural disaster, such as earthquake, tsunami, extreme rainfall and mudslide, combined with a nuclear accident were happening, the national emergency response team has more agencies get involved and the operation has been enhanced through multi central agencies coordination. And the timing to activate the nuclear emergency response mechanism has been elevated as one phase earlier, it means the response mechanism will be activated at level of Alert instead of Site Emergency before Fukushima Accident. If an Alert was notified to AEC from NPP, AEC emergency response team will be activated immediately, and NNERC will be formed soon after approved by AEC Chairmen, then the following response steps remain the same as previously described, and all staff of National Centre will move and merge into “Central Disaster Response Centre” to deal with complex disaster. The commander of Central Disaster Response Centre will be designated by Premier of Executive Yuan, and AEC chairman will be the co-commander.

16.4.2 Emergency Response and Preparedness

Based on the revelations of Fukushima nuclear disaster, the overall nuclear safety re-assessment in Taiwan was completed, and a number of related enhancement measures, including the expansion of the EPZ radius from 5 km to 8 km, were carried out.

Application of Atmospheric Dispersion Model :

In the past, the research had been focused on the atmospheric dispersion assessment near the NPP area. The so-called two or three dimensional atmospheric dispersion model had been established. After the Japan's Fukushima accident, the focus of research shifted to the long-range atmospheric dispersion. It is important to realize how to integrate those different models or to harness a highly recognized model in the international community. The cooperation with the US National Nuclear Security Administration of the Department of Energy (USDOE/NNSA) could benefit the ongoing model (XOQDOQ) development in Taiwan.

The Cooperation Cases of the Atomic Energy Council (AEC) with the US National Nuclear Security Administration of the Department of Energy (DOE/NNSA) ─ Case 1:

To Construct the Taiwan Radiation Assessment Network and to Get Rid of the Fukushima Shadow (by INER):

A severe accident caused by an extreme huge tsunami occurred in Japan's Fukushima Daiichi Nuclear Power Plant on March 11, 2011. Large amount of radioactive material was released to the atmosphere, and transported to the atmosphere with dispersion. The government, mass media, and the public in Taiwan paid much attention on the radiation effect from the Fukushima event. The Atomic Energy Council (AEC) re-evaluated and strengthened the capabilities and requirements for emergency response of nuclear accident in domestic nuclear power plants.

The first priority is to combine the dedicated technologies of Institute of Nuclear Energy Research (INER) and Central Weather Bureau (CWB) to develop a nation-level system for radiation evaluation and forecasting in response to nuclear accidents occurring abroad. An evaluated forecast of the radiation fallout impact can provide the government with a powerful information support for responding to the occurrence of foreign nuclear accidents. With this Radiation Assessment Network system, it will be helpful for strengthening the basis of decision making and for the authorities to take proper action guidelines during emergency response periods.

The dose evaluation system for domestic NPP is based on the extension of emergency response region. Developments of the assessing ability for the entire Taiwan area with a 2.5 km spatial resolution, simulation for multiple release points, and the reverse estimate of release source terms were included in the dose evaluation system.

16.4.3 Marine and Airborne Radiation Monitoring

(1) Marine monitoring: Coordination with Coast Guard and Ministry of Defense has been set up to establish the supportive procedure to monitor marine radiation level.

(2) Airborne monitoring technique: Airborne monitoring is an effective way to monitor the radiation level in the air in an effort to judge its possible radiological impact to the environment and the public. Meanwhile, the government can use this information to guide the public to take early evacuation or sheltering measures. In Taiwan, small unmanned aircrafts were developed mainly for meteorological studies. The hardware and software for the airborne radiation monitoring technique will be developed with the assistance of the USDOE/NNSA.

The Cooperation Case of the Atomic Energy Council (AEC) with the US National Nuclear Security Administration of the Department of Energy (DOE/NNSA) ─ Case 2:

Aerial Measurement Technology Development (by INER):

INER established the aerial measurement technology with support from US DOE/NNSA. Collaborating with the military CBRN (chemical, biological, radiological and nuclear) and helicopter groups, the aerial measurement was a major item in the annual NPP drill. Considering that access to a high dose rate area is difficult for a man-operated helicopter after nuclear accident, the lightweight portable radiation detector carried by unmanned aerial vehicle (UAV) was developed and tested to reduce the exposure of the responders.

16.4.4 Review of Radiation Detection Plan

Additional establishment of real-time environmental radiation monitoring stations and monitoring routes was planned in the radiation detection plan. Each NPP is required to be equipped with sufficient amount of radiation detection vehicles for necessary emergency detection.

16.4.5 Precautionary Evacuation and Nuclear Disaster Response Measures

After the Fukushima nuclear accident, AEC conducted analysis and study for the evacuation operation, which resulted in the following recommendations:

(1) Preventive Evacuation

(a) Evacuation of the General Public

Radioactive materials might be released at the beginning of a nuclear accident. Thus, a first priority is to perform the preventive evacuation measures for people living close to the NPP within the 3 km range as well as for the vulnerable populations and particular groups (e.g., those in hospitals, schools, nursing and elderly care centers, etc.) in EPZ.

As for the tourists and people participating in large-scale activities, they will be notified to leave in advance at the initial stage of the accident. Also, the entering of vehicles into the EPZ will be under control.

(b) Evacuation of the Primary and Junior High School Students

Every primary or junior high school inside the EPZ has already pre-planned its evacuation plan.

If a nuclear accident happened in normal school hours and, by analysis, it may deteriorate to result in radiation release outward, local governments will directly notify the affected schools and start sending the preparatory vehicles. Students will then be sent to the planned "host school" which is located 16 kilometers away from the NPP. By doing this, it may avoid the coming reversely to school of the students' parents to take their children and help reducing the level of traffic confusion and the risk of radiation exposure to the parents and students.

If the students' parents work near the school or their homes are near the school, the parents can conveniently take their children home, but the school must make the relevant registration operations. In the meantime, the department of education of the local government and the school are usually required to survey in advance the number of special vehicles for evacuating teachers and students and plan the number of vehicles needed in case of emergency as well as to make ready the relevant preparations.

(2) Downwind First for the Evacuation

A nuclear accident usually will progress gradually and the risk is increased gradually too. The basic principle of nuclear disaster hedge in implementing evacuations is "from the inside out" and "downwind first".

(3) Indoor Sheltering

For local residents' emergency response to a nuclear accident, taking indoor sheltering is another effective radiation protection measure besides the evacuation. The reinforced concrete is an effective shielding material. The radiation protection effect of taking sheltering inside reinforced concrete houses can be high. Thus, in regions where the government has not given instructions to evacuate, the major emergency response is taking indoor sheltering which is to avoid the unnecessary loss of life and property due to the hurried and confused evacuation. When the government gives the instruction asking local residents to take indoor sheltering, it will mean that at this time the radiation protection effect of taking indoor sheltering is better than that of evacuation.

(4) Update of the TPC's Public Protection Measures

As mentioned in Subsection 16.1.1.4, after the 311 nuclear accident in Japan, the emergency planning zones (EPZ) of the 3 operating NPPs in Taiwan were expanded from 5 km to 8 km radius. TPC completed a review report on "Analysis and Planning of the Protective Measures of the Public Living within the EPZ" and submitted to AEC and the local governments for review. The contents of this report consist of the population distribution, radiation detection program, public warning system, public assembling, evacuation and relocation, etc.

Based on this report, the local governments revised their response plans for the protection of the public within the EPZ. The contents of these plans consist of emergency response organization, sites, equipment, and the usual preparedness and response measures to assure the safety of the people, such as various ways warning alert notification, implementation of sheltering, notification of distribution and taking of the iodine tablets, implementation of evacuation, etc.

AEC also planned the response mechanism against the composite disaster resulting from a nuclear accident concurred with the occurrence of major natural disasters, which had been incorporated into the regulation "Directions on the Operations of National Nuclear Emergency Response Center" in February 2012. In addition, in the amendment of "National Disaster Prevention and Relief Act" promulgated by the President on April 13, 2016, the radiological disaster was formally included as one of the official disasters in this Act. It follows that, in the future, the prevention, preparedness, response and recovery operations for a domestic nuclear disaster by the central and local governments will be integrated to effectively utilize the national resources for rescue.

(5) Nuclear Emergency Exercises

> Each nuclear power plant (NPP) is required by the AEC to conduct an onsite emergency exercise at least once a year. Together with various ministries and local governments, the AEC also conducts a national nuclear emergency exercise (NEE) every year in one selected NPP.

The latest NEE with reference to the Fukushima experience was conducted in September 2016 at Maanshan NPP. In this NEE, the Pingtung County Government in conjunction with the relevant agencies (institutions) and civil volunteer groups practiced the early warning notifications, evacuation, and shelter operation. Emphasis was put on preventive evacuation drills and the establishment of mechanisms and practices of high quality shelter operation.

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 these 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.

Table 17.1 Comparison of the Siting Factors for the Yenliao, Laomei, Kuanyin and Tawu Sites

Site / Factor	Yenliao	Laomei	Kuanyin	Tawu
Seismology	•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.			•Tawu site is very marginal and cannot meet USNRC Siting Criteria. •SSE would be higher than 0.55g.
Geology	Geology of Yenliao, Laomei, and Kuanyin areas are favorable sites for nuclear power station.			Tawu site is less desirable because of its proximity to a major active plate boundary and possibly capable faults.
Foundation Conditions	•Underlain by competent rock close to the ground surface and has the best foundation conditions. •Covered by 0 to 10 meters of alluvium.	•Laomei is also underlain by sound rock, but covered by up to 30 meters of alluvium. •Has slope stability problem.	•Has a marginal foundation for a nuclear power station. •11 to 15 meters of overburden overlies a soft, weak rock formation. •Has liquefaction potential.	•Tawu is the least desirable site with up to 53 meters of overburden on top of rock. •Serious slope stability problems.
Other	•Located on the northeastern coast of the island. •Close to the Fulong Beach.	•Extremely difficult to construct 345 kV transmission lines. •Close to Baisa Beach	•Close to Kuanyin Beach. •Difficulty of fresh water supply.	The population and population growth rate are very low around the site.

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, and Poor = 1.

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 onclusions 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.

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.

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.

17.3 Reevaluation of Site-Related Factors after Fukushima Daiichi Accident

After the Fukushima Daiichi Accident, AEC required TPC to reevaluate each site's capability to cope with extreme natural disasters (including earthquake, tsunami, and flooding) and to adopt the conclusions of USNRC NTTF Report Tier 1 Recommendations. AEC also required TPC to take countermeasures in accordance with the following regulatory orders to further enhance the capability to cope with extreme natural disasters:

- XX-JLD-10101 Reevaluations of Seismic Hazards
- XX-JLD-10102 Reevaluations of Flooding (including Tsunami) Hazards
- XX-JLD-10105 Facility Walkdowns Related to Seismic and Flooding Hazards

The current implementation status of these reevaluations and related safety enhancements are described as follows:

(1) Reevaluations of Seismic Hazards

- Geological Survey (in progress)

  According to the newly identified near-site active faults, an extended geological survey in marine region has been completed in October 2014 and submitted to AEC in January 2015. The review was completed by AEC in June 2016. Further investigation on fault parameters and seismic activity was implemented in July 2015 and is still in progress.

- Senior seismic hazard anaylsis committee(SSHAC) level 3 process (in progress)
- Seismic probability risk assessment (SPRA). (in progress)

(2) Reevaluations of External Flooding Hazards

- Assessment of the influence of tsunami induced by the 22 potential earthquakes of massive scale.(completed)
- Investigation and assessment of the volcano landslide on seabed and the ancient tsunami.(in progress)

(3) Safety Enhancements against Seismic/Tsunami Hazards

- Enhancement of the earthquake-resistant capabilities of two success paths to shutdown the reactor and maintain it in a safe shutdown condition based on results of seismic margin assessment (SMA) (completed)
- Setting up the Central Weather Bureau (CWB) earthquake and tsunami alert system. (completed)
- Enhancement of the earthquake-resistant capabilities of raw water reservoirs, raw water piping and added flexible expansion joints. (to be alternatively and properly completed)
- Conducting an enhancement evaluation of safety-related SSCs for Chinshan NPP, followed by the SSE upgrade from 0.3g to 0.4g. (in progress, but TPC might request to rescind the order.)
- Building tsunami-protective walls for all plants with a margin of 6 meters above the current licensing basis (CLB). (in progress)
- Enhancing the water-tightness of the fire doors and penetrations of buildings containing important safety related equipment. (in progress)
- Inspection of all tsunami/flooding-protection devices and seal functions. (completed)
- Simulation on the mechanism of seismic and tsunami hazards. (completed)
- Addition of water-tight barrier on emergency sea water system. (completed)
- Enhancement of the motor-operated tsunami protective gates in CS. (completed)
- Procurement of 40 sets of diesel driven drain pumps to strengthen portable drain capabilities. (completed)

(4) Facility Walkdowns Related to Seismic and Flooding Hazards

- Seismic walkdowns for the 3 operating NPPs in March 2013. (completed)

- Flooding hazards walkdowns for the 3 operating NPPs in June 2013. (completed)
- Walkdowns of onsite facilities for the Lungmen NPP in June 2014. (completed).

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 regulatory requirements related to both the Nuclear Reactor Facilities Regulation Act and the Nuclear Emergency Response Act, the important processes to construct a nuclear power plant include:

(1) To define an exclusion area (EA) and a low population zone (LPZ),

(2) To submit the PSAR to get a Construction License,

(3) To submit the FSAR to get an initial fuel loading permit and then Operating License, and

(4) To define an Emergency Planning Zone (EPZ) 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 EA within 2 hours in a postulated nuclear accident occurred in the plant. The owner of the plant has to purchase all land inside the EA.

(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 30 days 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 Assessment (PRA), 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 to ensure nuclear power generation safety. The logic is as follows:

  Should the first layer of defense fail for any reason, the next layer will still offer adequate safety and protective functions. In case the second layer of defense also fails, there will still be a third or even a fourth layer of defense that will continue to ensure safety and protection.

  The goal of setting up multiple layers of defense is to reduce the possibility of a nuclear power accident and minimize its impact. The layers of defense-in-depth to prevent the release of nuclear fission products include: fuel pellet, fuel rod, connection between the reactor pressure vessel and the closed coolant system, and containment building. The containment building ensures that all radioactive substances from the reactor or cooling system are isolated from the outside 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 (RCS) 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 (RHR) system is designed to remove the core decay heat.

The reactor protection system (RPS) 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 RPS 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 NPPs 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 MCR is designed so that even if a serious accident occurs, the operators still 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 NPP in order to get control of the accident conditions and to take appropriate actions.

For example, in the Lungmen Project, the Lungmen NPP 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.

Jiang Yihua, the former Premier of the Executive Yuan, announced in an international press conference on 28 April 2014 that unit 1 of the Lungmen NPP will be mothballed once pre-operational safety checks are completed. Construction of unit 2, meanwhile, will be suspended immediately.

In keeping with the government’ s instructions to mothball the Lungmen NPP, TPC mapped out three different strategies: keeping the Lungmen NPP available for future commission and minimize costs; strengthening nuclear communication during this period; and making preparation for commercial operation. Based on the Principles for mothballing Nuclear Power Plants from the AEC, TPC submitted the Lungmen NPP Mothballing Plan to the Executive Yuan on 29 August 2014.

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 NPPs.

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 in association with its Enforcement Rules and Regulations stipulates that the MCR, SPDS, 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 MCR, human factors are considered so that the man-machine interface (MMI) 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 SPDS, 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 MMI 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 (HSI) of the MCR for the NPP 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 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 MCR 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 Subsection 12.3(5) of this report.

18.4 Fukushima Lessons Learned

The direct cause of the nuclear accident at Fukushima, an earthquake with magnitude 9.0 resulting in an over 14 meters high tsunami, is far beyond the design basis tsunami analyzed by the utility and approved by the regulator. Although there have been huge tsunamis attacking the east coast of northern area in the main island of Japan, the design basis tsunami at the Fukushima Daiichi site appears to only have been made to protect against a 5.7 meters high surge above sea level based on numerical simulation only. The NPPs of the TPC, both operating and being under construction, thus followed the lessons learned from the Fukushima Daiichi accident to re-visit the design basis and this resulted in the implementation of the following safety enhancement measures.

(A) 11 Technical Areas for Safety Reassessment of Taiwanese NPPs

By reference to measures recommended by various major nuclear authorities or international organizations, such as US Nuclear Regulatory Commission (USNRC), Nuclear Energy Institute (NEI), European Nuclear Safety Regulators Group (ENSREG), World Association of Nuclear Operators (WANO) and Japanese Nuclear and Industrial Safety Agency (NISA), in the Safety Reassessment Program conducted in Taiwan right after the Fukushima accident as described in Subsections 6.4.1 and 14.3.1, AEC required TPC to verify the capability of NPPs in response to both the DBA and beyond DBA. There are 11 technical areas for reassessment (by TPC) and review (by AEC) as follows:

(1) Capability of protection against loss of all AC power (SBO)

(2) Capability of protection against flooding and tsunami

(3) Integrity and cooling capability of spent fuel pool

(4) Capability of heat removal and ultimate heat sink (UHS)

(5) EOP and training

(6) Establishment of plant ultimate response guides (URG)

(7) Mutual support between units 1 and 2 of the same NPP

(8) Consideration of compound accidents

(9) Mitigation of the beyond DBA

(10) Preparedness of facilities, equipments and backup spare parts

(11) Manpower, organization, and safety culture.

(B) Additional Areas of Improvement Requirements

Although the results of this reassessment revealed neither immediate nuclear safety concern nor threat to the public health and safety, the AEC still requested that TPC focused on strengthening its reevaluation on design basis against earthquakes, tsunamis and heavy rainfalls, and enhancing its capability to mitigate a prolonged station blackout (SBO) for further improvement. Many areas of improvement requirements have been identified as follows:

(1) SBO

- Reanalysis for 24-hr SBO capability
- Enhancing the battery capacity in response to requirement to extend SBO coping time from 8 to 24 hours.
- Load distribution planning of air-cooled swing EDG for both units of the same NPP

(2) Flooding Protection

- Enhancement of flooding protection and water tight design for emergency service water room and other supporting facilities
- Additional spare parts for the emergency service water (ESW) and emergency core cooling system (ECCS) pumps, and storage of them at

higher ground

- Reevaluation of the design basis tsunami heights
- Physical separation of service water systems for different units

(3) Spent Fuel Pool

- Safety-related instrumentation for monitoring water level, temperature, etc. and the equipment setup for the 3 operating NPPs
- Safety-related power supply for SFP makeup
- On-site emergency power to pumps and instrumentation
- Seismic resistant spray to the spent fuel pool

(4) Vent

- Mobile air compressor to operate vent valves
- Re-examining the vent route for hydrogen
- Prevention of H_2 accumulation in the buildings

(5) Severe Accident Management

- Establishing the URG procedure to prevent an event from becoming an accident or severe accident or to sacrifice the possibility of future electricity re-generation from the affected NPP should it become necessary to inject sea water into the reactor in order to prevent the core from melting. (see also Subsection 6.4.3.5(2).)
- Procurement of more natural boric acid
- Building a seismic-resistant technical support center
- Simulation of the severe accident scenario of Fukushima-like case for NPPs

(6) Seismic Enhancement

- Strengthening the robustness of raw water reservoir and its piping
- Reevaluating the seismic hazard analysis

(7) Infrastructure

- Coordination with outside supports (military, fire department)
- Examination of internal capability and training in response to extended accident sequence

(8) Safety Culture

- IAEA safety principle (SF-1) implementation
- NRC safety culture statement and ROP inspection for cross-cutting issues

(C) Reinforcement for BDBA

The robustness of NPPs had been examined by the EU stress tests performed in Taiwan in case of beyond design basis events due to earthquakes, external flooding and extreme weather conditions, as well as loss of the power supply and loss of the heat sink. AEC requested TPC to implement more countermeasures to further enhance the capability to cope with extreme natural disasters. The reinforcement for BDBA includes:

(1) Safety Enhancement for Core Cooling (see Figure 18.1)

- Capacity of all water resources: on-site and off-site, and transfer and injection procedures developed.
- Fire engine resources: quantity, capacity, discharge pressure, and redundancy developed.
- Scheme for alternative reactor coolant injection (various paths) developed.
- Alternative UHS developed.
- Scheme for recovery of UHS developed.
- Procurement of portable air compressors and spare nitrogen bottles for SRVs and air-operated valves.

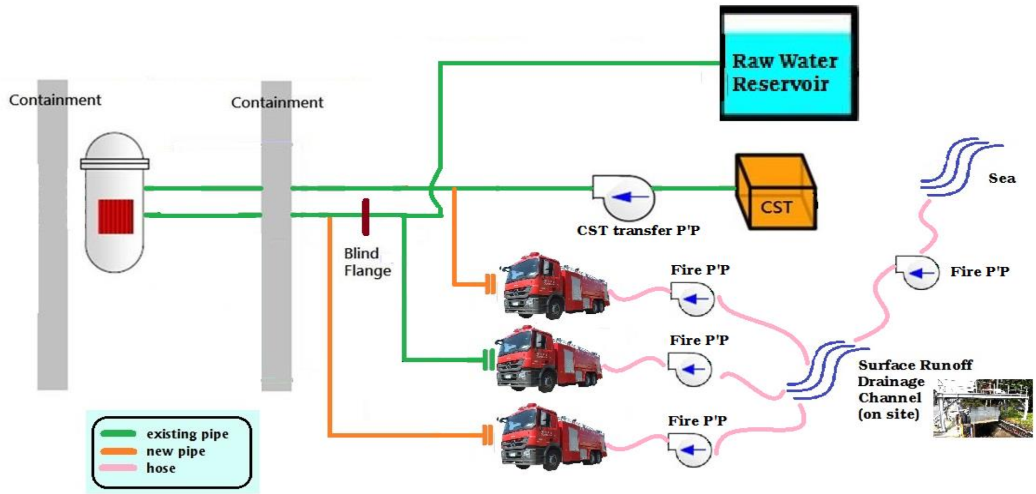


Figure 18.1 Safety Enhancement for Core Cooling

(2) Spent Fuel Pool Emergency Makeup (see Figure 18.2)

- Various SFP make up strategies developed.
- Extra makeup and spray flow paths installed according to NEI 06-12.
- SFP instrumentation enhanced, per NTTF 7.1, including upgrading the instruments for monitoring water level and temperature to safety grade equivalent.

(3) Containment Integrity and Hydrogen Control

- Addition of a robust and reliable containment filter venting system per EU's experience (in progress).
- Addition of passive autocatalytic recombiners (PARs) for MSNPP (PWR) per EU's experience (in progress).

- Development of containment early venting strategy
  - To reduce the temperature and pressure rises in the torus, and
  - To lengthen the injection time for RCIC and enhance the availability of RPV injection

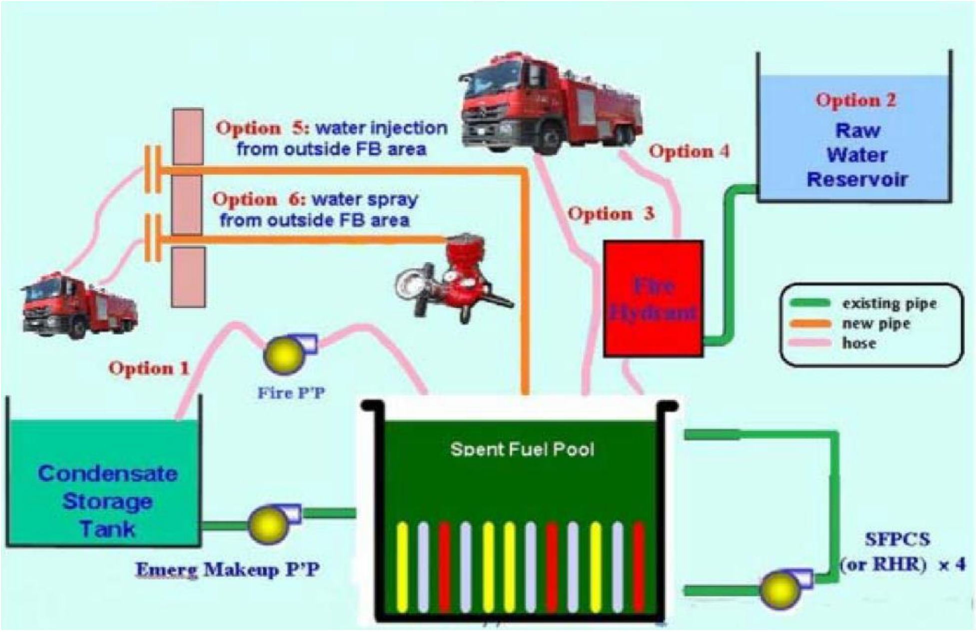


Figure 18.2 Emergency Makeup for Spent Fuel Pool

(4) Electrical Power Source (see Figure 18.3)

- The 5th EDG (swing EDG) can now supply emergency loads for both units simultaneously.
- The black-start D/G used to start G/T can now supply emergency loads for both units simultaneously.
- Bunker the air-cooled swing EDG for 3 operating NPPs and installation of air-cooled G/Ts for Lungmen NPP (inside a seismically isolated building) planned.
- 6 sets of 4.16 kV power vehicles and 26 sets of 480V portable D/Gs procured.
- Capacity of DC power extended in response to the requirement to extend SBO coping time from 8 to 24 hours.
- Portable generators and batteries for control power and supervisory instruments prepared.

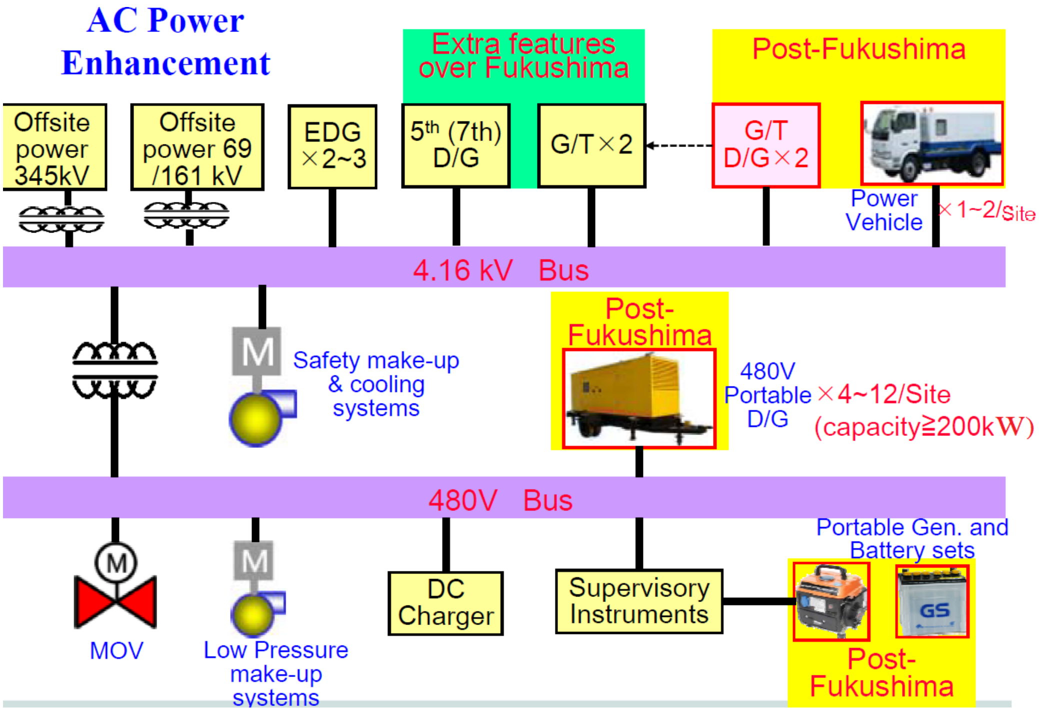


Figure 18.3 Safety Enhancement for AC Power Source

(5) Development of Ultimate Response Guidelines (URG)

The development background and main features of the URG are as follows:

- In light of lessons learned from the Fukushima accident, timely disposition in the MCR is the key of preventing an event from becoming an accident.
- The current EOPs may be to enhance for coping with complex external disasters.
- URG is specifically designed to cut off event evolution and make immediate actions to prevent core damage.
- URG will be integrated with EOPs, severe accident management guidelines (SAMG), and extensive damage mitigation guidelines (EDMG) (NTTF 8).
- The reactor core will be secured by emergency depressurization, containment venting and injection of any available water (even seawater) through any available injection paths whenever any of the following 3 conditions reached:
  - Plant suffered from larger than SSE earthquake and tsunami
  - SBO
  - Loss of UHS
- URG was named as DIVing plan, abbreviated from system depressurization, water injection and containment venting.

More information about URG can be found in Subsections 6.4.3.5, 10.2(11) and 19.4.2.

(D) Enhancement of the current licensing Basis

Figure 18.4 shows the enhanced capability including the enhancement of the design basis and addition of portable equipment to cope with BDBA.

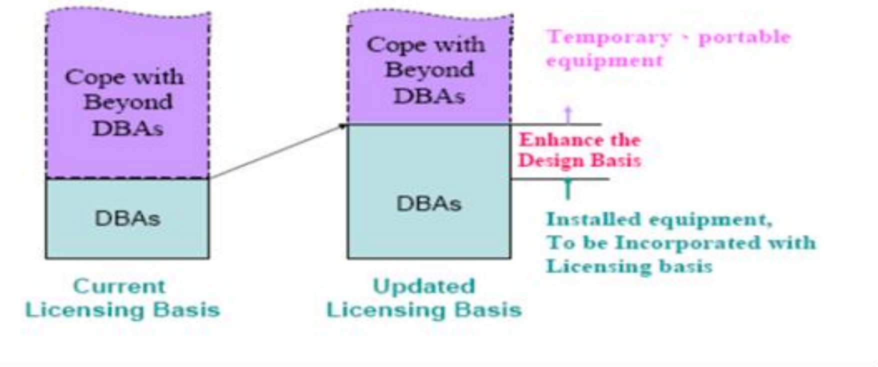


Figure 18.4 Capability to Cope with Beyond DBA

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 "Nuclear Reactor Facilities Regulation Act" of 2003 (Article 5), one must have a construction license (CL) in advance and satisfy the following conditions in order to construct a nuclear reactor installation:

- 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 as well as financial resources are adequate to operate the installation (one of the two pre-requirements for

qualifying to apply the CL of a NPP, as prescribed in Article 2 of "Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities" of 2004).

However, to obtain a CL of a nuclear reactor installation, an applicant must 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 according to Article 3 of "Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities" of 2004:

(1) Preliminary safety analysis report (PSAR),

(2) Environmental impact assessment (EIA),

(3) Applicant's financial capability, and

(4) Others as required and published by the regulatory body.

After receiving the above documents, the nuclear regulatory body (i.e. AEC) will issue its review conclusions in a safety evaluation report (SER) normally within one year.

After the construction work is completed, in order to get an approval for loading fuel into a newly constructed reactor for the first time, the holder of the CL must submit the following documents in 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) Application for initial fuel loading – 14 months before scheduled initial fuel loading,

(2) Final safety analysis report (FSAR) – 14 months before scheduled initial fuel loading,

(3) A plan, in accordance with the level of radiation dose received by the public after a nuclear accident, to the Regulatory Body for demarcating the exclusion area and the low population zone of the NPP in consultation with the Ministry of Interior, the government of municipality under the direct jurisdiction of the Executive Yuan, the county (city) government, and the relevant authorities – 14 months before scheduled initial fuel loading,

(4) Summary report on the corrective actions based on the inspection findings during the construction stage – 3 months before scheduled initial fuel loading,

(5) List of operating procedures – 2 months before scheduled initial fuel loading,

(6) Fuel loading plan – 2 months before scheduled initial fuel loading,

(7) Startup test plan – 2 months before scheduled initial fuel loading, and

(8) Reports on the systems' functional tests (or preoperational tests) – before scheduled initial fuel loading.

If an approval of initial fuel loading (IFL) is granted, then loading fuel into the newly constructed reactor can be performed for the first time. To comply with the regulatory requirements addressed in Articles 13 and 14 of "Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities" as amended in 2005, it is required that the application for an operating license (OL) has to be submitted within

18 months after the initial fuel loading was approved. In addition, the applicant needs to submit the approved EIA at least one year prior to the scheduled date of operation and an application form for OL as well as the following documents 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) Financial capability.

19.2 Operational Limits and Conditions

Article 6 of the Nuclear Reactor Facilities Regulation Act of 2003 and Article 2 of "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 CL 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), being part of the PSAR and FSAR as required by Article 4 of "Regulations on the Review and Approval of Applications for Construction License of Nuclear Reactor Facilities" and Articles 3 and 16 of "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 technical specifications 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 Article 13 of "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 2009), without the AEC's prior approval, neither a design amendment nor an equipment change shall be made, if it involves the revision of the TS.

The AEC encourages licensees to use the improved technical specifications (ITS) as the basis for the plant-specific TS. All three operating NPPs 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 Article 2 of "Regulations on the Review and Approval of Applications for Operating License of Nuclear Reactor Facilities" as amended in 2005, the licensee is required to submit a list of the plant operating 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 operating NPP are the plant operating 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 (SL) and reactor operators (including RO and SRO),
- 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: the control room operating procedures and 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 Operating Procedures
  - The emergency operating procedures (EOPs) 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 EOPs will be given in the following Subsection 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 NPP 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 operating procedures will be reviewed by the Station Operation Review Committee (SORC) of the NPP, and then approved by the Plant General Manager. In an operating NPP, the SORC is responsible for reviewing all safety-related affairs and making recommendations to the Plant General Manager. The major responsibilities of the SORC are described in more detail in Subsection 9.1.1.

As required by Article 9 of “Regulations on Quality Assurance Criteria for Nuclear Reactor Facilities” of 2003, any activities that may affect the quality of the plant must have appropriate procedures, instructions, or drawings. These activities, as specified in 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 surveillance requirements (SR) included in the TS 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 SR including the surveillance intervals requirements as specified in the technical specifications of the NPP.

The licensee of an operating NPP is required to provide the following reports to the regulatory body (AEC) in accordance with the associated time intervals as specified in Article 7 of “Enforcement Rules for the Implementation of the Nuclear Reactor Facilities Regulation Act” of 2003:

- Operation report ─ to submit a quarterly report within 30 days after the end of each quarter and an annual report within 60 days after the end of each year,
- Radiation safety and environmental radiation surveillance report ─ to submit a quarterly report within 60 days after the end of each quarter and an annual report within 90 days after the end of each year,
- Emergency event report ─ to report within 1 hour after identifying an event and submit the event report within 30 days, and
- Records on the radioactive waste production ─ to submit a monthly report within 30 days after the end of each month and an annual report within 3 months after the end of the year (as required in the administrative requirements of the TS of each NPP).

The Nuclear Safety Committee (NSC) in the headquarters of the licensee is the highest advisory organization in the TPC 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 NPPs 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. For reference, as described in Subsection 6.2.3.2, the regulatory inspections of the NPP include the resident inspections, regular inspections, expert team inspections, and special inspections as well as the unannounced inspections.

According to Article 9 of the “Nuclear Reactor Facilities Regulation Act” and 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 review (PSR) at least every 10 years as described in Subsection 6.2.3.4. In the PSR, 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 (AOO) and Accidents and the Ultimate Response Guidelines (URG)

19.4.1 Emergency Operating Procedures (EOPs)

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.

Based on the above considerations, the symptom-oriented EOPs have been developed and implemented in all three operating NPPs 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 NPP were developed by the TPC. Differences between the EOP and EPG have been properly documented and justified. The resultant EOPs shall comply with the requirements of the NUREG-0737, Item I.C.1. To ensure that the proper operating 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.

The development of emergency operating procedures (EOPs) and severe accident management guidelines (SAMGs) of the Chinshan and Kuosheng NPPs was based on the generic BWR Owners Group (BWROG) emergency procedure guidelines (EPGs) and severe accident guidelines (SAGs). The development of EOPs and SAMGs of the Maanshan NPP was based on the Westinghouse Owners Group (WOG) generic procedures and guidelines. The EOPs and SAMGs are also reviewed and updated as the latest technology and research results become available. Regular training and exercises on these procedures are required for each NPPs.

19.4.2 Ultimate Response Guidelines (URG)

The operating conditions of a NPP can be classified into 4 categories:

(1) Normal operations,

(2) Abnormal events/transients,

(3) Accidents, and

(4) Severe accidents.

Thus, the NPP has in place 4 corresponding plant operating procedures for operators to follow as shown in Figure 19.1:

(1) Operating procedures,

(2) Abnormal operating procedures (AOP),

(3) Emergency operating procedures (EOP), and

(4) Severe accident management procedures (SAMP).

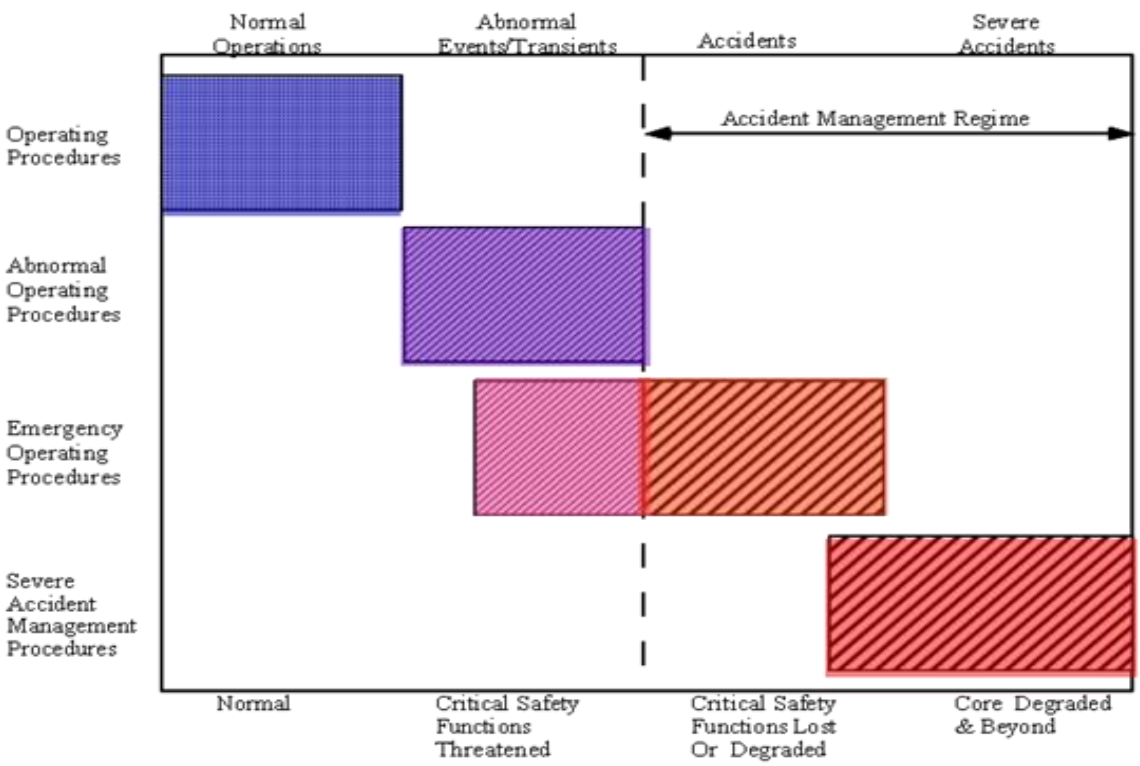


Figure 19.1 Relationship between Operating Conditions of a NPP and the Corresponding Plant Operating Procedures

As mentioned in Subsections 6.4.3.5 and 19.4.1, current AOPs and EOPs are based on symptom-basis. They are suitable for handling internal events. When there is a large-scale severe compound external event, like the Fukushima accident, whose effects are on the entire plant site, an urgent response will be required. Therefore, after the Fukushima Daiichi nuclear accident, each NPP of the TPC has developed its ultimate response guidelines (URG) with respect to the plant-specific (or site-specific) features.

The concept of the site-specific basis URG is to take a decisive action to get ready for an alternative cooling water injection by any means (even if the reactor may be damaged) in order to avoid the meltdown of reactor core and spent fuels (in SFP), in case the NPP faced a severe compound accident. The URG cooling water injection is a simultaneous process of reactor Depressurization, reactor coolant Injection, and containment Venting (DIVing). DIVing is the ultimate measure used to protect the NPP reactor from core melt.

The aim of URG is to develop an effective way for alternatively injecting cooling water into the RPV to avoid core meltdown through control of the reactor pressure and water level. The URG can be viewed as a defense-in-depth supplement to the EOP in order to prevent an accident from becoming a severe (core-melt) accident, similar to the concept adopted in NEI 12-06: "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide." Figure 19.2 shows a comparison between FLEX and URG. The URG can also be viewed as an improvement to the current EOPs based on lessons learned from the Fukushima accident. The relation among URG, AOP, EOP and SAMP is shown in Figure 19.3.

The plant procedure 1451 "Ultimate Response Guideline (URG)" is a new procedure to be established in the TPC's NPPs after the Fukushima Daiichi combined accident in order to guide the reactor operators and related decision-makers (plant shift managers) to prevent an accident from becoming a severe accident or to respond to the DBA and BDBA to ensure the availability of alternative ultimate heat sinks (UHS) including the raw water or even sea water.

In the URG, a two-step depressurization strategy is adopted for prolonged SBO. As an illustration, in the early stage of a SBO event in a BWR-type NPP (Chinshan or Kuosheng), a slow and controlled depressurization of the reactor is performed when the reactor core isolation cooling system (RCIC) is still available. In the second stage, a fast and emergency depressurization of the reactor is performed when the RCIC is no longer available. With the slow and controlled depressurization process, the reactor can be brought to and maintained at a relatively safe state (i.e., large water inventory with low pressure) while at the same time the RCIC trip set-point is not reached. Under this relatively safe state, the impact of the emergency depressurization of the reactor following the failure of RCIC will be minimized and the reactor water inventory can be supplemented by various low pressure injection methods. Otherwise, if the reactor is fast depressurized in the early stage of SBO at a relatively high danger state, it will lead the reactor core to become uncovered even though the water level is high before depressurization.

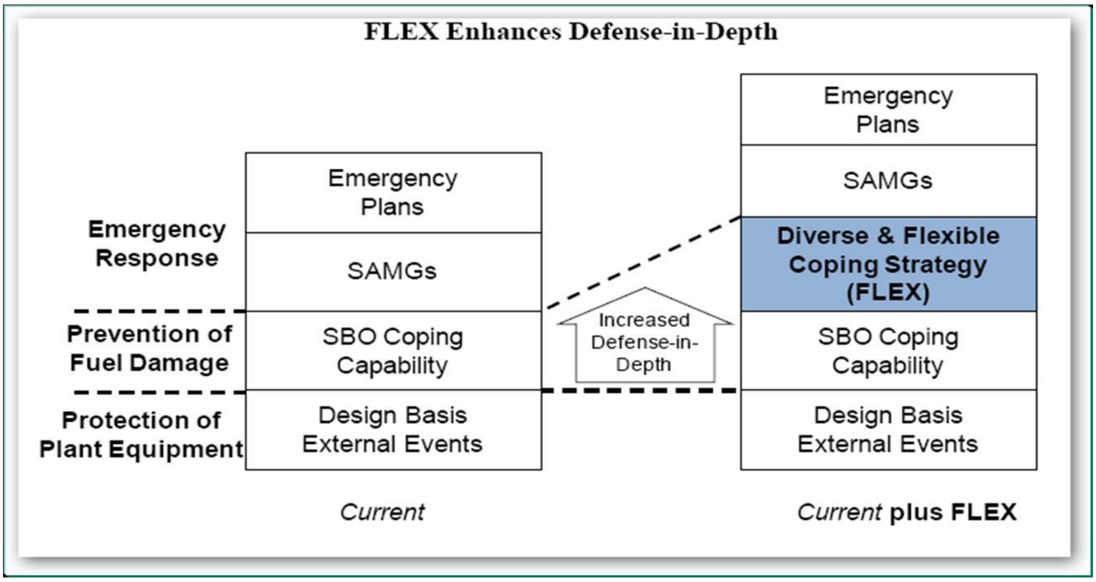
Normally, when an accident occurs, the EOP is to be employed. However, if any of the following three conditions is met, the EOP will be bypassed and the URG is to be initiated:

(1) Loss of all permanently installed AC power supply including the offsite power, onsite backup EDGs, and emergency backup swing EDG and GTGs,

(2) Loss of motor-driven reactor cooling water makeup capability except the steam-driven water makeup capability, or

(3) Automatic reactor scram due to strong earthquake and a tsunami alarm

announced concurrently.



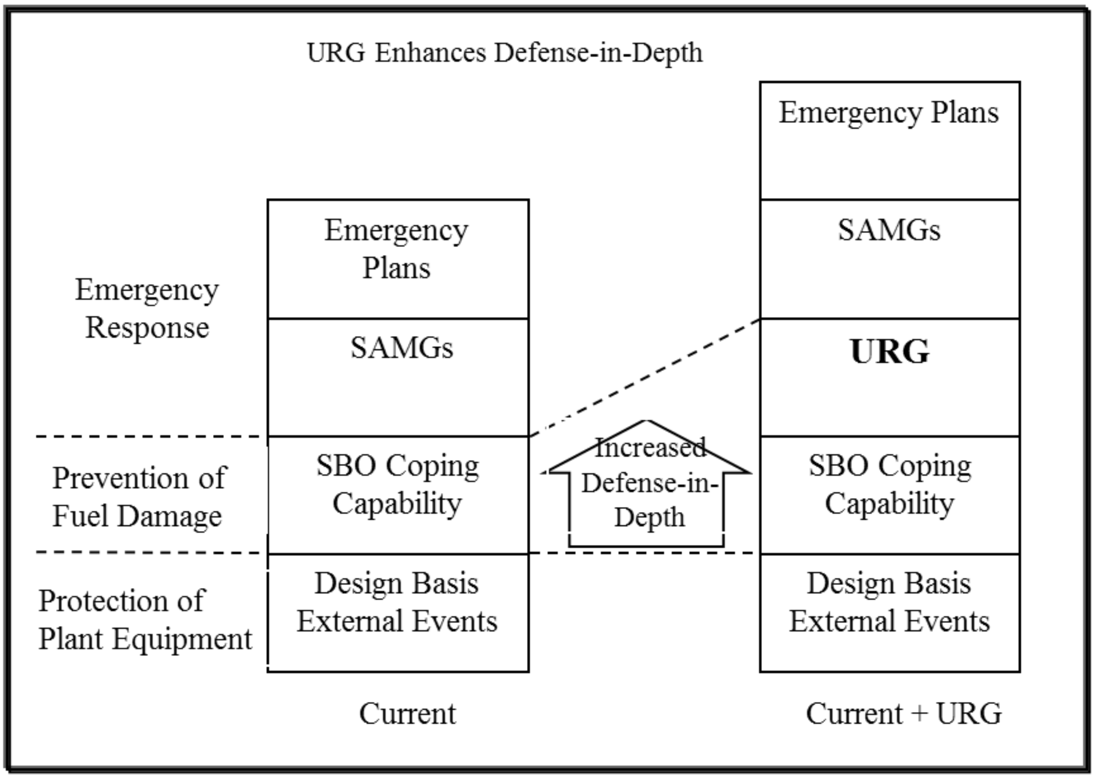


Figure 19.2 A Comparison between the FLEX and the URG

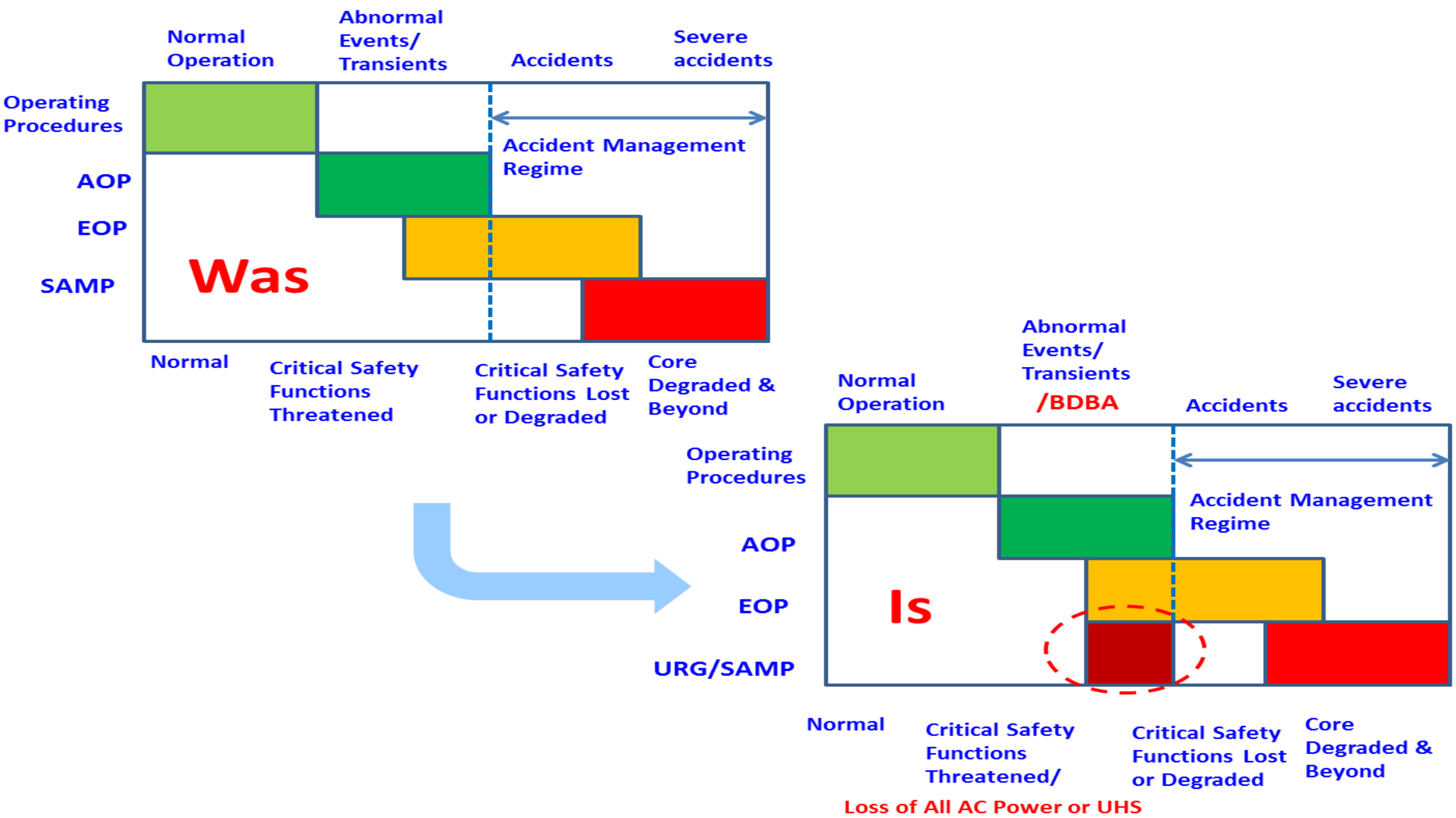


Figure 19.3 Relation among URG, AOP, EOP and SAMP

Figure 19.4 gives the relationship between EOP and procedures of URG. As shown in this figure, if the URG process is initiated, the following two-stage processes will be followed step-by-step:

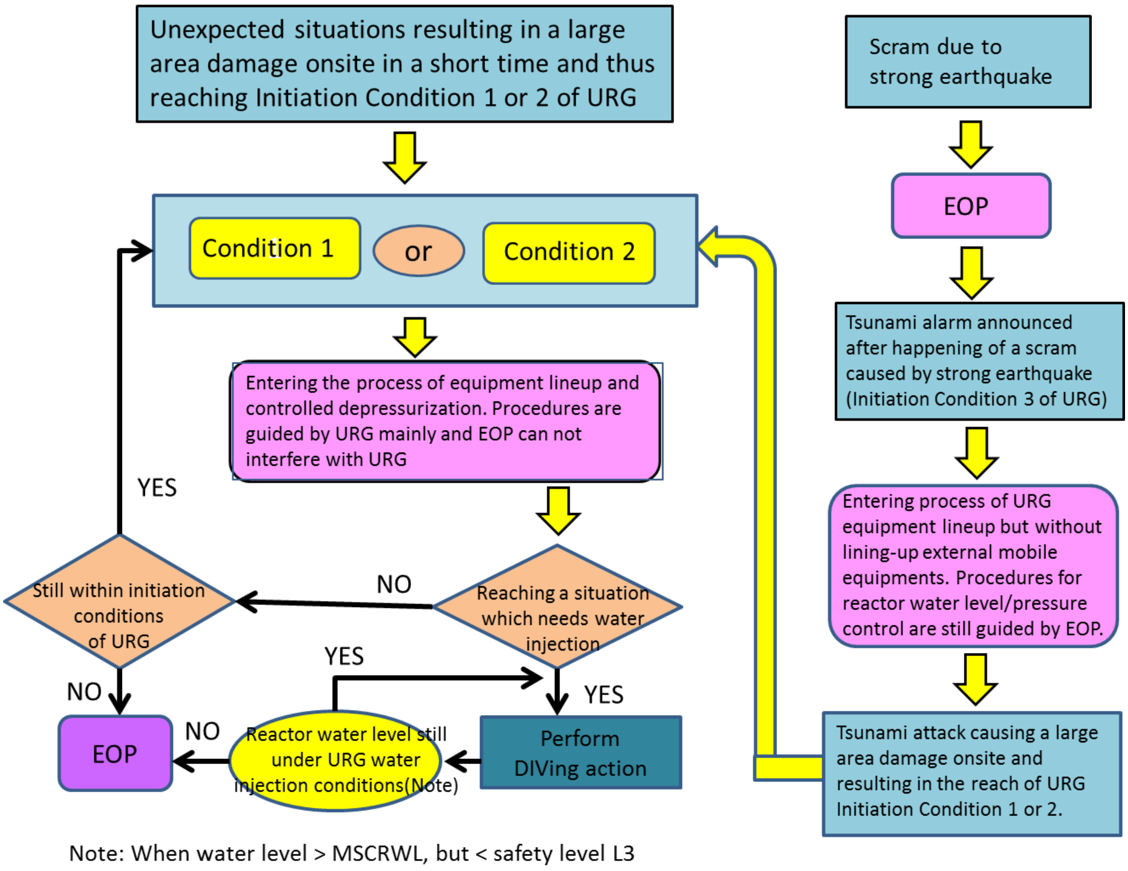


Figure 19.4 Relation between EOP and Procedures of URG
(MSCRWL: minimum steam cooling reactor (or RPV) water level)

(1) First Stage — Equipment Lineup & Controlled Depressurization (A Crucial Stage for the URG)

(1A) When there is an unexpected accident happened resulting in a large area damage onsite in a short time and thus the reaching of Initiation Condition 1 or 2 of URG, the process of URG will then be followed with EOP being bypassed (i.e., procedures are guided by URG mainly and EOP can not interfere with URG) and the following 2 actions will be implemented simultaneously:

(i) Equipment Lineup

Get ready within 1 hour the lineup of all equipment necessary for injecting cooling water from the service water, raw water and/or sea water to the reactor at low pressure. This process includes the lineup of equipment in the flow path of reactor cooling water injection and for the emergency depressurization of the reactor, containment venting to the atmosphere, SFP emergency makeup, etc., as well as the lineup of the mobile water injection equipment.

In the meantime, the NPP has to report to the Nuclear Safety Duty Center (NSDC) of AEC and the Chairman of the Nuclear Emergency Planning Executive Committee (NEPEC) of TPC. (The Chairman of NEPEC is also the TPC's Vice-President in Nuclear.)

(ii) Controlled Depressurization of the RPV

Start the slow and "controlled depressurization" of the RPV by actuating one SRV on the main steam line and, in the meantime, keep the RCIC system in operation in order to reduce the reactor pressure to 10 ~ 15 kg/cm^2 gradually and maintain the reactor water at high level.

(1B) If the URG initiation is due to Initiation Condition 3, then before the tsunami actually attacks the plant site, the above-mentioned equipment lineup process will be carried out but excludes lining-up of the external mobile equipment. However, at this time, procedures for the reactor water level/pressure control are still guided by EOP to depressurize and cool-down the reactor by discharging heat to the main condenser (MC). Once the tsunami attacks the site, then immediately stop the use of EOP and enter the procedures of URG to start the above-mentioned controlled depressurization process of the RPV. Lineup of movable equipment outside the plant buildings will be initiated after the tsunami has passed.

(2) Second Stage ─ URG Cooling Water Injection (DIVing Process)

Anytime whenever it is judged that the reactor is going to lose its core cooling and reactor coolant injection capability and fulfills the actuation condition for DIVing, the shift manager (SM) of the NPP will make the decision to start the DIVing process. One of the pre-requirements for the DIVing process to start the fast and "emergency depressurization" by opening all SRVs of the ADS is that the initial reactor pressure must be below 15 kg/cm^2.

In order to make sure the feasibility of the URG procedures, the TPC invited experts from the BWR Owners Group (BWROG) and the PWR Owners Group (PWROG) to hold Technical Forums in Taiwan in February 2014 and October 2014, respectively. On the other hand, four review meetings by the AEC for the URG had been held on July 17, 2012, March 28, 2014, July 22, 2015 and September 29, 2016, respectively. The operability of the plant-specific features in the URG has been justified on the basis of rigorous systematic review and thorough accident analysis and the AEC will keep close look on the development of the URG which is subject to the review of AEC. Besides, AEC requests the TPC's NPPs to comply with the requirements of FLEX in NEI 12-06 (per USNRC NTTF Report Tier 1 recommendation 4.2) and to strengthen and integrate the EOPs, SAMGs and EDMGs with the URG consistent with the USNRC NTTF Report Tier 1 recommendation 8. More information about the TPC's URG is given in Subsections 6.4.3.5, 10.2(11) and 18.4(C)(5).

19.5 Engineering and Technical Support

In each NPP itself, there is a Station Operation Review Committee (SORC) whose principal responsibility is to provide recommendations about the plant safety affairs for the plant general manager (PGM). There are 11 to 13 committee members with the PGM being the chairman of the SORC and 3 deputy PGMs the official members. The SORC is responsible mainly for:

- Reviewing the plant operating procedures and their modifications or changes,
- Reviewing the proposed tests and experiments which may affect the nuclear safety,
- Reviewing the proposed modifications or changes of the technical specifications (TS),
- Reviewing the proposed design change requests (DCRs) or modifications to the system or equipment which may affect the nuclear safety,
- Investigating the TS violation event and providing recommendation to prevent such event from happening again,
- Reviewing the abnormal events required by Section 16.6.9.2 of the FSAR to report,
- Etc.

As for the engineering and technical supports for the plant operations from outside 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 as well as maintenance. However, the principal backup supports for the plant operation are from various TPC departments in the headquarters (refer to Subsections 9.1.1 and 11.2.2.1 for organization of TPC), other TPC's NPPs, 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, DNE, other NPPs of TPC and the INER.

(2) Plant chemistry and health physics supports were provided by the DONG, other NPPs of TPC and the INER.

(3) Fueling and refueling operation supports were provided by the DONG, other NPPs of TPC and the INER.

(4) Maintenance support was provided by the DOM, DONG, other NPPs of TPC 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.

As mentioned in Subsection 6.3.8, the Institute of Nuclear Energy Research (INER) conducts the research and development (R&D) programs in recent years in the areas of nuclear safety such as the evaluations of license renewal for a NPP, the medium and small scale power uprates study, level 2 probabilistic risk assessments (PRA) of the operating NPPs, source term evaluation, seismic risk re-assessment of a NPP, high efficient solidification technology (HEST) study for the low level radioactive waste (LLRW or LLW), nuclear facility decommissioning and radioactive waste management, radiobiological medicines, the establishment of the accreditation platform for the nuclear grade industrial technologies, etc. Besides, INER can also form a technical team or establish a project for solving a particular safety issue when requested.

Besides the outside supports mentioned above, there is Station Operation Review Committee (SORC) in each NPP itself, whose principal responsibility is to provide recommendations about the plant safety affairs for the plant general manager (PGM). There are 11 to 13 committee members with the PGM being the chairman of the SORC and 3 deputy PGMs the official members. The SORC is responsible mainly for:

- Reviewing the plant operating procedures and their modifications or changes,
- Reviewing the proposed tests and experiments which may affect the nuclear safety,
- Reviewing the proposed modifications or changes of the technical specifications (TS),
- Reviewing the proposed design change requests (DCRs) or modifications to the system or equipment which may affect the nuclear safety,
- Investigating the TS violation event and providing recommendation to prevent such event from happening again,
- Reviewing the abnormal events required by Section 16.6.9.2 of the FSAR to report,
- Etc.

19.6 Reporting of Incidents Significant to Safety

19.6.1 Regulatory Requirements for Reporting Incidents

The requirements of reporting the abnormal or emergency events by the licensee timely are stipulated in Article 10 of the “Nuclear Reactor Facilities Regulation Act” of 2003, Article 7 of “Enforcement Rules for the Implementation of Nuclear Reactor Facilities Regulation Act” of 2003, and “Regulations on Immediate Notification Requirements and Reportable Event Report for Nuclear Reactor Facilities” of 2004.

According to the technical guidelines specified in the Appendix 2 of “Regulations on Immediate Notification Requirements and Reportable Event Report for Nuclear Reactor Facilities” of 2004, the licensee of an operating NPP 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 limiting conditions for operation (LCO) in the plant's TS,

(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 offsite 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 offsite 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 (i.e., reactor scram) 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 (NSDC), 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, consisting of Articles 17 to 19, of the Regulations on the Restart of Nuclear Reactor Facilities after Operating Outage of 2003 as amended in September 2005 and 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, which is required to report as specified in the technical specifications, a detailed report of the situation, corrective actions and measures to prevent recurrence must be submitted to the AEC within 30 days. The detailed requirements for this report are given in "Regulations on Immediate Notification Requirements and Reportable Event Report for Nuclear Reactor Facilities" of 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 (RE) for each of the TPC's three operating NPPs during the past ten years till August 2014 is shown in Table 19.1.

Table 19.1 Number of Reportable Events of the TPC's NPPs

Year	Number of Reportable Events			
	Chinshan	Kuosheng	Maanshan	Total
2004	2	4	1	7
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
2010	3	2	0	5
2011	2	3	1	6
2012	5	2	2	9
2013	2	1	2	5
2014	2	5	1	8

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 Penalty for Violations of Regulatory Requirements

19.7.1 Violations by the Nuclear Facilities

According to "Regulations on the Penalty for Violations of Regulatory Requirements by the Nuclear Facilities" of 1988 as amended in January 2008 (Articles 5, 9 to 15), the following penalty rules associated with violations of regulatory requirements by a nuclear installation are addressed.

Whenever there is a violation of regulatory requirements by a nuclear power facility, the regulatory body, AEC, will issue an administrative order to correct it. Violations of the regulatory requirements by a nuclear facility can be classified into five degrees. The first degree of violation is the most serious, while the 5th degree is the least. Appendix of the "Regulations on the Penalty for Violations of Regulatory Requirements by the Nuclear Facilities" gives the classifications of violations.

If a violation of the first or second degree happened in a NPP, which meant there was a serious violation, the AEC will immediately issue a correction order and may order the plant to stop the relevant activities, to reduce power, to stop reactor operation, or to implement special corrective measures, based on the relevant atomic energy laws and/or regulations. In addition, the AEC should release news of this violation to the media immediately.

If the violation is of the third degree, the NPP shall be required to submit a corrective action plan for approval by the AEC. In the meantime, the AEC may release news to the media, if necessary.

As for a fourth or fifth degree violation which is termed as the minor violation, the AEC may (or may not) ask the NPP to submit a corrective action plan for approval.

Table 19.2 shows the annual number of violations by all NPPs in recent years. The total number of violations showed a trend of decreasing.

Table 19.2 Annual Number of Violations by the NPPs

year	Number of Violations*					
	5th degree	4th degree	3rd degree	2nd degree	1st degree	Total
2011	7	7	1	1	0	16
2012	5	8	2	0	0	15
2013	7	4	0	0	0	11
2014	5	3	0	0	0	8

* The first degree violation is the most serious one, then the second violation, and so forth.

19.7.2 Violations by the Reactor Operators

Articles 27 to 32 of "Regulations on Nuclear Reactor Operators' Licenses" of 2004 as amended in December 2009 deal with the requirements on the behavior and performance of a reactor operator (RO) as well as the penalty in case of violation.

An RO on-duty is required to be examined by alcohol and drug tests. In addition, the reactor operator must pass the physical examinations as required by Article 12 of the above-mentioned Regulation.

In case any of the following violations happens, the AEC may suspend the license of the RO for 3 to 18 months depending on the degree of violation:

- Violation of the operation rules which may affect the safety of personnel, environmental ecology, reactor operation, or equipment,
- Being absent without official leave (AWOL) while on-duty,
- Ingestion of alcoholic drink while on-duty,
- Violation of requirement by Item 2 of Article 11 of “Regulations on Nuclear reactor Operators’ Licenses” of 2004 as amended in December 2009 in operating the nuclear reactor, and
- Refusing to take the physical examination as required.

If the violation belongs to one of the following, the AEC may revoke his RO license:

- Violating regulations or technical specifications (TS) with the result of loss of the safety protection function of a nuclear reactor or a severe degradation in the safety margins,
- Operating the nuclear reactor facility improperly with the result of safety margins of TS being overrun, while having his RO license been suspended before,
- Refusing to hand-in the license or stop operating the reactor after having his RO license been suspended,
- Failing to pass the drug test examination, and
- Ingesting alcoholic drink or being absent from his post without leave when on-duty, while having his RO license been suspended before.

(Other information about the penalty for violations of regulations can be found in Subsection 9.2.)

19.8 Operating Experience Feedback

19.8.1 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.

The AEC also dispatched its liaison representatives to station in Washington DC, Vienna and Paris in order to coordinate international exchange affair and nuclear cooperation activities with the NRC, IAEA and OECD/NEA, respectively. In the meantime, these representatives collected the most recent nuclear-related information and operating experience there and sent back to the AEC.

Besides, the AEC has required the TPC to constantly collect regulatory information issued by the USNRC.

On the other hand, the TPC obtains operating experiences from the General Electric Service Information Letter, Westinghouse Technical Bulletins, Boiling Water Reactors Owners Group (BWROG), Pressurized Water Reactor Owners Group (PWROG), 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.

19.8.2 Establishment of a System for the Feedback of Operating and Maintenance Experiences

To share the important operating and maintenance experiences among different NPPs in Taiwan, the TPC worked out a program, called the Operating 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.

19.8.3 Lessons Learned from Domestic and International Operating Experiences and/or Incidents

A deterministic approach complemented with the probabilistic risk assessment (PRA) studies and models is adopted for TPC's NPPs assessments related to loss of electrical power and loss of ultimate heat sink (UHS). The PRA database is regularly updated every 3 years according to the TPC's procedure. In this update process, the operational experience feedback (OEF) is systematically analyzed and the results are used as one of the inputs for the licensee's plant enhancement measures and for improvements of the regulatory requirements/guidance. For example, at the Maanshan NPP a diesel engine driven auxiliary feedwater pump (AFP) was installed as a result of the lesson learned from the specific SBO event occurred on March 18, 2001 at the Maanshan plant (which is locally called the "318 Event").

Another example of OEF was that AEC required TPC to improve the outage procedures for EDGs in accordance with lessons learned from the SBO event occurred on April 7, 2011 at the Japanese Higashidori NPP. (Refer to Subsection 6.4.2 for further information.)

19.8.4 Lessons Learned from Emergency Drills/Exercise

In Taiwan, an on-site nuclear emergency response drill (NERD) (or on-site nuclear exercise) is required for each NPP to be conducted every year to test the plant's capability to respond to an emergency, and giving the opportunity for relevant personnel to perfect their skills so that when facing to a real emergency they will remain calm, bring the accident rapidly under control and minimize the damage.

In addition, a national nuclear emergency exercise (NEE) is conducted both onsite and offsite annually at one of the operating NPPs in turn. The NEE exercises include radiation measurements, radiation dose evaluations, public sheltering and evacuation, accommodation of evacuees, mock distribution of iodine tablets, decontamination and emergency medical assistance.

The 2014 Nuclear Emergency Exercise (Code name: No. 20 NS Drill (Nuclear Safety Drill)) took place at the Kuosheng NPP and in New Taipei City and Keelung City in July and August 2014. It was assumed that an earthquake, happened off-shore of the Chinshan District of New Taipei City, caused the loss of all AC power in the Kuosheng NPP which resulted in a loss of coolant accident (LOCA). The effects of NPP safety improvement measures developed and implemented in the wake of the Japan Fukushima Daiichi NPP accident were examined and evaluated during the exercise.

A two-stage approach was adopted in this exercise: desktop practice (1 day) and field drill (3 days). Participants included staff from related emergency response agencies at the central and local governments, Taiwan Power Company, Taiwan Water Company, Taipei Veterans General Hospital, Chinshan Branch of National Taiwan University Hospital, New Taipei City Nursing Home and local residents. A total of about 2,000 people involved in the exercise.

Prior to the exercise, a series of communication activities were held, such as home visits, workshops, various group discussions and seminars in the school and press conferences, to let the public understand all the practices and efforts undertaken by the government.

Through this nuclear emergency exercise, AEC and TPC together with other emergency response agencies have validated plenty of the sophisticated response measures and emergency preparedness (EP) established since the "Programs for Safety Reassessment" was conducted after the Japan Fukushima accident.

Special features and results of this exercise are as follows:

- To validate the capability of a NPP in conducting the ultimate response guidelines (URG) procedures in case of loss of all the AC power and the sea ultimate heat sink (UHS).
- To validate the capability in conducting the air, sea and ground radiation monitoring.
- To practice the establishment of the Forward Coordination Post (FCP) of National Nuclear Emergency Response Center (NNERC).
- To validate the suitability and feasibility of local emergency response plan for public protection.
- To practice the emergency notification systems including the emergency siren, mobile/local broadcasting system, TV, radio and text messages, etc.

19.9 Radioactive Waste Management

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.

19.9.1 Low Level Waste Management

19.9.1.1 Low Level Waste Treatment and Storage

The AEC's low level radioactive waste (LLRW) (or simply low level 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 the end of August 2015, a total of 100,277 drums (55-gallon each) of the LLW are stored in Taiwan (not including those stored in the offshore islet Lanyu). Among them, more than 90 percent of the LLW, by volume, 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.

With the use of the High Efficiency Solidification Technology (HEST) developed by the Institute of Nuclear Energy Research (INER) and as a result of the plant staff's efforts, the annual generation of the solidified LLW from the 3 operating NPPs drastically dropped from a peak of nearly 12,000 drums (200 liters each) in 1983 to only about 176 drums in 2014, as shown in Figure 6.9.

19.9.1.2 Low Level Waste Final Disposal

On May 24, 2006, the "Act on Sites 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 protection 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.

In July 2012, two recommended candidate sites for local county referendum were announced by MOEA. However, both local counties of recommended candidate sites declined the MOEA's request for holding a referendum. TPC was asked by MOEA to continue strengthening communication and public relationship in both counties.

If a recommended candidate site passes the local referendum, it will become a formal candidate site. After further passing the Environmental Impact Assessment Reviewing Process, this candidate site can then be designated as a final disposal site by the Executive

Yuan.

19.9.2 Spent Nuclear Fuel Management

As for the spent nuclear fuel (SNF) management, the on-site interim dry storage is considered as a favorable option in Taiwan before implementing the final disposal.

In the 3 operating NPPs in Taiwan, the SNF discharged from the reactor cores are initially all stored in the existing spent fuel pool (SFP) of each nuclear power plant. The original storage capacity of the SFPs of these NPPs was too small and thus the SFPs of Chinshan and Kuosheng underwent re-racking work twice, while the SFP of Maanshan was re-racked once. However, even after this re-racking work, the SFPs of Chinshan NPP Unit 1 and Kuosheng NPP will be filled up in the near future. Therefore, installation of on-site spent fuel dry storage facilities for Chinshan and Kuosheng NPPs is needed urgently, so as to accommodate the SNF generated during their 40-year operation. The SFPs of Maanshan and Lungmen NPPs are expected to be able to accommodate the SNF generated during their 40-year operation.

The national strategy for present SNF management is:

(1) Storage in SFP for the short term,

(2) Onsite dry storage for the medium term, and

(3) Final disposal for the long term.

This management strategy will be properly adjusted according to the development of international situation.

19.9.2.1 Onsite Dry Storage of Spent Nuclear Fuel

As mentioned in Subsection 14.1.5(2), INER is the main contractor of the Chinshan independent spent fuel storage installation (ISFSI) program for onsite dry storage of SNF. The dry storage system of Chinshan is the INER High Performance System (INER-HPS). The INER-HPS dry storage cask is a vertical concrete cask (VCC) (Figure 19.5). The TPC planned to install 30 INER-HPS casks. Each cask had the capacity of 56 spent fuel assemblies. The total capacity of the Chinshan NPP dry storage facility is 1,680 spent fuel assemblies.

The TPC submitted an application for the construction license (CL) of the Chinshan ISFSI to the AEC on March 2, 2007 and AEC issued the CL to the TPC on December 3, 2008. The construction of Chinshan ISFSI started on October 18, 2010. The AEC then carried out the construction inspection to ensure the quality of the facility. The soil preparation (including water and soil conservation) of the Chinshan ISFSI site started in January 2011 and was completed in June 2013. Construction of the concrete base and installation of auxiliary equipments of the Chinshan ISFSI were completed in February 2013.

On May 23, 2012, the AEC granted the pre-operation plan of the Chinshan ISFSI proposed by the TPC. The first stage pre-operation (or the overall functional tests) was conducted in June 2012 and completed on November 14, 2012. The TPC submitted the overall functional tests report (the “Performance Test and Verification Report”) to AEC on March 8, 2013 for review. As a result, the AEC authorized the conductance of the second stage pre-operation (or the hot test) on September 24, 2013. However, due to lack of the

Soil and Water Conservation Facility Completion Approval issued by the local government (the New Taipei City), the hot test of Chinshan ISFSI was still idle until now.

Figure 19.5 Appearance of an INER-HPS Dry Storage Cask

Fabrication of all the planned 25 vertical concrete casks of the Chinshan dry storage system was completed in September 2014 with the related handling and other auxiliary equipment installed and passed the functional tests.

On the other hand, the main contractor for the Kuosheng program is Nuclear Assurance Corporation (NAC) International, USA. The dry storage system of Kuosheng is NAC-MAGNASTOR-87 (Figure 19.6) with a capacity of 87 fuel assemblies per VCC and 27 VCCs in maximum (2,349 assemblies).

The Kuosheng ISFSI program was approved by MOEA on August 10, 2009. The TPC then carried out the tender operation and made an awarding on November 12, 2010 to entrust the CTCI Machinery Corporation (Taiwan) and NAC International (USA) to construct the facility. The MAGNASTOR cask system, as shown in Figure 19.6, which was designed by the NAC, was adopted by TPC for the Kuosheng NPP dry storage facility. The cask system had already gotten the license for spent fuel dry storage from the USNRC. Each MAGNASTOR cask has the capacity of 87 spent fuel assemblies and the TPC planned to install 27 casks. So, the total capacity of the Kuosheng NPP dry storage facility is 2,349 spent fuel assemblies.

TPC submitted to AEC the application for the construction of the Kuosheng ISFSI on February 14, 2012. After six rounds of technical review, the AEC on September 3, 2013 held a conclusive review meeting and approved the application.

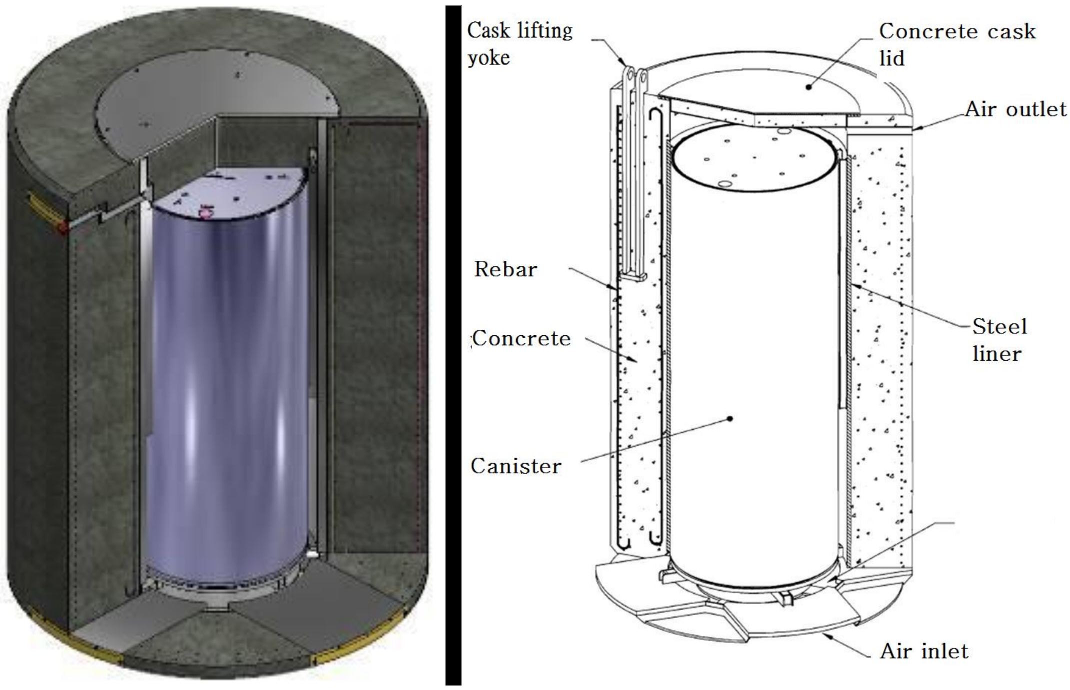


Figure 19.6 A Schematic Drawing of the Kuosheng Dry Storage Cask (MAGNASTOR)

19.9.2.2 Final Disposal of Spent Nuclear Fuel

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 SNF.

Since December 1983, the AEC, TPC, INER, Central Geological Survey (CGS), and Industrial Technology Research Institute (ITRI) have organized a task force to draft the "Research Plan on Disposal of Spent Nuclear Fuel" and carried out four stages of research and development (R&D) on the final disposal of high level radioactive waste (HLRW). The four stages of this R&D program were stages for "preliminary research and development of the final disposal concept (1986 ~ 1988)," "initial work plan (1988 ~ 1991)," "technical preparation for site area investigation (1993 ~ 1998)," and "conductance of investigation and development of technology (1999 ~ 2004)," respectively.

On December 25, 2002, the AEC promulgated the "Nuclear Materials and Radioactive Waste Management Act." In compliance with this newly promulgated Act, the TPC submitted a "Spent Nuclear Fuel Final Disposal Plan (abbreviated as the Disposal Plan)" in 2004, which was approved by the AEC in 2006. The Disposal Plan is divided into the following 5 stages (as shown in Figure 19.7):

Stage 1: Characterization and Evaluation of Potential Host Rocks (2005 ~ 2017),

Stage 2: Investgation and Confirmation of Candidate Sites (2018 ~ 2028),

Stage 3: Detailed Site Investigation and Testing (2029 ~ 2038),

Stage 4: Design and License Application of the Repository (2039 ~ 2044), and

Stage 5: Construction of the Repository (2045 ~ 2055)

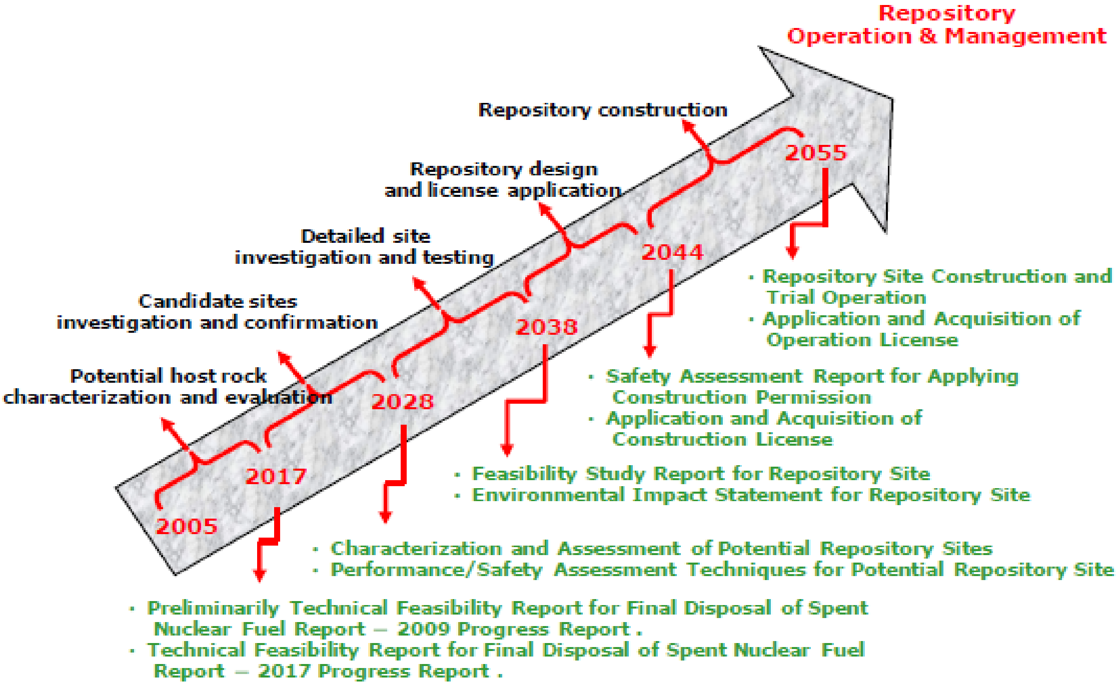


Figure 19.7 Long-Term Plan of Taiwan's SNF Final Disposal Program

In accordance with Article 37 of the "Enforcement Rules for the Nuclear Materials and Radioactive Waste Management Act" of 2008, the Disposal Plan shall be reviewed and amended every 4 years. Thus, the TPC amended the Disposal Plan twice and the amended plans were submitted in May 2010 and 2014, respectively and then approved by the AEC in January 2011 and May 2015, respectively. The repository is scheduled to start operation in 2055.

Currently (in 2016), the R&D program of the Disposal Plan is at the first stage. A progress report entitled "Preliminary Technical Feasibility Study for Final Disposal of Spent Nuclear Fuel (abbreviated as SNFD2009 report)" was submitted to AEC in September 2009 and approved by AEC in July 2010. This SNFD2009 report compiled the R&D results of Taiwan's spent fuel disposal programs over the past 20 years. It confirmed that there exist some potential host rocks in certain regions of Taiwan which are worthy of further investigation. The relevant reports were published on the AEC's website.

According to the Disposal Plan, the first stage: "Characterization and Evaluation of Potential Host Rocks (2005 ~ 2017)" will be completed in 2017. It is scheduled that the "Technical Feasibility Assessment Report on Spent Nuclear Fuel Final Disposal" will be completed and an international peer review will then be conducted by 2017 to validate the Taiwanese task group's technical capability of SNF final disposal.

Should the international promotion of spent fuel disposal programs be subject to hindrance and delayed schedules, the TPC has proposed an alternative contingency plan for the SNF final disposal program. According to this plan, after the completion of stage two, if TPC is still unable to propose a candidate site by 2028, it will start an alternative measure in 2029 by considering a centralized SNF storage facility. The site for the centralized SNF storage facility will be confirmed and its environmental impact assessment (EIA) will be completed by 2038. Construction of the centralized spent fuel storage facility shall be completed and its operation shall start by 2044, as shown in Figure 19.8.

Considering that reprocessing of the SNF not only recycles valuable resources like uranium and plutonium but also reduces the volume of high level waste, TPC is exploring the feasibility of shipping a small portion of spent fuel for overseas reprocessing in parallel with the spent fuel final disposal plan. According to the Trilateral Nuclear Safeguards Agreement and TECRO-AIT 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.

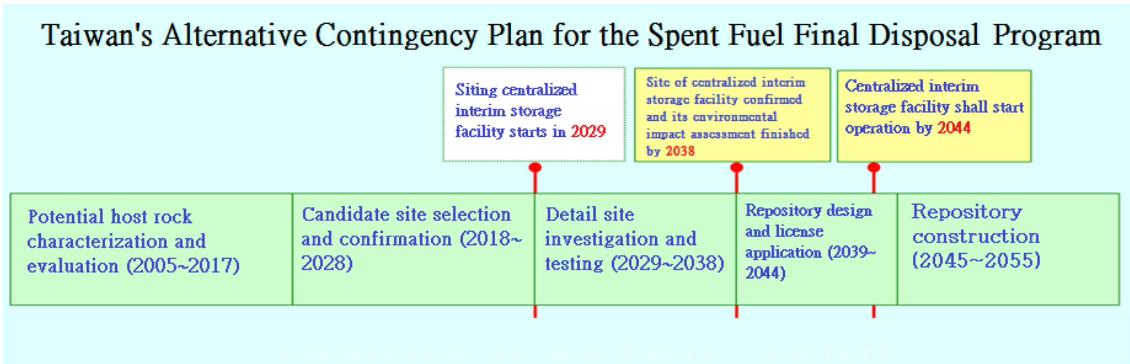


Figure 19.8 An Alternative Contingency Plan for the SNF Final Disposal Program

19.10 Transparency of Nuclear Information

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 safe operation of the plant and for an easily understood indicator of the safety status of an operating NPP to the public.

With sincere attitude and fair view-points the TPC also presents information about the company in six aspects to the public through the TPC's web site. The six aspects include information on management, power generation, demand & supply of electricity, customers, environment, and construction engineering. In the aspect of power generation, for example, one can obtain information about the current status and performance records of the fossil power and/or nuclear power management, renewable energy, electricity purchased by TPC from the independent power producers (IPPs) (private utilities), and measures in response to the Fukushima Daiichi nuclear accident.

More about the transparency of TPC’s nuclear power information can also be found on the TPC “核能看透透” Website: http://wapp4.taipower.com.tw/nsis.

APPENDIX A　ACRONYMS

Abbriviation	Full Name
10 CFR	United States Title 10 of the Code of Federal Regulations
ABWR	advanced boiling water reactor
AC or ac	alternating current
ACF	annual capacity factor
ADS	automatic depressurization system
AE	alert event
A/E	architect/engineer
AEC	Atomic Energy Council (Taiwan)
AECDNT	Department of Nuclear Technology, AEC (Taiwan)
AFB	auxiliary fuel building
AFP	auxiliary feedwater pump
AFS	auxiliary feedwater system
AFW	auxiliary feedwater (system)
AH	ampere-hour
AIT	American Institute in Taiwan
ALARA	as low as reasonably achievable
AMT	accident management team
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AOP	abnormal operating procedure
AOV	air-operated valve
ASME	American Society of Mechanical Engineers
ASTS	automatic seismic trip system
ATRIUM-10	Upgraded BWR Fuel Assemblies, AREVA
ATWS	anticipated transient without scram
BDB	beyond design basis
BDBA	beyond design basis accident
BDBE	beyond design basis earthquake
BKR	breaker
BOE	Bureau of Energy, MOEA

BOP	balance of plant
BSMI	Bureau of Standards, Metrology and Inspection, MOEA
BWR	boiling water reactor
BWR-4	boiling water reactor of type 4
BWR-6	boiling water reactor of type 6
BWROG	BWR Owners Group
CAMP	Code Applications and Maintenance Program
CAP	corrective action program
CCP	centrifugal charging pump (PWR)
CF	capacity factor
CFR	Code of Federal Regulations (of US)
CGS	Central Geological Survey
CL	construction license
CLB	current licensing basis
CM	configuration management
CNC	Commission of National Corporations, MOEA (a supervisory organization of the TPC)
CNS	Convention on Nuclear Safety, IAEA
COA	Council of Agriculture
CODAP	Component Operational Experience, Degradation and Ageing Program
COL	combined license (or combined construction and operating license)
COLR	core operating limits report
COOPRA	Cooperative PRA Research Program
COPS	containment overpressure protection system
CP	construction permit
CPD	Cooperative Program on Decommissioning
CRE	collective radiation exposure (one of WANO's PIs)
CS	Chinshan (NPP)
CSARP	Cooperative Severe Accident Research Program
CSC	Convention on Supplementary Compensation for nuclear damage
CSCW	combined structure cooling water system or combination structure cooling water system
CSNPP	Chinshan Nuclear Power Plant
CST	condensate storage tank

CTCI	CTCI Corporation (中鼎工程股份有限公司)
CTS	customer technical specifications
CWS	circulating water system
DB	design basis
DBA	design basis accident
DBE	design basis earthquake
DBE	design basis event
DBT	design basis threat
DC	direct current
DCR	design change request
DG or D/G	diesel generator
DGBAS	Directorate-General of Budget Accounting and Statistics
DIVing	system Depressurization, water Injection and containment Venting (a process of reactor emergency Depressurization, reactor coolant Injection, and containment Venting simultaneously used in URG)
DNBM	Department of Nuclear Backend Management, TPC
DNE	Department of Nuclear Engineering, TPC
DNR	Department of Nuclear Regulation, AEC
DNS	Department of Nuclear Safety, TPC
DNT	Department of Nuclear Technology, AEC
DOE	Department of Energy, US
DOM	Department of Maintenace, TPC
DONG	Department of Nuclear Generation, TPC
DOP	Department of Planning, AEC
DPGM	deputy plant general manager
DPRO	Disaster Prevention and Relief Office of the Executive Yuan
D/Q	deposition factor
DRP (AEC)	Department of Radiation Protection, AEC
EA	exclusion area
E&C	E&C Engineering Corp. (益鼎工程股份有限公司)
EC	European Commission
ECCS	emergency core cooling system
EC/ENSREG	ENSREG of EC
ECS	emergency cooling system

ECW	emergency circulating water system (for KSNPP)
ECW	essential chilled water system (for CSNPP)
EDG	emergency diesel generator
EDMG	extensive damage mitigation guidelines
EE	engineering evaluation
EHV	extra high voltage
EIA	environmental impact assessment
EIS	environmental impact statement
El.	elevation
ENSREG	European Nuclear Safety Regulators Group
EOC	end of cycle
EOF	Emergency Operation Facility
EOP	emergency operating procedures
EP	emergency preparedness
EPA	Environmental Protection Administration, ROC (環保署)
EPG	emergency procedure guidelines
EPIC	emergency public information center
EPRI	Electric Power Research Institute (of US)
EPU	extended power uprates
EPZ	emergency planning zone
ERBP	Emergency Response Basic Plan
ERF	Emergency Response Facility
ERM	Environmental Radiation Monitoring
ERMN	Environmental Radiation Monitoring Network
ERMS	Environmental Radiation Monitoring System
ERT	Expert Review Team (of Executive Yuan)
ESF	engineered safety features
ESW	essential service water system (Chinshan)
ESW	emergency service water system (Kuosheng)
ETA	ethanolamine (ethanol 酒精)
EU	European Union
EU-PR	European Union Peer Review
EU-PRT	European Union Peer Review Team

EY	Executive Yuan
FCMA	Fuel Cycle and Materials Administration (物管局)
FCP	Forward Coordination Post (前進協調所)
FCVS	filtered containment venting systems (for PWR)
FLEX	diverse and flexible coping strategies (NEI 12-06)
FLR	forced loss rate (one of WANO's PIs)
FMCRD	fine motion control rod drive
FME	foreign material exclusion
FR	fuel reliability (one of WANO's PIs)
FSAR	final safety analysis report
FSC	Financial Supervisory Commission
FSG	FLEX Support Guidelines
FW	feedwater
FY	fiscal year
g	standard value of the gravitational acceleration (1 g = 9.81 m/s^2)
GDC	general design criteria (10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants")
GE	General Electric Company
GEA	general emergency accident
GIO	Government Information Office
GT or G/T	gas turbine (= gas turbine generator)
GTBSDG	gas turbine black start diesel generator
GTG	gas turbine generator (= gas turbine)
GV	governor valve
GWe	giga–watts electrical (= 10^9 watts)
h	hour
HEP	human error probabilities
HEST	high efficiency solidification technology
HFE	human factor engineering
HIS	hydrogen ignition system
HLRW	high level radioactive waste
HPC	health physics center
HPCI	high pressure coolant injection (system)
HPCS	high pressure core spray (system)

HPES	human performance enhancement system
HPIC	high pressure ionization chamber
HPS	high performance system, ISFSI
HRA	human reliability analysis
HSI	human-system interface
HX	heat exchanger
I&C	instrumentation and control
IAEA	International Atomic Energy Agency
ICP-MS	Induction Coupled Plasma Mass Spectrum
ICRP	International Commission on Radiological Protection
IEEE	Institute of Electrical and Electronics Engineers
IFL	initial fuel loading
IMS	Information Management System
IN	Information Notice (of NRC)
INER	Institute of Nuclear Energy Research, AEC
INER-HPS	INER High Performance System (dry storage cask)
INEX	International Nuclear Emergency Exercises
INPO	Institute of Nuclear Power Operations
INSAG	International Nuclear Safety Advisory Group
IPA	integrated plant assessment
IPP	independent power producer
IRA	integrated reliability analysis
IRPA	Ionizing Radiation Protection Act
ISA	integrated safety assessment (for PSR every 10 years)
ISAR	integrated safety assessment report
ISFSI	independent spent fuel storage installation
ISO	International Organization for Standardization
ITRI	Industrial Technology Research Institute
ITS	improved technical specifications
JLD	Japan Lessons-learned project Directorate
kg or Kg	kilogram
KLOE	kilo-liter oil equivalent
km	kilometer
KS	Kuosheng (NPP)
KSNPP	Kuosheng nuclear power plant

kV	kilovolt
kVA	kilovolt-ampere
kW	kilowatt
kWh	kilowatt-hour
L2	Level 2
L3	Level 3
LC	load center (負載中心)
LCM	life cycle management
LCO	limiting conditions for operation
LLW	low level waste (= LLRW)
LLRW	low level radioactive waste (= LLW)
LM	Lungmen (NPP)
LMNPP	Lungmen nuclear power plant
LOCA	loss of coolant accident
LOOP	loss of offsite power
LOV	loss of voltage
LP&SD	low power and shutdown
LPZ	low population zone
LY	Legislative Yuan
m	meter
MAAP	Modular Accident Analysis Program
MACCS2	the computer code MACCS2
MC	main condenser
MCC	motor control center
MCR	main control room
MDG	mobile diesel generator
mGy	milli-gray
mGy/yr	milli-gray per year
mGy/yr-unit	milli-gray per year per unit
MHI	Mitsubishi Heavy Industries, Japan
MIRU	Maintenance Integrated Risk Utilities (computer program)
MMCS	maintenance management computerization system (computer program)
MMI	man-machine interface
MMS	moment magnitude scale (M_w)

	(The Richter scale was succeeded in the 1970s by the moment magnitude scale (MMS). This is now the scale used by the United States Geological Survey to estimate magnitudes for all modern large earthquakes.)
MOE	Ministry of Education
MOEA	Ministry of Economic Affairs, Taiwan
MOF	Ministry of Finance
MOFA	Ministry of Foreign Affairs, ROC (Taiwan)
MOHW	Ministry of Health and Welfare
MOI	Ministry of Interior
MOL	Ministry of Labor
MOND	Ministry of National Defense
MOST	Ministry of Science and Technology
MOTC	Ministry of Transportation and Communication
MOV	motor-operated valve
MR	maintenance rule
MS	Maanshan (NPP)
MSCRWL	BWR minimum steam cooling reactor (or RPV) water level (最小蒸汽冷卻水位)
MSL	mean sea level
MSNPP	Maanshan nuclear power plant
μSv/h	micro-sievert per hour
mSv	milli-sievert
mSv/yr	milli-sievert per year
mSv/yr-unit	milli-sievert per year per unit
MUR	measurement uncertainty recapture
MVA	million volt-ampere
M_w	moment magnitude scale (of earthquake) (The Richter scale was succeeded in the 1970s by the moment magnitude scale (MMS). This is now the scale used by the United States Geological Survey to estimate magnitudes for all modern large earthquakes.)
MW	megawatts (= 10^6 watts)
MWe	megawatts electrical (= 10^6 watts)
MWt	megawatts thermal (= 10^6 watts)
NAC	Nuclear Assurance Corporation International, USA
NCC	National Communications Commission

NCT	Nuclear Communication Team, TPC
NDCORE	National Database Center of Occupational Radiation Exposures
NDE	non-destructive examination
NDT	non-destructive test
NEA	Nuclear Energy Agency (of OECD)
NEE	Nuclear Emergency Exercise (onsite + off-site) (= NS Drill 核安演習)
NEI	Nuclear Energy Institute (formerly NUMARC and USCEA)
NEPEC	Nuclear Emergency Planning Executive Committee, TPC (核能發電廠緊急計畫執行委員會(簡稱緊執會))
NERC	National Emergency Response Center
NERD	nuclear emergency response drill (onsite)
NERF	Nuclear Emergency Response Fund
NERO	Nuclear Emergency Response Organization, TPC
NERU	Nuclear Emergency Response Unit, TPC
NESC	Nuclear Emergency Support Center
NEST	Nuclear Energy Society, Taipei
NFA	National Fire Agency of MOI, ROC
NISA	Japanese Nuclear and Industrial Safety Agency
NNAERP	National Nuclear Accident Emergency Response Plan (replaced by ERBP by AEC in July 2005)
NNEO	National Nuclear Emergency Organization
NNERC	National Nuclear Emergency Response Center (中央核子災害應變中心)
NNSA (DOE/NNSA)	US National Nuclear Security Administration of the Department of Energy
NPP	nuclear power plant
NPT	Non-Proliferation Treaty
NQA	Nuclear Quality Assurance
NRA	Nuclear Regulation Authority, Japan
NRC	Nuclear Regulatory Commission, U.S.
NRF	nuclear reactor facility
NSC	Nuclear Safety Committee (TPC)
NSCW	nuclear service cooling water system
NSDC	Nuclear Safety Duty Center, AEC (核安監管中心)

NS Drill	Nuclear Safety Drill (= NEE)
NSSS	nuclear steam supply system
NTHU	National Tsing Hua University
NT$	New Taiwan Dollar
NTTF	Near-Term Task Force (of NRC)
NucNet	Independent Global Nuclear News Agency
NUPIC	Nuclear Procurement Issues Committee, US
NUREG	nuclear regulatory
NuSTA	Nuclear Science and Technology Association, Taiwan
O&M	operation and maintenance
OBE	operating basis earthquake
OE	operating experience
OECD	Organization for Economic Cooperation and Development
OECD/NEA	Nuclear Energy Agency of OECD
OEF	operational experience feedback
OJT	on-the-job training or on-job training
OL	operating license
OLTP	original licensed thermal power
OSC	Operation Support Center (or Operating Support Center)
PAG	protective action guides
PAR	passive autocatalytic recombiner
PBNC	Pacific Basin Nuclear Conference
PC	power center (對應於 MCC)
PDA	personal digital assistant
PECL	Pacific Engineers & Constructors, Ltd. (泰興工程顧問股份有限公司)
PGA	peak ground acceleration
PGM	plant general manager
PI	performance indicator
PMP	probable maximum precipitation (最大可能降雨量)
PNC	Pacific Nuclear Council
PORV	power operated relief valve
PRA	probabilistic risk assessment
PRiSE	a PRA model based Risk Significance Evaluation tool
PRT	peer review team

PSAR	preliminary safety analysis report
PSER	preliminary safety evaluation report
PSHA	probabilistic seismic hazard analysis
PSR	periodic safety review
PU	power uprate
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
Q	safety-qualified
Q1	1st quarter
Q2	2nd quarter
Q3	3rd quarter
Q4	4th quarter
QA	quality assurance
QC	quality control
R&D	research and development
RAI	required additional information
RAMP	RAdiation protection code analysis and Maintenance Program
RBCW	reactor building cooling water (system)
RBSW	reactor building service water (system)
RCCV	reinforced concrete containment vessel
RCIC	reactor core isolation cooling (system)
RCP	reactor coolant pump
RCS	reactor coolant system
RDSS	research and development substitute service (研發替代役)(內政部研發替代役制度資訊管理系統網址：rdss.nca.gov.tw)
RE	reportable event
RER	reportable event reports
RG	Regulatory Guide, NRC
RHR	residual heat removal (system)
RIS 2002-03	NRC Regulatory Issue Summary 2002-03
RL	Radiation Laboratory, TPC (台灣電力股份有限公司放射試驗室)
RLE	review level earthquake
RMC	Radiation Monitoring Center, 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
RPCS	Radiation Protection Control System
RPS	reactor protection system
RPV	reactor pressure vessel
RS-001	Review Standard for Extended Power Uprates
RSAR	reload safety analysis report for the BWR
RSER	reload safety evaluation report for the PWR
RW	radioactive waste (or radwaste)
RWA	Radioactive Waste Administration, AEC
SAEA	site area emergency accident
S & W	Stone & Webster Construction, Inc.
SAG	severe accident guidelines
SAMG	severe accident management guidelines
SAR	safety analysis report
SAT	Systematic Approach to Training
SBO	station blackout (≡ a complete loss of normal offsite ac power and onsite backup EDGs power, not including the swing EDG and gas turbines for emergency backup) (= loss of all permanently installed AC power sources)
SC	safety culture
SDP	significance determination process
SDR	Special Drawing Rights (特別提款權) : 又稱為「紙黃金」，是國際貨幣基金組織（IMF）於 1969 年進行第一次國際貨幣基金協定修訂時創立的用於進行國際支付的特殊手段，1 SDR 約等於 50 元新台幣。
SDVM	steam dryer vibration monitoring
SEDG	swing emergency diesel generator
SER	safety evaluation report
SER	significant event reports (of WANO)
SF	safety fundamentals
SFP	spent fuel pool

SFPACS	spent fuel pool additional cooling system
SG	steam generator
SL	shift leader
SM	shift manager
SMA	seismic margin assessment
SMAS	special skill alternative service
SMS	strong-motion seismometer (強震儀)
SNF	spent nuclear fuel
SNFD2009	Spent Nuclear Fuel (final) Disposal report in 2009 of TPC
SOER	significant operating experience reports (of WANO)
SOP	standard operating procedure
SORC	station operation review committee, TPC
SPDS	safety parameter display system
SPRA	seismic probabilistic risk assessment (or Seismic PRA)
SPU	stretch power uprates
SR	surveillance requirements
SRM	Staff Requirements Memorandum
SRO	senior reactor operator
SRV	safety relief valve
SSAS	special skill alternative service (特殊專長替代役)
SSC	structures, systems and components
SSE	safe shutdown earthquake
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	soil-structure interaction
SSP	safety system performance (one of WANO's PIs)
ST	stress tests
STS	standard technical specifications
SUJB	State Office for Nuclear Safety of the Czech Republic.
SV	safety valve
TAF	Taiwan Accreditation Foundation
Taipower	Taiwan Power Company
TBD	To be determined
TDAFP	turbine-driven auxiliary feedwater pump
TECRO	Taipei Economic and Cultural Representative Office in the US
TEPCO	Tokyo Electric Power Company

TG or T/G	turbine generator
TIT	Taipower Institute of Training
TLAA	time-limited aging analysis
TLD	thermo-luminescent dosimeter
TMI	Three Mile Island NPP
TMI accident	1979 accident at Three Mile Island NPP Unit 2
TPC	Taiwan Power Company
TRIM	TPC Risk Integrated Monitor
TRM	technical requirement manual
TRMS	Taiwan Radiation Monitoring Station, AEC
TS	technical specifications
TSC	technical support center
TSM	technical support missions
UA7	unplanned automatic scrams per 7,000 hours critical (one of WANO’s PIs)
UAT	unit auxiliary transformer
UCF	unit capability factor (one of WANO’s PIs)
UCLF	unplanned capability loss factor (one of WANO’s PIs)
UFSAR	updated final safety analysis report
UHS	ultimate heat sink
	(sea water) ultimate heat sink
UK	United Kingdom
UN	United Nations
UPS	uninterruptible power supply
URG	ultimate response guidelines (機組斷然處置程序指引)
US	United States
USA	United States of America
USNRC	United States Nuclear Regulatory Commission
V	volt
VAC	volt alternating current
V&V	verification and validation
VCC	vertical concrete cask, ISFSI
VDC	volt direct current
VDNS	Vienna Declaration on Nuclear Safety
W	Westinghouse Electric Corporate

WANO	World Association of Nuclear Operators
WANO-TC	World Association of Nuclear Operators – Tokyo Center
WMS	weak-motion seismometer (弱震儀)
WOG	Westinghouse Owners Group
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 NPP (1st NPP), TPC, Maanshan NPP (3rd NPP), TPC,

Kuosheng NPP (2nd NPP), TPC, Lungmen NPP (4th NPP), TPC,

Department of Nuclear Back-end Management (DNBM), TPC,
Department of Nuclear Generation (DONG), TPC,
Department of Nuclear Engineering (DNE), TPC,
Department of Nuclear Safety (DNS), TPC,
Nuclear Emergency Planning Executive Committee, TPC,
Nuclear Emergency Response Unit, TPC,
Taiwan Power Company (TPC) Headquarters,
Department of Nuclear Regulation (DNR), AEC,
Department of Nuclear Technology (DNT), AEC,
Department of Planning, AEC,
Department of Radiation Protection, AEC,
Fuel Cycle and Materials Administration (FCMA), AEC,
Radiation Monitoring Center (RMC), AEC, and
Nuclear Regulatory Technology Support Center, AEC (核能管制技術支援中心)

Editors and main contributors to this report were:

Yang, Chao-Yie (楊昭義), Chief Editor,
Lee, Chien Hsiung (李堅雄), and
Chen, H. T. (陳渙東).

Other contributors included:

Lin, G. T. (林繼統) – AEC,
Liao, Lih-Yih (廖俐毅) – INER,
Yang, C. W. (楊慶威) – INER,
Cheng, S. Y. (陳勝裕) – INER,
Chow, Ting (周鼎) – INER,
Lo, L. C. (羅龍珠) – INER, and
many other technical and regulatory experts from the AEC and INER.

The coordinator for the whole project were C.W. Yang (楊慶威) in the beginning and S. Y. Cheng (陳勝裕) subsequently.

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	1,840
Installed Electrical Power Capacity, 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: Reload Cycle	1.90 4.0
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 at high water level, ft^3	71,658
Total Free Space Volume at high water level, ft^3	101,342
DESIGN BASIS EARTHQUAKE (DBE)	
Safe Shutdown Earthquake (SSE)	0.30 g (PGA)
Operating Basis Earthquake (OBE)	0.15 g (PGA)

II. Major Technical Characteristics of the Kuosheng NPP

Items	Units 1 & 2
THERMAL-HYDRAULIC DESIGN	
Rated Thermal Power, MWt	3,001
Installed Electrical Power Capacity, MWe	985
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: Reload Cycle	1.88 4.0
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
DESIGN BASIS EARTHQUAKE (DBE)	
Safe Shutdown Earthquake (SSE) Operating Basis Earthquake (OBE)	0.4 g (PGA) 0.2 g (PGA)

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
Installed Electrical Power Capacity, MWe	951
Reactor Coolant System:	
System Pressure, nominal design, psia	2,280
System Pressure, minimum steady state, psia	2,220
System Pressure, nominal operating, psia	2,250
Coolant Inlet Temperature, nominal, °F	554.2
Reactor Pressure Vessel Inlet Temperature, °F	554.2
Reactor Pressure Vessel Outlet Temperature, °F	621.4
Total Reactor Coolant Flow Rate, gpm	277,800
Steam Generator:	
Feedwater Temperature, °F	442.6
SG Steam Outlet Temperature, °F	537.2
Steam Pressure, psia	979
Total Steam Flow Rate, lb/hr	12.55×10^6
Heat Transfer: Maximum Thermal Output, kW/ft	13.4
Average Thermal Output, kW/ft	5.53
Maximum Heat Flux, Btu/hr-ft^2	505,089
Average Heat Flux, Btu/hr-ft^2	201,130
Minimum DNBR (for design):	
Typical Flow Channel Transients	$\geq$ 1.23
Thimble Flow Channel	$\geq$ 1.22
NUCLEAR DESIGN	
Feed Enrichment (First Core), w/o:	
Region 1	1.6
Region 2	2.4
Region 3	3.1
Reload	4.60 & 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
DESIGN BASIS EARTHQUAKE (DBE)	
Safe Shutdown Earthquake (SSE) Operating Basis Earthquake (OBE)	0.4 g (PGA) 0.2 g (PGA)

IV. Major Technical Characteristics of the Lungmen NPP

Items	Units 1 & 2
THERMAL-HYDRAULIC DESIGN	
Rated Thermal Power, MWt	3,926
Installed Electrical Power Capacity, MWe	1,350
Reactor Coolant System: System Pressure, nominal in steam dome, psia Core Coolant Flow Rate, lb/hr	 1,040 115.1×10^6
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 Average Thermal Output, kW/ft Maximum Heat Flux, Btu/hr-ft^2 Average Heat Flux, Btu/hr-ft^2	13.4 4.2 432,296 135,496
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 Fuel Rod Array Overall FA Length, in.	 872 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	 92 0.404 0.007 0.026 Zircaloy-2
Fuel Pellet: Material Density, % of theoretical Diameter, in.	 Uranium Dioxide (UO_2) 97 0.345

Length, in.	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
DESIGN BASIS EARTHQUAKE (DBE)	

Safe Shutdown Earthquake (SSE) Operating Basis Earthquake (OBE)	0.4 g (PGA) 0.2 g (PGA)

ANNEX 2 REGULATORY REQUIREMENTS/ORDERS IN THE AFTERMATH OF FUKUSHIMA DAIICHI NUCLEAR ACCIDENT

In general, due to the fact that all its reactors are designed and manufactured by the US, Taiwan's NPPs shall follow all applicable laws and regulations of the country of origin, i.e. the ones of the USNRC. In addition, in light of the lessons learned from Fukushima, AEC also required TPC to adopt good practices from Japan and Europe to further strengthen the robustness of its NPPs on the basis of the Nuclear Reactor Facilities Regulation Act.

Article 14 of this Act states that: "If there is anything that does not conform to the prescription or if public health/safety or environmental ecology may be endangered, the competent authorities shall order the licensee to improve the situation or take any other necessary measures within a prescribed time period. If the situation is serious, the licensee does not improve it or does not take necessary measures within the prescribed period, the competent authorities may order the licensee to cease the working on the scene, or operation thereof, or may revoke the license or permit the operation only under a limited power."

Based on the above-mentioned legal structure, AEC issued the following regulatory orders in three batches to TPC requiring implementation of enhancements on nuclear safety on 5/11/2012, 6/6/2013, and 6/3/2014, respectively:

A. Regulatory Orders Issued on November 5, 2012

Building on the results of the stress tests (STs) conducted in Taiwan and insights from the actions being taken by other regulators, the AEC issued following orders to TPC on November 5, 2012, while the TPC may propose alternatives subject to AEC approval:

<u>Orders Issued by AEC's Department of Nuclear Regulation (DNR):</u>

1. XX-JLD-10101*: Requiring seismic hazard re-evaluations implementing the recommendation from the USNRC NTTF Report Tier 1 recommendation 2.1 to conduct seismic and flood hazard re-evaluations.
2. XX-JLD-10102: Requiring flood hazard re-evaluations implementing the USNRC NTTF Report Tier 1 recommendation 2.1 to conduct seismic and flood hazard re-evaluations.
3. XX-JLD-10103: Requiring TPC to simulate the mechanism of seismic and tsunami hazards and the resulting risks based on comments from an AEC review meeting.
4. XX-JLD-10104: Requiring the enhancement of the water tightness of buildings (or build seawall, or tidal barrier) to a level of 6 meters above current licensing bases based on the actions being taken at Japanese NPPs and as referred to in the USNRC NTTF Report, to address the uncertainty from the original design basis tsunami height by adding 6 meters of protection.
5. XX-JLD-10105: Requiring seismic, flood and other external events walkdowns consistent with the USNRC NTTF Report Tier 1 recommendation 2.3 to conduct seismic and flood walkdowns.

6. XX-JLD-10106: Requiring TPC to take actions to address Station Blackout (SBO) consistent with the USNRC NTTF Report Tier 1 recommendation 4.1 on SBO regulatory actions.
7. XX-JLD-10107: Requiring at least 2 Emergency Diesel Generators (EDGs) to be in an operable state all the time even when the reactor is shut down so that if one unit is shut down with one EDG under maintenance and the swing EDG is assigned to it according to the new requirement, the capability of the swing EDG to back up the other unit is restricted.
8. XX-JLD-10108: Requiring TPC to enhance emergency DC power supply to secure the batteries storage capacity of at least 8 h without isolating the load and at least 24 h after the unnecessary loads are isolated.
9. XX-JLD-10109: Requiring TPC to extend the SBO coping time to at least 24 h based on specific issues for Taiwan's NPP in that the original requirements of USNRC Regulatory Guide (RG) 1.155 do not include the effects resulting from earthquake and tsunami.
10. XX-JLD-10110: Requiring TPC to install an extra seismic qualified gas-cooled EDG at high elevation for each NPP to address specific defense-in-depth issues with electrical power supplies for Taiwan. AEC accepts the alternatives for this order to provide the watertightness of the swing EDG building.
11. XX-JLD-10111: Requiring TPC to install an alternate UHS consistent with recommendations from the ENSREG action plan.
12. XX-JLD-10112: Requiring TPC to implement the actions of the USNRC's Post-9/11 action (B.5.b) (SBO and Advanced Accident Mitigation) to stage response equipment on or near site to respond to extreme external events (see USNRC 10 CFR 50.54(hh)(2)).
13. XX-JLD-10113: Requiring TPC to address the USNRC NTTF Report Tier 1 recommendation 4.2 on equipment covered under USNRC regulation 10 CFR 50.54(hh)(2).
14. XX-JLD-10114: Requiring TPC to install reliable hardened vents for Mark I and ABWR containments and request the installation of filtration for all different containment designs consistent with the recommendation of USNRC NTTF Report Tier 1 recommendation 5.1 on reliable hardened vents for BWR Mark I and Mark II containments.
15. XX-JLD-10115: Requiring TPC to install SFP instrumentation consistent with the recommendation of the USNRC NTTF Report Tier 1 recommendation 7.1 on SFP instrumentation.
16. XX-JLD-10116: Requiring TPC to strengthen and integrate the EOPs, SAMGs and EDMGs with the URGs developed by TPC following the Fukushima accident consistent with the USNRC NTTF Report Tier 1 recommendation 8 on strengthening and integration of EOPs, SAMGs, and EDMGs.
17. XX-JLD-10117: Requiring TPC to perform a volcanic PRA for its NPPs and to study the impacts from ash dispersion based on comments during a high-level review meeting.
18. XX-JLD-10118: Requiring TPC to enhance the water-tightness of the fire doors of essential electrical equipment rooms based on specific concerns with

the location of the equipment at Taiwan's NPPs and recommendations from the Japanese regulatory body for NPPs in Japan.

19. XX-JLD-10119: Requiring TPC to enhance the seismic resistance for the fire brigade buildings to cope with BDBE conditions to address specific issues for Taiwan's NPPs and a good practices from EU-PRs (EC/ENSREG Peer Reviews of Taiwan STs for NPPs).
20. XX-JLD-10120: Requiring TPC to improve the reliability of off-site power supplies to address specific issues for Taiwan's NPPs and recommendations from the Japanese regulatory body for NPPs in Japan.
21. XX-JLD-10121: Requiring TPC to improve the seismic resistance of raw water reservoirs at the NPPs and to consider the installation of impermeable liners to address specific issues for Taiwan's NPPs and consistent with the measures being taken by the Tokyo Electric Power Company (TEPCO) in Japan to install impermeable liners.
22. XX-JLD-10122: Requiring TPC to install the passive autocatalytic recombiners (PARs) to prevent hydrogen explosions consistent with recommendations in the ENSREG action plan.
23. CS-JLD-101101: Requiring TPC to conduct an enhancement evaluation of safety related SSCs for the Chinshan NPP followed by the upgrading of the licensing basis SSE from 0.3g to 0.4g for specific SSCs relied upon to respond to an accident. (An order from the Executive Yuan)
24. MS-JLD-101301: Requiring TPC to address the issue with the PWR reactor coolant pump (RCP) seal loss-of-coolant-accident leakage issue for Maanshan NPP consistent with the ENSREG action plan.

Orders Issued by AEC's Department of Nuclear Technology(DNT):

1. HQ-JLD-1013001: Requiring TPC to update "radiation protection measures and planning for the residents within emergency planning zone (EPZ) of nuclear power plant" in response to the expanded EPZ from 5 km to 8 km based on the lessons learned from Fukushima for all NPPs in Taiwan.
2. XX-JLD-1013002 and 1013004: Requiring TPC to address staffing and communications issues for emergency preparedness consistent with the USNRC NTTF Report Tier 1 recommendation 9.3 on emergency preparedness regulatory actions.
3. XX-JLD-10104(AECDNT): Requiring TPC to reinforce the structure of the existing nonseismically qualified Technical Support Centre (TSC) used for emergency response to address specific seismic concerns of the NPPs in Taiwan. (AECDNT: AEC's Department of Nuclear Technology.)
4. XX-JLD-1013003: Requesting TPC to consider building a seismically isolated TSC building based on the practice being implemented in Japan in light of the Fukushima accident and consistent with lessons learned provided by the IAEA.

Orders Issued by AEC's Fuel Cycle and Materials Administration (FCMA):

1. RL-JLD-1012042: Requiring TPC to procure 40 mobile detection equipment with automatic data transmission capability for four NPPs to enhance capability of radiation fallout monitoring in a timely manner.

2. RL-JLD-1012043: Requiring TPC to install 13 radiation monitoring stations within the EPZ of NPPs to set up a radiation monitoring preparedness platform and strengthen radiation monitoring capability.

3. RL-JLD-1012044: Requiring TPC to procure four radiation detection vehicles to enhance mobile radiation monitoring capability.

B. Regulatory Orders Issued on June 6, 2013

Based on the OECD/NEA peer review results of stress tests for the Taiwanese NPPs, the AEC's DNR issued the following regulatory orders on June 6, 2013:

Orders Issued by AEC's Department of Nuclear Regulation (DNR):

1. XX-JLD-10201: Requiring TPC to conduct fault displacement analysis for new evidences of Shanchiao and Hengchun Faults near the NPPs (within a radius of 8 km).
2. XX-JLD-10202: Requiring TPC to provide the interface between existing post-earthquake and post-tsunami operating procedures of NPPs.
3. XX-JLD-10203: Requiring TPC to systematically assess the combinations of events in the areas of flooding and extreme natural events at NPPs.
4. XX-JLD-10204: Requiring TPC to examine the probable maximum precipitation (PMP) with regional topographical maps of NPPs.
5. HQ-JLD-10201: Requiring TPC headquarters to deploy a local seismic network (one in the north and one in the south) to capture small earthquakes in order to understand whether or not the pattern of the epicenters indicate correlation with postulated tectonic features.

C. Regulatory Orders Issued on March 6, 2014

Based on the EC/ENSREG peer review results of stress tests for the Taiwanese NPPs, the following regulatory orders were issued on March 6, 2014 by the AEC's DNR:

Orders Issued by AEC's Department of Nuclear Regulation (DNR) :

1. XX-JLD-10301: Requiring TPC to perform a thorough geological and geomorphological assessment for plant site damage caused by dip slope sliding and landslide on a site-specific basis and to provide a continuous monitoring and early warning system for slopes susceptible to damage caused by dip slope sliding and landslide.
2. XX-JLD-10302: Requiring TPC to conduct post-seismic walkdown inspection on non-seismic category I structures, systems and components (SSCs).
3. XX-JLD-10303: Requiring TPC to consider the development of strategies to minimize the quantity of contaminated water produced during accident and to evaluate the installation of closed cooling water loops which may include the mobile heat exchangers and high pressure alternate injection equipments.
4. XX-JLD-10304**: Requiring TPC to strengthen the capability of a BWR in depressurizing the RPV by utilizing diversified measures.
5. XX-JLD-10305: Requiring TPC to improve the habitability in the main control

room (MCR) and local shutdown panel areas under accident conditions.

6. XX-JLD-10306: Requiring TPC to consider a systematic assessment of combinations of events including multi-unit and multi-site accidents (since some NPPs are located in relatively close vicinity).

7. XX-JLD-10307: Requiring TPC to improve the resistance of the plant site infrastructure against earthquakes and have heavy mechanical equipments ready for obviating the roadblocks.

---

* XX stands for “order applicable to all 4 NPPs of TPC”;

CS, KS, MS, or LM stands for “order applicable to Chinshan, Kuosheng, Maanshan, or Lungmen NPP, respectively”;

HQ stands for “order issued to TPC headquarters”;

RL stands for “order applicable to TPC’s Radiation Laboratory”; and

JLD stands for “Japan Lessons-learned project Directorate”.

** not applicable to MSNPP.