THE REPUBLIC OF CHINA

NATIONAL REPORT

FOR THE

CONVENTION ON NUCLEAR SAFETY

SEPTEMBER 2004

ATOMIC ENERGY COUNCIL, EXECUTIVE YUAN

TAIWAN, REPUBLIC OF CHINA

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INTRODUCTION

The indigenous energy resources are scarce in this country. The potential hydropower was approximately 5050 MW according to the survey result in 1996, and the amount of coal, oil, and natural gas produced domestically are very small. More than 97％of the energy supply needs to be imported from overseas. Therefore, diversification of energy sources is essential to maintain secured energy supply in this country.

In the aspect of nuclear power development, the first nuclear power installation in this country, Chinshan Unit 1, started its commercial operation in 1978. With the efforts made by all working members of domestic nuclear community later on, the total installed capacity of nuclear power is 5,144 MW from 3 nuclear power Stations, each with 2 units. By the end of December 2003, nuclear installations constituted 15.5％of the total installed power capacity, yet actually generated 21.5％of the total electric power in year 2003. Currently, the fourth Nuclear Power Station with two ABWR units is under construction. It is scheduled to be commercial in the year 2007. The total installed capacity of nuclear power will then become 7,844 MW, and the share to the total power capacity is predicted to be 18.5% by that time.

The Atomic Energy Council (AEC) is the governing authority for all atomic energy related affairs in the country. It was founded in 1955 as a ministerial level department under the highest administrative authority, the Executive Yuan. Before the establishment of nuclear power stations and diverse civilian applications, the principal missions of AEC in the early years were limited to the management of international affairs concerning atomic energy and the promotion of the peaceful applications of atomic energy in this country. For recent years, three nuclear power stations have been stably operated and applications of atomic energy in medicine, agriculture, industry, and research fields have been expanding in great pace. Therefore, the most important tasks of AEC have been shifted to safety regulation, radiation protection, radwaste administration, and R&D for nuclear technology.

There are three affiliated organizations under AEC. They are the Institute of Nuclear Energy Research (INER), the Fuel Cycle and Materials Administration (FCMA), and the Radiation Monitoring Center (RMC). INER is the only nuclear R&D institute in this country. The major R&D areas include nuclear engineering, nuclear safety, radiation protection and detection, radiological medicine, and environmental protection technology. FCMA has two major responsibilities: the safety regulation of the treatment, transportation and final disposal of radwaste; the safety regulation of the import, export, storage, and transfer of nuclear materials as well as nuclear fuels. The major responsibility of RMC is the monitoring of natural and man-made ionizing radiation in the environment.

The Taiwan Power Company (TPC), a state owned utility, generated all the electrical power needed in this country before June 1999. Because of the privatization trends, more and more private power companies connect to the electrical grid recently, the percentage of electrical power generated by TPC decreased to 78.3％(in which 3.9％by hydropower, 52.9％by fossil power, and 21.5％by nuclear power) by the end of 2003.

In the nuclear power sector, TPC currently has three power stations in operation and one station under construction. With the best efforts spent by the TPC staffs, the performance of the three operating nuclear power stations keeps on improving in recent years. In terms of the ten performance indicators issued by WANO in the year of 2002, only the collective radiation exposure and the chemistry index of BWR are slightly below the world average. Currently these items are also the major areas targeted by the three operating nuclear power stations to improve performance.

Generally speaking, the operations of nuclear power stations in this country are relatively satisfactory with respect to safety and reliability. To foster good safety culture and to ensure a high level of nuclear safety will continue to be the primary goal of both AEC and TPC. The review process of the Convention on Nuclear Safety is a good practice for us to examine our performance in nuclear power business and to share experiences with other countries. It is of great importance to the international community to ensure that the use of nuclear energy is safe, well regulated, and environmental sound, as that stated in the preamble of the Convention on Nuclear Safety. This statement is also our expectation in preparation of this national report.

ARTICLE 6. EXISTING NUCLEAR INSTALLATIONS

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

6.1 Existing Nuclear Installations

There are a total of 4 nuclear power stations in Taiwan, each with two reactor units (for detail, please refer to Annex 1). These include 3 power stations in operation and another one under construction. Among the 3 operating nuclear power stations, Maanshan has two 3-loop pressurized water reactors (PWRs) with nuclear steam supply systems (NSSS) supplied by Westinghouse (W), while for the other two plants, Chinshan and Kuosheng, each has two boiling water reactors (BWRs) with NSSS supplied by General Electric (GE). Among these 4 BWRs, two Chinshan units are BWR-4 with Mark I containments, while the other two Kuosheng units are BWR-6 with Mark III containments. The Lungmen power station is under construction and will have two advanced BWRs with their NSSS supplied by GE. In order to enhance the safety, reliability, operability, and maintainability of the two new power reactors, TPC has incorporated into their design requirements the operation experience feedback of its 6 commercial power reactors and the design modification with the successful experiences from international nuclear industries. Although these two new power reactors are of a standard design certified by the U.S.NRC, the licensing process of USNRC 10CFR52 is not applicable in Taiwan. The local regulations still follow a two-step licensing process similar to USNRC 10CFR50. The Construction Permit (CP) was granted for these two new power reactors in March 1999. Fig.6-1 shows the locations of these existing nuclear installations in Taiwan.

6.2 Significant Corrective Actions

<u>The Chinshan Torus Crack Event</u>

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

- Repair of the broken pipe and the cracked torus shell.
- Regular check of repaired parts.
- Revision of nitrogen charging procedures to prevent adverse effects caused by low temperature.
- Improvement of nitrogen heating system to enhance its temperature control function during nitrogen charging operation.
- Installation of temperature detectors on the important equipment near nitrogen charging pipes.

<u>The Chinshan Core Shroud Repair</u>

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

- The design requirements of core shroud as described in FSAR must be maintained.
- The designed lifetime of the repaired components must be no less than the remaining life of the Unit 1 reactor (including the consideration of plant life extension).
- The materials for repair must conform to the ASME code and the suggestions made by BWRVIP.
- The prevention of IGSCC and IASCC must be considered in the selection of repair materials.
- No welding is allowed in the repair processes.
- Loading and stress analyses must be performed for the repaired components, core shroud, and reactor vessel with all accidental conditions postulated in FSAR.
- After the repair, only 0.46% of core flow is allowed to leak from the core bypass to

the downcomer region under the worst accident condition.

TPC also made integrated safety assessment before the repair work and the major conclusions were as follows.

- The probability of occurrence and the severity of the consequences of all the accidents postulated in FSAR are not significantly changed due to the repair of core shroud.
- The repair will not give rise to the occurrence of possible accidents that are not included in FSAR.
- The safety margins reserved in the Technical Specifications are not affected due to the repair.

The repair work was contracted to GE and completed in 1995. The probabilistic risk assessment with the plant conditions after the repair showed that the increase of core damage frequency was 1.6E-7. It was about 0.1 % of the total core damage frequency of Unit 1. Therefore, the safety impact of the core shroud repair to the plant safety was considered to be negligible.

<u>The Chinshan Low Pressure Turbine Blade Crack Event</u>

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

Nevertheless, after 6024 hours of operation, cracks were again found on several new L-2 blades and the root cause was still unknown. Under this circumstance Chinshan plant, to ensure safety operation, could only shut the reactor down and change the turbine rotor, having blade creeks thought to be critical, every 6 to 8 months. This kind of practice had been performed until all the LP turbines were installed with new rotors supplied by a different vendor in 1999. These new rotors were forged section by section and then welded together. By this way of manufacturing, the problems of stress corrosion cracking associated with the shrunk-on type rotor were effectively solved. In addition, the ability to prevent the problems caused by torsional vibration and stall flutter was greatly improved in the design of the new rotors. Chinshan plant has been finally relieved from this operational burden associated with turbine blade failure since 1999.

<u>The Kuosheng Workers Over Exposure Accident</u>

At the night of March 20, 1993, several workers were transferring spent control rods from the reactor to the spent fuel pool of unit 2 of Kuosheng plant. Because of improper operation during the transfer process, 6 workers received abnormal radiation dosage. The highest dose taken by one of these workers was 299.9mSv. AEC sent two section chiefs, one from Nuclear Regulatory Department and another from Radiation Protection Department, to Kuosheng site on March 21st after TPC's notice. Several review meetings were held in the following days to find out the root causes and corrective actions for the accident. The major actions required by AEC included:

- The hanging equipment of control rods on the side of spent fuel pool is not allowed to use any more.
- The review process of radiation safety and working safety associated with special operation needs to be improved.
- For operations with high risk of radiation exposure, there must be health physics personnel at the location of operation to monitor the radiation level and oversee the working procedures.
- Workers have to carry alarm type dosimeter while working in high radiation areas.

The Kuosheng RPV Over-Pressure at Low-Temperature Event

On November 7, 1993, the outage maintenance of Kuosheng Unit 1 had been performed for 63 days. An RPV Leak Test was completed at 05:00 and the RPV pressure was about 72.1 kg/cm^2 at that time. Thirty minutes later, a group of maintenance workers started to perform Scram Timing Test for the control rods, with the RPV pressure kept at about 70.7 kg/cm^2. At 07:41, after the recirculation pump B was switched from low speed to high speed for vibration test, the RPV pressure started to increase. Consequently, a reactor high-pressure scram occurred at 07:45 with the RPV pressure at about 81.5 kg/cm^2. Recirculation pumps A and B also tripped a few seconds later due to ATWS protection signal. In order to mitigate this situation, a reactor operator manually stopped CRD pump and raised RWCU dump flow at 07:46. The RPV pressure dropped to 72 kg/cm^2 at 16 seconds later.

The major root cause of this event is considered to be the switching of recirculation pump from low speed to high speed when the RPV water is nearly in solid state. Fortunately, the maximum RPV pressure in the event was about 81.5 kg/cm^2, which is well below the safety limit (about 93.2 kg/cm^2). The temperature rising rate did not exceed the safety limit of 55℃/hr during the event either. However, it was found in the relevant operational manual that only minimum RPV pressure (21.1 kg/cm^2) and maximum RPV temperature (100℃) are required for the switching of re-circulation pump from low speed to high speed. Obviously the upper limit of RPV pressure should also be added to the requirement. Therefore, correction of the operational manual and relevant training of operators are the most important corrective actions for this event.

The Kuosheng MCR Control Panel Loss of Power Event

On March 16, 2000, the outage maintenance of Kuosheng Unit 1 had been performed for 13 days. At time 13:55, a "NSSS/BOP ANNUNCIATOR POWER SUPPLY FAILURE" alarm appeared and then all the alarms in the control room were out of work. The reactor operator informed the working crew to stop all the maintenance activities and declared it a second category nuclear accident (ref. Article 16 Emergency Preparedness). All members of Technical Support Center (TSC) reported at 14:10 to the superintendent according to the emergency procedure.

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

- To install protective cover on DC Breaker 1DE01B to prevent it from inadvertent touch.
- To install redundant power supply for the NSSS and BOP annunciators in the control room.

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

The Maanshan Turbine Building Fire Event

At a few minutes past 05:00 in the afternoon of July 7, 1985, the Maanshan Unit 1 operated with 97% power and the electrical output was about 885 MWe. The reactor was under automatic control and the operation looked smooth. However, a historical accident happened in the turbine building of this unit a few minutes later. The major scenarios of this accident are described as follows.

- At 17:21:00, the operators in the main control room felt high-frequency and low-amplitude quakes. Then it was found that both the reactor and the turbine were automatically tripped.
- At 17:21:40, 5 out of the 10 turbine vibration monitors reached full-scale indication, while the other 5 dropped to zero. This phenomenon implies that the turbine experienced very strong vibrations.
- At 17:21:42, the record of process computer showed that the turbine tripped earlier than the reactor.
- At 17:21:45, it was found that the hydrogen pressure inside the generator quickly dropped, implying that a large amount of hydrogen leaked out of the generator.
- During the first 40 to 60 seconds of the event, the operators in the main control room heard a huge “ban” came out from the turbine building.
- At 17:22:00, 5 to 6 local operators entered the turbine building and found heavy smoke coming out of the generator. They started to put out the fire immediately.
- At 17:27:30, the automatic fire extinguishers around the turbine and generator started to spray water, but the fire kept on.
- At 17:40:00, the plant fire department joined the fire extinguishing action.
- At 17:50:00, the fire department from the county joined the fire extinguishing action. The fire was put out at 18:50 and the event was finally terminated at the same time.

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

- To remove the blades at the last stage of low-pressure turbines of Unit 2 (the unit without event), it hence changes the natural frequency of torsional vibration to 119.05 Hz. This condition would be effective only for one operational cycle and the load of the unit was limited to 500 MWe to ensure safety of operation.

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

- The fire lasted for one hour and thirty minutes during the event. The reasons are (1) turning off the lubrication oil pump for the low-pressure turbine was not early enough, and (2) the firemen tried to save the generator so that water was not used to extinguish fire at the beginning. Hardware and software modifications to the fire fighting systems have been made to prevent the previous mistakes from happening again.

<u>The Maanshan Control Rods Crack Event</u>

On September 24, 1988, after a reactor trip at Maanshan Unit-1, the digital rod position indicator (DRPI) showed that the rod cluster control assembly (RCCA) R41 was stopped at step 12, which represented a distance of 7.5” away from the fully inserted position. (Rod positions are identified in terms of steps, step 0 for fully insertion and step 228 for fully withdrawn). In addition, two subsequent rod drop tests performed on R41 showed it stopped at steps 12 and 18 respectively.

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

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

<u>The Maanshan Station Blackout Incident</u>

On March 17, 2001, Units 1 and 2 of the Maanshan Nuclear Power Station were tripped at 03:21 and 03:23, respectively. This event was caused by the instability of the offsite super-high voltage (345 KV) power transmission line, which in turn was caused by seasonal sea smog containing salt deposit. While the reactor was maintained in hot standby condition after reactor scram, the 345 KV offsite power supply system was still unstable. At 00:41 of the next day, the power supply of essential bus A of Unit 1 was automatically transferred from the 345 KV to the 161 KV offsite power supply because of the loss of the 345 KV offsite power. A few minutes later, at 00:46, the 161 KV offsite power was lost too. Right after that, the emergency diesel generator (EDG) A was successfully started, but cannot supply power to the essential bus A due to a bus grounding fault. In the meantime, the EDG B could not generate power because of losing excitation either. This situation of losing all AC power (i.e., station blackout) in Unit 1 lasted for almost two hours until the swing EDG (i.e. the fifth EDG in the Maanshan Nuclear Power Station) successfully started at 02:50 and supplied power to the essential bus B at 02:54 of that day. This incident, a site emergency incident without any release of radioactive material, was considered to be a Station Black incident.

After the incident, the two units of the Maanshan were required to remain shutdown for corrective actions. For unit 1, the operators were asked to find out the root causes and corresponding counter measures of the incident, and to make sure that no major equipment was damaged. For unit 2, it was also required to make sure that no potential risk may cause an incident similar to that happened in Unit 1. The two units were allowed to re-start again, after AEC confirmed that the above requirements were fulfilled.

<u>The Lan-yu LLW Storage Facility Repacking Project</u>

The waste drums that were shipped to the Lan-yu storage facility in early years have been emplaced up to 20 years. Some of the waste containers and their waste-form are believed to have degraded due to the nature of the storage conditions and the methods used to cement the waste. The Lan-yu Island was originally one of the candidate sites for the final disposal of LLW. Due to the strong protest of the local native inhabitants on May 4, 2003, the government has announced to remove the Lan-yu Island from the list of final disposal site candidates and the waste will be relocated to the final disposal facility when it is available.

To maintain the safe storage of the LLW drums shipped to the Lan-yu facility and to make preparation for the waste relocation to the final disposal site, TPC has launched a repacking project in February 2004. The treatment process for the three categories of waste drums is as follows:

- Corroded drums: removing rust and repainting the containers.
- Damaged container with intact waste-form placing 12 drums into a 3m×2m×1m galvanized carbon steel container, which will then be grouted into position with cement.
- Drums with severely degraded waste-form: crush the drums into powder and mixed with cement-based agent, then pour the resulting product into a new 200 liter galvanized carbon steel drum.

The total expense of the repacking system is about 10 million US dollars including an over-trench mobile retrieval facility, a treatment plant and a buffer storage warehouse (still under construction). The project was scheduled to be completed within six years.

6.3 Major Safety Assessments (Reviews)

During the planning stage of a project to install a new nuclear power station, the plant holder has to submit a feasibility report and an environmental impact assessment report of the project for AEC and EPA (Environmental Protection Agency) to review. After all the nuclear safety concerns to the public are clarified and these two reports were approved, the holder is allowed to start bidding processes for the plant. AEC will then review the preliminary safety analysis report (PSAR) of the plant submitted by the plant holder. Usually the PSAR review process takes about 12 months to complete after its submission to AEC.

After the approval of PSAR, a construction permit will be issued and the construction of the plant is allowed to start. The construction phase of a new nuclear power station is generally subdivided into five sub-stages, namely design, manufacturing, installation, pre-operational test, and startup test. During these sub-stages in the construction phase, the major mandate of the regulatory body is to ensure that safety and construction quality in the PSAR are maintained. AEC's regulation further requires the plant holder to complete a final safety analysis report (FSAR) for review before its first fuel loading. The whole review process takes about 12 months. An operating license valid for forty years will be issued after AEC's approval. The Regulation also requests the holder to submit a comprehensive safety assessment report for approval six months before every ten year operation of a plant. Contents of the report should include (1) Review and assessment of the plant operation history related to operational safety, radiation safety and radioactive waste management; (2) Review of the committed betterment or enforcement items; (3) Specific follow-up issues, betterment promises and their schedules summarized from the above item (1) and (2) for the next ten year operating cycle. (4) Items requested by the AEC.

On the other hand, the inspection of operating nuclear installations is one of the most important tasks of AEC. For daily operations, resident inspectors of the plant perform daily monitoring and regulation on site. Occasionally, spot inspection is performed without pre-notice to enhance the alertness of the plant operators. At the end of each operation cycle, a nuclear power unit has to be shutdown for fuel reloading, inspection, maintenance, and modification of structures, systems and components to assure stable operation in the next cycle. In general, the quality of these works affects the safety and reliability of operation. Therefore, it is essential for AEC to examine the quality of these activities performed. AEC has established stringent requirements to audit the implementation quality of these outage activities performed by utility in order to assure the unit operational safety and stability.

All the detailed requirements of safety assessments through the life of nuclear power stations are described in Article 14. In addition, the operation of the nuclear power stations in Taiwan has been reviewed by a number of expert groups. For example, the Institute of Nuclear Power Operations (INPO), based in Atlanta, GA, USA, has organized several intensive and in-depth reviews of the operation of nuclear power stations in Taiwan. The World Association of Nuclear Operators (WANO) also sent a team from Japan for a brief visit in 1991. The reports of these visits are very valuable but they are not available to the general public. Nevertheless, none of the previous groups uncovered any problems that were deemed serious enough to warrant shut down of any of the reactors, even temporarily, in Taiwan.

In the year 1991, an "International Expert Consulting Team" was appointed by the Minister of Economic Affairs to evaluate the performance of the three nuclear power stations operated by TPC. The team member consists of:

- Richard Wilson, Team Leader

  Mallinckrodt Professor of Physics, Harvard University, USA

- Kenzo Amoh

  Toshiba Corporation, Japan

- Jen Chang Chou

  Institute of Nuclear Energy Research, ROC

- Paul Dozinel

  Electrabel, Belgium

- Henri Guimbail

Electricite de France, France

- Tohru Hasegawa

  Hitachi Engineering and Service Ltd, Japan

The team visited each nuclear power stations and received briefings from TPC Headquarters, AEC, and the plant visited. A few local residents were also interviewed by the team. In the end the team compared the performance of TPC's nuclear power stations with that of the world plants by a number of Performance Indicators. It was concluded that the operation in Taiwan was comparable to other nuclear power stations and that the performance was, in most respects, improving. The major recommendations of this team are listed as follows.

- The first finding of the team is the lake of communication capability in TPC, both within TPC itself and between TPC and the Government, or toward the public. This shows up in various ways and is a problem in all aspects of the nuclear power operation. Coupled with a failure to understand and explain the meaning of measurements, it is a major source of misunderstandings to the public.

- The team has noticed a lack of flexibility in all levels leading to a long time taken to implement recommendations. This seems to be resulted from a decision making structure which is too centralized, both in the Head Office and in the Superintendent, to delegate adequately to their staff.

- The team found that an adequate Safety Culture had not penetrated to all levels of the power station staff. This is a serious problem to be addressed. It may be the result of the previous two issues. How can one gain the commitment of all staff if justifications are not clear enough or it takes too long to implement the improvements?

- The team has also noticed and agreed with AEC's desire to be independent. This is extremely important in developing a normal regulatory process. However, the team considered that AEC still has some way to go in achieving the independence in data understanding and issue identification.

- The team concluded that TPC satisfactorily operates its' nuclear power stations and is moving in the direction of improvement. The team believes that if the issues of communication, flexibility of thoughts and attention to safety culture are all properly addressed, TPC has the potential to join the world group of excellent operators soon.

All the comments made by the pervious review groups have been taken seriously by AEC and TPC. Action plans corresponding to these comments were set up and followed until all relevant issues are corrected and satisfied. For example, TPC has gleaned from the INPO reports over 100 recognizably items for each reactor, and set about addressing them all. With these efforts, it is fair to say that the operations of the nuclear power stations in this country are relatively satisfactory with respect to safety and reliability.

6.4 Programs and Measures for Safety Upgrading

(1) Update of FSAR

The regulations associated with FSAR updating in Taiwan are similar to that of the 10CFR50.71 in the United States. The update of FSAR for a unit is usually done within 6 months after the refueling outage. It may consist of Design Change Requests, FSAR change packages and Non-conforming Condition Dispositions of the plant. In addition to the hard copies of the FSAR (which are the legal licensing bases), computer files (which only include written descriptions and tables but not the figures) are also saved for reference.

(2) Update of Technical Specifications

The Technical Specifications (TS) of nuclear power stations are checked and revised every 12 to 18 months to reflect the up-to-date condition of the plants. Any TS change shall be approved by AEC. When the requested changes are granted, the affected pages will be revised and distributed to each holder of the controlled copy of TS. Moreover, the Chinshan power station has implemented Improved Technical Specifications (ITS) on Feb. 26, 2002. Chinshan is the first twin-unit nuclear power station in Taiwan and has been operated with customer TS for more than 24 years. When it adopted the ITS, its unit 1 was in normal operation while the unit 2 was shut down for refueling outage.

In the 1990's, TPC noticed the development of the ITS in USA and started a project to convert Chinshan's customer TS to ITS. Both TPC and AEC spent a lot of efforts on this project. In converting the TS, dozens of programs and hundreds of procedures were reviewed and revised. The entire operating crew of Chinshan plant was trained several times.

Although the converting process was very energy consuming, the outcome of the implementation of ITS is very fruitful. For example, the limiting conditions for operation are more safety oriented, the allowed outage times and the surveillance requirements are optimized, and the strong bases support all the requirements. From the experiences of Chinshan, both old standard TSs for Maanshan and Kuosheng are going to be converted to ITS in the near future. TPC expects the converting process could be shortened to within 2 years. In addition, TPC will still keep track of the TS development in the nuclear industry worldwide.

(3) International Cooperation of Seismic Study

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

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

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

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

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

(4) Probabilistic Risk Assessment

The development and application of Probabilistic Risk Assessment (PRA) technology can be divided into three phases in this country. In the first phase, beginning in 1982, AEC initiated a PRA program for domestic nuclear power stations. Comprehensive PRA models were completed for Kuosheng, Maanshan and Chinshan plant in the year 1985, 1987 and 1991, respectively. Possible core melt scenarios and their frequencies induced by internal events as well as external events were investigated, including earthquakes, typhoons, fires and internal floods.

Later on, the PRA models completed in the first phase could not account the updated plant status with successive design changes. Besides, these models were installed on the mainframe computer with implicit complexity, they were very 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 TPC in the second phase (from 1994 to 1997). The major tasks of this project include PRA model update, model conversion into PC-based tools, and the risk analysis for plant outage. At the end of this phase, 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 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 able to drive down O&M costs without impairing safety. This approach is 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)” was sponsored by TPC to develop an integral risk management system based on the plant specific living PRA models. This risk management system combines 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 obtain precise plant configuration for making decisions easily.

(5) Re-qualification of Licensed Operators

The license of a nuclear reactor operator is only effective for two years. The licensed operator taking part in the plant operation should take retraining courses required by relevant regulations. Thirty days before license expiration, the operator needs to prepare such documents as the physical examination report for the latest 6 months, employer's recommendation letter, and a retraining certificate to apply for a license renewal.

AEC may perform the selective operator re-qualification. The selective process includes both for the overall plant and for the operator. The selection criteria for the plant itself include its operational performance, the operator retraining program, regulation-violating events due to operational faults, and the efforts of promoting operator training. On the other hand, the operator may be selected for significant operational faults in the past two years, poor performance evaluated by the plant, or suspicious records of retraining. This operator may continue to operate reactor only if he passes the re-qualification.

6.5 Position as to Further Operation of Nuclear Installations

Based on the results of the continuous monitoring of the plant performance and evaluation of the plant safety, the government of this country concludes that operation of the existing nuclear installations is appropriate.

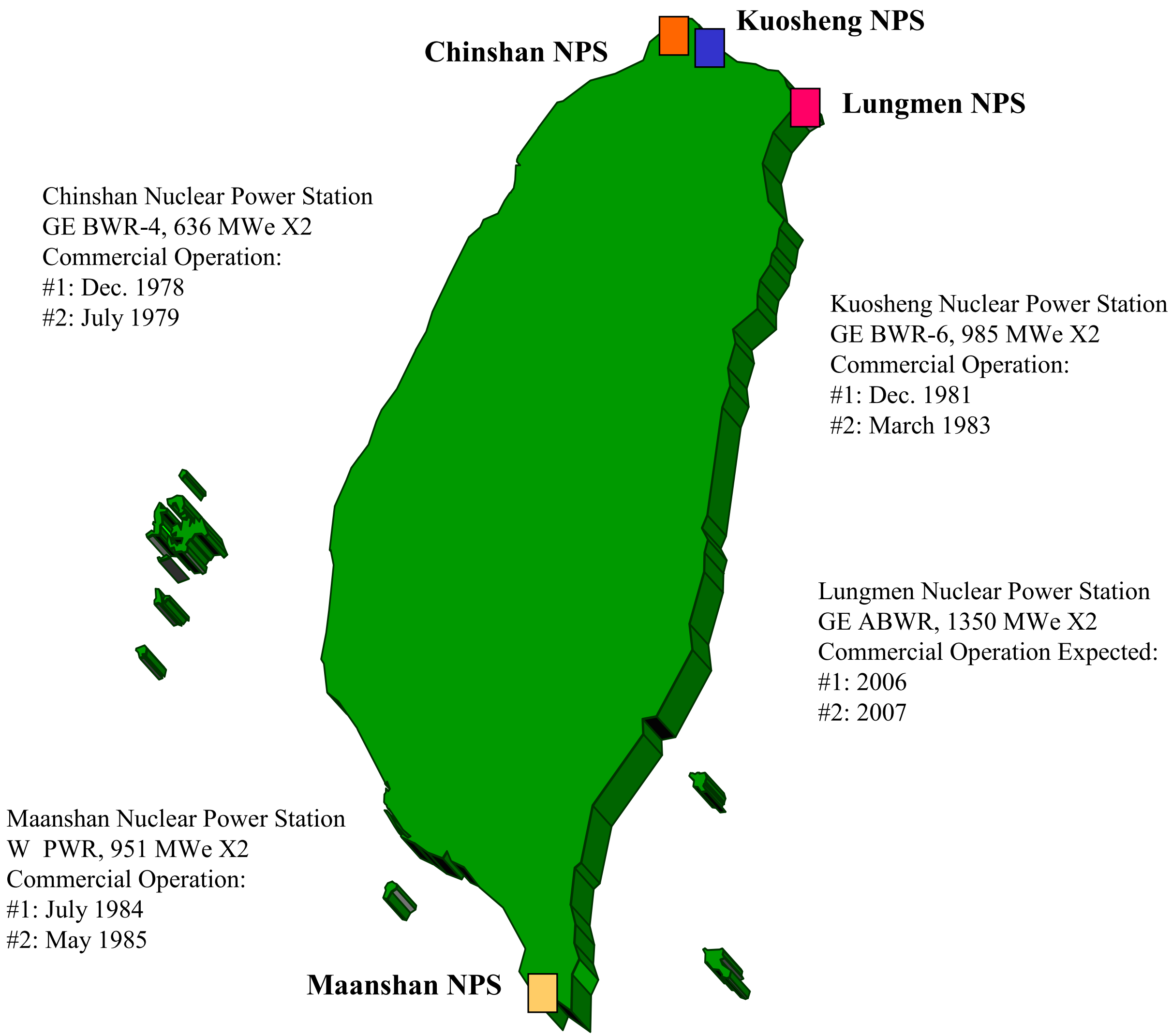


Figure 6.1 Locations of Nuclear Power Stations in Taiwan

ARTICLE 7. LEGISLATIVE AND REGULATORY FRAMEWORK

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

2. The legislative and regulatory framework shall provide for:

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

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

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

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

7.1 Legislative and Regulatory Framework

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

To assure the principle of “administration by Act”, the Act of 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 which was set to be 2 years. In response to the promulgation of this Act, major modifications of the Atomic Energy Act as well as other regulations and guidelines have been proposed.

In the following sections, current Acts, regulations and requirements will be described. Selected content of the proposed Acts will be provided as supplemental information.

7.2 Acts, Regulations and Requirements

This section describes six basic Acts, Enforcement Rules associated with the basic Acts, and the regulations.

7.2.1 Basic Acts

The six basic Acts, namely, Atomic Energy Act, Nuclear Reactor Facilities Regulation Act, Nuclear Damage Compensation Act, Ionizing Radiation Protection Act, Nuclear Materials and Radioactive Waste Management Act and the Nuclear Emergency Response Act were all passed by Legislative Yuan and signed by the President.

Atomic Energy Act

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

1. General Principles
2. Responsible Agency for Atomic Energy
3. Research and Development of Atomic Energy
   Science and Technology
4. Development and Utilization of Atomic Energy Resources
5. Regulatory Control of Nuclear Materials, Fuels, and Reactors
6. Radiation Protection
7. Encouragement, Patent and Compensation
8. Penal Provisions
9. Supplementary Provisions

Nuclear Reactor Facilities Regulation Act

This Act, promulgated in 2003, is to regulate nuclear facilities in order to protect the public heath and safety. It is composed of 44 articles, which are grouped into 5 chapters as in the following:

1. General Principles
2. Regulations of Construction and Operation
3. Regulations of Off- Commissioning and Decommissioning
4. Penal Provisions
5. Supplementary Provisions

Nuclear Damage Compensation Act

The compensation for the damages resulting from the peaceful uses of atomic energy is governed by the Nuclear Damage Compensation Act. This Act was promulgated in 1971, amended in 1977 and became effective in 1998. It is composed of 37 articles that are grouped into 5 chapters as in the following:

1. General Provisions
2. Liabilities for Damage Compensation
3. Maximum Amount and Guarantee for Liabilities
4. Right to Claim for Damage Compensation
5. Supplementary Provisions

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

Ionizing Radiation Protection Act

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

1. General Principles
2. Radiation Safety and Protection
3. Management of Material, Equipment or Practice
4. Penal Provisions
5. Supplementary Provisions

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

Nuclear Materials and Radioactive Waste Management Act

The Act, promulgated in 2002, is enacted to administrate radioactive material, to prevent radioactive hazard and to protect public health and safety. The Act is composed of 51 articles that are grouped into 5 chapters as in the following:

1. General Principles
2. Administration of Nuclear Materials and Nuclear Fuel
3. Administration of Radioactive Wastes
4. Penal Provisions

5. Supplementary Provisions

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 of the lives, bodies, and properties of our citizens. This Act is composed of 45 articles that are grouped into 7 chapters as in the following:

1. General Principles
2. Organizations and Responsibilities
3. Preparedness Measures
4. Response Measures
5. Recovery Measures
6. Penal Provisions
7. Supplementary Provisions

7.2.2 Enforcement Rules

The six basic Acts mentioned above are acts with general and fundamental principles and concepts. Necessary enforcement rules have been provided for four of them to address the details. The status of the enforcement rules is shown below:

1. Enforcement Rules for the Implementation of "Nuclear Reactor Facilities Regulation Act"

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

2. Enforcement Rules for the "Nuclear Damage Compensation Act"

   Under the requirement of Article 36 of the Nuclear Damage Compensation Act, Enforcement Rules for the "Nuclear Damage Compensation Act" was issued by the AEC in 1998.

3. Enforcement Rules for the "Ionizing Radiation Protection Act"

   Under the requirement of Article 56 of the Ionizing Radiation Protection Act, Enforcement Rules for the Ionizing Radiation Protection Act has been developed and was issued on December 25, 2002.

4. Enforcement Rules for the "Nuclear Materials and Radioactive Waste Management Act"

   Under Article 50 of the Nuclear Materials and Radioactive Waste Management Act, Enforcement Rules for the Nuclear Materials and Radioactive Waste Management Act was issued by the AEC in July 2003.

7.2.3 Regulations

In addition to the basic Acts and the enforcement rules described above, various regulations have been issued by the AEC. The Administrative Regulations, technical standards, and working notices in the areas of nuclear regulation, radiation protection and radioactive material are described below:

1. Regulations in nuclear regulation

   The Regulations applicable to the nuclear power stations are presented here.

   - Administrative Regulations for the restart after shut down of a nuclear reactor facility
   - Administrative Regulations for the prompt reporting and reporting of abnormal events for a nuclear reactor facility
   - Administrative Regulations for the authorized nuclear inspection and agency certification of a nuclear power station
   - Regulation for the dedication of commercial grade items and certification of dedication agency
   - Quality assurance criteria for a nuclear reactor facility
   - General design criteria for a nuclear reactor facility
   - Administrative Regulations for the physical examination of a licensed operator of nuclear reactor facility
   - Administrative Regulations for the suspension of operation license of a nuclear reactor facility
   - Administrative Regulations for the application of operation license for a nuclear reactor facility
   - Administrative Regulations for the application of construction permit for a nuclear reactor facility
   - Administrative Regulations for the application of the renewal of operation license for a nuclear reactor facility
   - Administrative Regulations for the decommission of a nuclear reactor facility
   - Administrative Regulations for the entrusted inspection of a nuclear reactor facility

2. Regulations in radiation protection

There are 18 related regulations for the Ionizing Radiation Protection Act as shown in

Table 7.1.

3. Regulation in radioactive material

   The regulations applicable to the radioactive material are further divided into 6 areas:

   1. General Principles
   2. handling
   3. storage
   4. transport
   5. treatment
   6. nuclear material and nuclear fuel

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 the 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 operation period of a nuclear reactor facility, 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, or to revoke the operation license, or to ask the licensee to operate the reactor at reduced power. To inflict the above mentioned penalty, a written statement describing the decision should be delivered to the licensee. In case of emergency, license suspension or revoking can be inflicted 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. Classification of Penal Provisions was defined there and the fines for civil penalties were raised significantly. In most situations, civil penalties and demands for timely improvements will be imposed upon first. Depending on licensee's willingness for improvement, additional penalties will be further imposed upon if the licensee didn't meet the requirements.

Table 7.1 Regulations related to the Ionizing Radiation Protection Act

No.	Names of Related Regulations
1.	Enforcement Rules of the Ionizing Radiation Protection Act (promulgated on 2000.12.25, implemented since 2003.2.1)
2.	Safety Standards for Protection against Ionizing Radiation (promulgated on 2003.1.30, implemented since 2003.2.1)
3.	Regulations for the Safe Transport of Radioactive material (promulgated on 2003.1.8, implemented since 2003.2.1)
4.	Standards for the Establishment of Radiation Protection Management Organizations and Radiation Protection Personnel (promulgated on 2002.12.11, implemented since 2003.2.1)
5.	Accreditation and Administrative Regulations for Personal Radiation Dose Evaluation Agencies (promulgated on 2002.12.11, implemented since 2003.2.1)
6.	Administrative Regulations for Radiation Protection Personnel (promulgated on 2002.12.11, implemented since 2003.2.1)
7.	Standards for Radiation-Induced Serious Environmental Contamination (promulgated on 2003.1.30, implemented since 2003.2.1)
8.	Standards for Limiting Radioactivity in Commodities (promulgated on 2002.12.4, implemented since 2003.2.1)
9.	Administrative Regulations for Radioactive Material and Equipment Capable of Producing Ionizing Radiation and Associated Practice (promulgated on 2003.1.22, implemented since 2003.2.1)
10.	Regulations for Administration of Radiation Protection Service Related Business (promulgated on 2002.12.25, implemented since 2003.2.1)
11.	Criteria for Management of Radiation Workplaces and Radiation Monitoring of their Environment (promulgated on 2002.12.25, implemented since 2003.2.1)
12.	Administrative Regulations for Operators of Radioactive Material or Equipment Capable of Producing Ionizing Radiation (promulgated on 2002.12.25, implemented since 2003.2.1)
13.	Administrative Regulations for the Operators of Production Facilities of Radioactive Material (promulgated on 2003.1.22, implemented since 2003.2.1)
14.	Classification of High Level Radiation Facilities and Administrative Regulations for Their Operators (promulgated on 2003.1.22, implemented since 2003.2.1)
15.	Radiation Protection and Control Regulations for Military Institutions (promulgated and implemented since 2003.2.26)
16.	Standards for Exemption of Radiation Sources from Regulation (promulgated on 2003.1.29, implemented since 2003.2.1)
17.	Standards for Collection of Regulation Fees for Ionizing Radiation Protection (promulgated on 2003.1.15, implemented since 2003.2.1)
18.	Regulations for Prevention and Management of Incident of Radiation Contaminated Buildings (promulgated and implemented on 2003.3.26)

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 The Regulatory Body

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

8.1.1 Mandate

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

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

The AEC, in the implementation of the aforementioned regulatory tasks and R&D works, adheres to the following principles: "Strict regulation, safety first, stringent control, and transparent information. 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 nuclear energy and materials, including radioactive material, are conducted with proper regards for public health and safety, and to protect the environment from the radiation out of nuclear reactors, materials, and waste facilities. The basic charter for these regulatory responsibilities is the Atomic Energy Act of 1968, through which Legislative Yuan （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 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 accelerator-produced radioactive material, in addition to uranium and thorium. The AEC also has been given the authority to regulate machine-produced radiation, such as the emissions from X-ray units or linear accelerators.

The AEC’s licensing authority also extends to the military sector, which uses radioisotopes or machine-produced radiation in the respective hospitals, academies and research laboratories, and radiopharmaceuticals in the hospitals. The AEC’s responsibilities include both safety and safeguards through which the agency ensures the security of machines and materials against radiological sabotage, lost and thefts.

8.1.3 Structure of the Regulatory Body

This section explains the structure of the AEC. It covers the Council, various offices and their responsibilities, and advisory committees and their functions.

8.1.3.1 The Council

The AEC consists of more than 10 commissioners, mostly representatives of relevant ministries or agencies within the Executive Yuan and experts from academia. The Chairman presides over the Council with the assistance of two Vice Chairmen and Secretary General to oversee the Council affairs. Directly under their supervision are five departments, three offices, and three affiliated agencies. The Council also has five advisory committees on nuclear policy and safety.

The Chairman, with the assistance from Vice Chairmen and Secretary General, Oversees Council affairs and supervises the affiliated agencies.

The five departments and three offices, working directly under the Council's administration, include the technical units such as Department of Planning, Department of Nuclear Regulation, Department of Radiation Protection and Department of Nuclear Technology; and also the administrative units such as Department of General Administration, Office of Personnel, Office of Accounting and Office of Security. The Office of Congressional Liaison is another mission oriented unit separated from these departments.

The three affiliated agencies are the Institute of Nuclear Energy Research, the Fuel Cycle and Materials Administration and the Radiation Monitoring Center.

The five advisory committees are the Advisory Committee on Nuclear Safety, the Advisory Committee on Ionizing Radiation Safety, the Advisory Committee on Environmental Protection for the Fourth Nuclear Power Station, the Advisory Committee on Nuclear Accident Investigation and Evaluation and the Advisory Committee on Nuclear Legislation.

The AEC employs approximately 180 personnel with FY2003 budget of NT$367 millions (not including the budget of the three affiliated agencies) as shown in Tables 8.1.

8.1.3.2 Offices of the Council

The responsibilities of the five Departments, one mission oriented unit, three offices, and three affiliated agencies are shown below.

Department of Planning

The primary responsibilities for the Planning Department include control and oversight for major policy implementation, planning, integration and assessments of 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; 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,
- Other planning assignments.

Department of Nuclear Regulation

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

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

- Issuance of licenses to nuclear reactor operators,
- Review of nuclear fuel reload safety analysis,
- Review, regulation and inspection of nuclear reactor decommissioning,
- Issuance of nuclear fuel licenses,
- Review, regulation and inspection of the design, construction, transfer, dismantling and disposal of nuclear fuel production facilities,
- Regulation of nuclear fuel usage,
- Other regulatory tasks related to nuclear energy.

Department of Radiation Protection

The primary responsibility for the Department of Radiation Protection is to ensure radiation safeties of nuclear facilities, environment, medical and non-medical applications of radioactive material 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 material and equipment capable of producing ionizing radiation and operating personnel,
- Regulation of radiation safety for radioactive material and equipment capable of producing ionizing radiation,
- Inspection of ionizing radiation site and its environmental radiation,
- Regulation of the safe transport of radioactive material,
- Development of Regulations for radiation safety,
- Issuance of radiation 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,
- Other assigned responsibilities.

Department of Nuclear Technology

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

- Investigation and evaluation of abnormal reactor events
- Analysis and evaluation of nuclear power station operation
- 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 Monitoring Center
- Management and security of nuclear information
- Other assigned responsibilities

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 AEC's administrative works. This office monitors legislative proposals, bills, and hearings, and informs the AEC of the views of Parliament on 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,

- To answer the related matters requested by the congressional members.

Administrative Units

The four administrative units are: Department of General Administration, Office of Personnel, Office of Accounting and Office of Security. The Department of General Administration is responsible for documentation and property management, and administrative support to all departments and offices. The Office of Personnel, Office of Accounting and Office of Security are responsible for the general administrative matters related.

Institute of Nuclear Energy Research（INER）

The Institute of Nuclear Energy Research was established in 1968 under the Atomic Energy Council. INER is now under transformation in order to fulfill to the long-term development goal as well as to comply with the government re-structures policy. This transformation allows INER utilizing broadly our nuclear energy technology to the environmental protection and civilian applications. Hence, INER has established three research centers and a marketing center. The three research centers are: Nuclear Safety Technology Center (NSTC), Environmental and Energy Technology Center (EETC), and Radiation Application Technology Center (RATC). The centers operate with 11 existing functional divisions in a matrix manner. The Marketing Center will emphasize on technology transfer and business promotion.

As a national laboratory, INER's missions are:
- Establish advanced R&D capabilities.
- Utilize our technologies to Taiwan's local industry.
- Benefit to the human living.

INER employs approximately 1,000 personnel including researchers, technicians, and supporting staff. The researchers are all professionals with graduate degrees: 150 with PhD and over 200 with Master degree. The FY2003 budget of INER is NT$2,291 millions.

Fuel Cycle and Materials Administration（FCMA）

The Fuel Cycle and Materials Administration, an Subsidiary agency under 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 radioactive waste storage facility located in Lan-Yu, an offshore island of Taiwan. That facility was designed to receive all the solidified low-level radioactive wastes generated in the country, especially that from Taiwan Power Company (TPC). However, it was transferred to TPC in July 1990.

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

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

1. Licensing and certification of facilities associated with the design, construction, operation and decommissioning/closure of installations for radwaste treatment, storage, and disposal.
2. Regulation and inspection for the treatment, storage, transport, disposal, import and export of radwaste.
3. Regulation and inspection for the import, export, storage, utilization, discard, and transfer of nuclear source materials.
4. Regulation and inspection for the import, export, storage, discard, and transfer of nuclear fuels.
5. Development of regulations and technical standards for the radioactive material.
6. International cooperation with respect to radioactive material regulation.
7. Education and public communication with respect to radioactive material regulation.
8. Policy and strategy development for the management of radioactive material.
9. Promotion of the research and development on radwaste management technologies.
10. Other matters related to radioactive material management.

FCMA employs about 42 personnel with FY 2003 budget of NT$74 Million.

Radiation Monitoring Center（RMC）

The Taiwan Radiation Monitoring Station was established in 1974 as an affiliated agency under 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. It has been given a new name, Radiation Monitoring Center, since July 1996.

The major tasks of the Center are:

- Formulation and implementation of environmental radiation measurement plans,
- Measurement of natural ionizing radiation in the environment,
- Measurement of radioactive fallout,
- Measurement of ionizing radiation in the vicinity of nuclear and other facilities with radioactivity,
- Measurement of 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 radiation measurement technology,
- Providing information and advice to the public on environmental radioactivity,
- Other matters related to environmental radiation monitoring.

The RMC employs approximately 30 personnel with FY2003 budget of NT$64 millions.

8.1.3.3 Advisory Committees

This section explains the structures and functions of the five advisory committees within AEC.

Advisory Committee on Nuclear Safety

The Committee consists of more than 10 members with expertise in science and engineering. It advises on the potential hazards of proposed or existing nuclear reactor facilities, the adequacy of proposed safety standards, and other matters the Council request. The statute requires that the Committee reviews certain types of applications, such as the construction permits or operating licenses for nuclear power reactors or research reactors.

Advisory Committee on Ionizing Radiation Safety

The Committee consists of more than 10 members with expertise in science and engineering, including physicians, scientists and other representatives from the medical community. This Committee advises on radiation safety issues and gives expert opinions on the medical uses of radiation and radioisotopes. It also advises the AEC management, as required, on matters of radiation policy.

Advisory Committee on Environmental Protection of the Fourth Nuclear Power Station

The Committee consists of 19 members with expertise in science and engineering with emphasis on environmental protection. The members also include representatives from the Taipei county, tourist bureau, mayors and council chairmen from two local townships. This Committee meets every 2 to 3 months to discuss TPC's implementation of environmental protection and monitoring items and advises AEC on various environmental concerns during the construction period. For example, air quality, noise, vibration, traffic conditions, disposal of excavated soil, liquid discharge from the construction facilities and living complex. It also advises the AEC on landscape gardening.

Advisory Committee on Nuclear Accident Investigation and Evaluation

The Committee will be setup after a major nuclear accident and damage claims from the public. The Committee will investigate the accident, its extent, injuries or fatalities of the public, and property damages. The results of the investigation will be the basis for liability claims under the Nuclear Damage Compensation Act.

Advisory Committee on Nuclear Legislation

The Committee consists of more than 10 members from relevant agencies within the Executive Yuan or private firms with expertise in law or nuclear disciplines. This Committee advises AEC on proposed nuclear legislation before submitting to Executive Yuan or Legislative Yuan for approval, or important lawsuit involving AEC, or petition from the citizen.

8.1.4 Financial and Human Resources

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

8.1.4.1 Funding from Taxpayer and Fees

Two types of fees are requested to partially recover the AEC's budget in addition to taxpayer's money. First, license and safety review fees, established in Appendix 12 of Enforcement Rules of Atomic Energy Act, recover the AEC's costs of providing individually identifiable services to applicants or licensees. The services provided by the AEC are the review of the applications for the issuance of new licenses or approvals, amending or renewing licenses or approvals, and review of reload and topical reports. Second, the annual fees, established by the same Appendix of Enforcement Rules, recover the generic (e.g., inspection, testing and research) and other regulatory costs that are not recovered through license and safety review fees. The amounts of these fees, amended every two to five years, are based on the manpower requirement and their salaries approved by the Parliament.

8.1.5 Position of the AEC in the Governmental Structure

This section explains the relationship of the AEC to the Executive Yuan（Executive Branch）, the Counties, and Legislative Yuan（Parliament）.

8.1.5.1 Executive Yuan

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

Ministry of Foreign Affairs（MOFA）

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

The Directorate General of Budget, Accounting and Statistics（DGBAS）

The DGBAS is the agency responsible for budget, accounting and statistics affairs within the central government as well as local county governments and has been established for 70 years. The AEC submits the annual budget requests, including proposed personnel requirements, to this agency for approval.

Ministry of Economic Affairs（MOEA）

The MOEA is in charge of the matters regarding national economic administration and construction. Its major functions encompass the administration of industry, commerce, trade and international cooperation, small and medium enterprises, investment, intellectual property, technological research and development, energy, water resources, mining, standards, inspection, weights and measures, and subsidiary state-run enterprises（i.e. public corporation）.

The Taiwan Power Company (TPC), established on May 1, 1946, is a state owned public corporation under the MOEA. TPC employs approximately 27,000 personnel with assets worth NT$1,185 billions. As of December 2003, the total installed capacity reached 33,290MWe (Nuclear：5,144MWe, 15.5%； Fossil：23,633MWe, 71.0％； Hydro：4,511MWe, 13.5％； Wind： 2MWe). Its main mission is to maintain the stable supply of electric power with good quality and low price.

Environmental Protection Administration（EPA）

The EPA, a ministerial-level agency within the Executive Yuan, was founded in 1987 with the mission of protecting and improving the environment nationwide. For immediate response, the EPA will stand at the front line to address serious environmental problems on a real-time basis.

The EPA vigorously exercises the authorities granted to implement control measures, to prevent pollution, and to support international environmental initiatives. Through preservation of ecological balance and environmental quality, the ultimate objective is to achieve sustainable development. The annual budget of the EPA is approximately NT$11.44 billion for the year 2003.

After passage of Environmental Impact Assessment Act in December 1994, the review of the environmental report of a new nuclear power station and other nuclear facilities, e.g., spent fuel interim storage facility and low-level radioactive waste repository, has been shifted from AEC to EPA.

National Fire Fighting Administration (NFA), The Ministry of the Interior

The Disaster Rescue Command Center was formally established in July 2000 after the Chi-Chi Earthquake of September 21, 1999, under the National Fire Fighting Administration Agency（NFA）of 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 the emergency planning and response at the nuclear power stations. The NFA will assist AEC's licensing process especially in the offsite emergency planning and response by document review as well as the observation and evaluation of emergency drills at the nuclear power stations.

Department of Health（DOH）

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

Council of Labor Affairs（CLA）

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

8.1.5.2 The Counties

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

However, some Counties have shown a desire to participate in matters relating to nuclear power stations. In response, the AEC declared its intent to cooperate with Counties in the area of nuclear safety by keeping Counties informed of matters of interest to them, and considering the participation of County officials in AEC inspection activities or the advisory committee. However, Counties are authorized only to observe and assist in AEC inspections, not to conduct their own independent health and safety inspections.

The Taiwan Power Company, the largest producer of radioactive wastes, also plays a major role to convince Counties and Townships in decisions on the siting of low-level or high-level waste repositories.

8.1.5.3 Legislative Yuan（LY）

The Constitution provides that the Legislative Yuan, constituted of public-elected representatives, shall be the supreme legislative organization of the country and exercises 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(3) of organic regulation of Procedure Committee, this

Committee is responsible to deliberate polices of technology, information, public construction, and bills related to powers of Academia Sinica, National Science Council, Atomic Energy Council, Public Construction Commission Executive Yuan, Department of Industrial Technology of Ministry of Economic Affairs, The Directorate General of Telecommunications Ministry of Transportation and Communications, Central Weather Bureau. According to Article 2 of organic regulation of each Committee, Each Committee shall deliberate bills submitted by Legislative Yuan and petitions of the public. At the beginning of a session, each committee may invite government agencies concerned to make business reports and be present at the committee meetings to present their views.

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

Basically the functions of the AEC are separated from those of the Ministry of Economic Affairs（MOEA）. The MOEA, and its affiliated public corporation, Taiwan Power Company, has the responsibility of promoting the peaceful use of nuclear power through research and demonstration projects, and the construction and operation of nuclear power stations.

However, outside the energy sector AEC does promote and at the same time regulate the radiation application. For example, the gamma irradiation plants require AEC license to design, construct and operate, with AEC's regular inspection. On the other hands, AEC also supports research to use gamma irradiation plant for the purpose of improving the health of the public.

8.3 The Policy of “Nuclear-Free Homeland” and its Implementation

“Nuclear-Free Homeland” development is a general consensus among political parties in Taiwan. The Government is in the process of promoting and implementing a variety of programs toward this goal. The contents of the “Nuclear-Free Homeland” are not limited to the nuclear power issues only. It is a multi-dimensional concept that includes “termination of threats by nuclear weapons”, “refusal of nuclear accidents and radiation hazards”, “ensuring safety of nuclear power”, “development of renewable energy sources as alternatives”, and “prospect of nuclear peaceful applications”.

To push for the establishment of a nuclear-free homeland, the Executive Yuan formed the “Committee for the Development of a Nuclear-Free Homeland” in 2002 so as to hammer out a comprehensive energy development plan that would provide sufficient energy while taking into account the economic and social development, and being in line with the global trends and the implications of international treaties. Under the Committee, there were eight subcommittees, which were then restructured into five subcommittees in September 2003. These subcommittees are led by cabinet members to take charge of the different aspects of the development, including:

(1) Contents and driving mechanism for the development of a nuclear-free

homeland;

(2) Planning for the operation of the existing nuclear power stations;
(3) Nuclear waste management and public (community) participation;
(4) Development of energy conservation and clean energy industries; and
(5) Audit of existing nuclear plants and assessment of Lungmen project related issues.

AEC is in charge of "the Subcommittee for Supervising Safety of the Existing Nuclear Power Stations and Assessing Lungmen Project Related Issues". Major tasks include: to reinforce safety inspection on existing nuclear plants, to enhance inspection of construction quality at the Lungmen Station, and to help resolving issues associated with this project. FCMA and TPC's Nuclear Backend Management Department (NBMD) are heavily involved in the implementation of "the Nuclear Waste Management Subcommittee". In addition, both NBMD and INER have active roles in "the Subcommittee for Planning the Operation of Existing Nuclear Plants", particularly in the area of technology development for decommissioning.

As far as "rule making" is concerned, the Cabinet has pledged to present laws to phase out the existing nuclear power stations, to regulate storage sites for nuclear waste and to promote the use of cleaner sources of energy. In November 2002, the Legislative Yuan passed the Environmental Basic Law. In Article 23, the Law requires the Government to set plans and take steps to fulfill, in phases, its goal of turning Taiwan into a nuclear-free homeland. Meanwhile, special emphasis must be placed on nuclear safety regulation, radiation protection, radwaste management and environmental radiation monitoring, so as to protect the public from radiation hazards. To this end, a new law on the development of a nuclear-free homeland was drafted by the Ministry of Economic Affairs and submitted to the Legislative Yuan in May 2003 for review and approval. The bill sets policy directions and basic outlines for steering the implementation of specific development steps towards the goal of nuclear-free homeland by relevant government agencies. The enactment of the law will provide legal basis for amending related laws and establishing necessary regulations and guidelines to carry out relevant actions.

Table 8.1 Budget Authority and Staffing by Appropriation

Appropriation	Budget (Million NT Dollars)		FY Staffing (Man-year)	
	FY2002	FY2003	FY2002	FY2003
AEC Headquarters	351	367	178	178
INER	2,447	2,291	1,044	999
FCMA	74	74	41	39
RMC	61	64	32	30
Total	2,933	2,796	1,295	1,246

Note：Budgets for the purchase and renovation of new AEC headquarters building in 2002 are not included

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 Responsibility of the License Holder

The holder of a construction permit assumes the responsibility to construct a nuclear power station as approved and with the conditions imposed by the regulatory body at the time when the construction permit was issued. On the other hand, the holder of an operating license assumes the responsibility to operate a nuclear power station as approved and with the conditions imposed by the regulatory body at the time when the operating license was issued.

The responsibility of the AEC is the regulatory oversight of licensee activities to ensure safety is maintained. The AEC comprehensively reviews the safety of a nuclear power station design and the capability of an applicant to design, construct, and operate a facility.

Failure to conform to the conditions imposed on the construction permit or operating license would subject the licensee to enforcement actions, which can include fine, modifying, suspending, or revoking the license. The AEC can also order particular corrective actions or may ask the court to take criminal prosecution or fine.

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

The AEC, in accordance with the Atomic Energy Act of 1968, assumes the responsibility to verify, by means of regulatory inspections described in Section 8.1, that the license holder of nuclear installation complies with the license conditions during the design and construction stage and even throughout the lifetime of the plant. If a violation takes place, the AEC immediately requests the license holder to take corrective and complementary measures so as to secure the safety of the nuclear power station.

The operating license applicant of a nuclear power station shall receive pre-operational inspections from AEC to verify that the nuclear power station is constructed as previously approved in Construction Permit. After that the applicant can start nuclear fuel loading. Then, with the completion of all the pre-operational testing, including criticality and power ascending tests, the applicant will receive operating license for commercial operation.

The operator of a nuclear power station 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.

If the operator of a nuclear power station failed to meet the license conditions, the AEC may order the revocation of the license or the suspension of the business for a given period of time. If the performance of the plant did not meet the standards or if safety measures for the operation of the plant were unsatisfactory, the AEC may request the operator to take corrective actions or suspend the operation of the plant.

ATICLE 10. PRIORITY TO SAFETY

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

10.1 Safety Policy

In accordance with the "Atomic Energy Act", the Atomic Energy Council (AEC) was founded as a ministerial level government entity in 1955. The major missions of AEC are nuclear safety regulation, radiation protection, radwaste management, and research & development of atomic science & technology for peaceful application. While carrying out these tasks, the AEC adheres to the principles of "Strict Inspection, Safety First, Stringent Regulation and Information Transparency". Even in the process of research & development of atomic energy application, safety remains to be the highest priority as well.

10.2 Safety Culture

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

TPC's safety culture fostering program consists of three phases, these are the learning phase (1988~1992), the cultivation phase (1992~1997), and the enhancement phase (1997~now). In the period of learning phase, managers of all levels in the nuclear departments were requested to study IAEA safety culture literatures such as INSAG-1, INSAG-3 and INSAG-4. Engineers and operators had to attend training courses and symposiums associated with safety culture fostering program. In addition, managers were also requested to take every opportunity to teach their subordinates about safety culture.

At the beginning of the cultivation phase, the Nuclear Safety Policy Statement of the company was declared that nuclear safety management would be the most important affair in nuclear power generation, and nuclear power stations were to stress the efforts on establishing safety culture. The Statement also emphasized the necessity of responsibility, discipline, self-evaluation, prevention of abnormal events, and pursuit of excellent performance. To implement these requirements, 62 action items were developed under 5 categories, namely, duty, training, discipline, regulation and execution. In addition, 18 safety culture indicators were selected with regards to these 5 categories to evaluate the performance of nuclear power stations. These indicators were:

- Indicators related to Duty
  (1) Proposals of plant betterment and its effectiveness
  (2)Abnormal events initiated by human error
  (3)Abnormal events initiated by procedure deficiency
  (4)Repeated abnormal events.

- Indicators related to Training
  (1) Repeated abnormal events initiated by human error

- Indicators related to Discipline
  (1) Events of violation judged by regulatory authority （AEC）

- Indicators related to Regulation
  (1) Deficiencies found by AEC inspectors before TPC's auditors did.
  (2) Deficiencies found by AEC inspectors before corrective actions were performed.

- Indicators related to Execution
  Same as the ten performance indicators set by World Association of Nuclear Operators (WANO).

In the period of enhancement phase, the results in the previous two phases were reviewed and the safety culture program was modified based on the past experience. The following targets and action items were developed in the modified program:

<u>Target 1</u>: Declare safety commitment and establish safety culture

(1) Sign the "safety commitment".
(2) Use organization's activities and rewards to promote employee's work performance and safety culture level.
(3) Establish safety culture from the top down to the bottom.
(4) Require employees of all levels pay attention to safety.
(5) Enhance "Walking Management" of every one, especially for managers of all levels.

<u>Target 2</u>: Practice the following 10 human error preventive measures

(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 contractor's employee
(9) Experience feedback
(10) Human error root cause analysis.

<u>Target 3</u>: Establish good procedures and request procedure adherence

(1) Push the procedure adoption program to work; require the procedure adopter and user to take the responsibility of the integrity and correctness of the procedures.

(2) Require managers of all levels to guide their subordinates to follow the procedures.

<u>Target 4</u>: Promote self-evaluation capability

(1) Push the self-evaluation program to work
(2) Push the systematic assessment program for plant system to work.

<u>Target 5</u>: Promote 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 solve the potential problems in a system.
(3) Perform "root cause analysis" and monitor the remedy actions for the correction effectiveness.

<u>Target 6</u>: Promote training effectiveness

(1) Push the personnel qualification and certification program to work.
(2) Improve the quality of training program.
(3) Establish an unbiased and objective evaluation system for training programs.

Along with the above programs, TPC also conducted safety culture assessment to evaluate the effects of these programs on safety performance and to pinpoint the weakness associated. This assessment included two parts, one was safety culture indicators review and the other was performance evaluation. For the latter part, a team consists of members from TPC head office and three nuclear power stations went to each plant site for the safety culture performance evaluation. In addition, the Commission of National Corporations of Ministry of Economic Affairs, a supervisory organization of TPC, also organized a team with experts from universities, government agencies and news media to assess the plant safety culture annually. All findings by these teams were fed back to the plants for the improvement of safety culture.

10.3 Commitment to Safety

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

On the nuclear power station side, in order to enhance the safety of nuclear power generation, TPC announced a "Nuclear Power Operational Safety Policy" at the beginning stage of the safety culture fostering period. The Policy Statements, revised on November 8, 2001, included the following:

(1) The nuclear power operational safety is the responsibility of every one involved in nuclear power generation business.

(2) The Acts, regulations, standards, specifications and operating procedures related to 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 requirements arise. The existing rules must be strictly followed until new ones are approved by appropriate authority.

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

(4) To conform to the regulatory requirements, TPC has to do its best to prevent the occurrence of abnormal events and regulation violations

(5) TPC's goal is not only to look for the superior performance of operational safety, but also to conform to the strict requirements of regulations.

10.4 Regulatory Control

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

AEC was founded in 1955 at the ministerial level under the Executive Yuan. Principal mission of 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 atomic energy in the country. In recent years, the most important tasks of AEC have been shifted to safety regulation, radiation protection, radwaste administration, and R&D for nuclear technology and civilian nuclear applications. The legislative and regulatory framework, Acts, regulations, and requirements associated with nuclear safety are described in Article 7. And the structure and responsibilities of AEC are introduced in Article 8.

10.5 Voluntary Activities and Good Practices Related to Safety

Among many voluntary activities related to nuclear safety, the first one worth mention is the experience feedback. In order to learn from the past experience, worldwide operational as well as regulatory information are constantly collected and studied by AEC and 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 one domestic plant were reported to the other plants, so that similar mistakes can be avoided and good safety measures can be shared. To share important operating and maintenance experiences among plants, TPC worked out a program, namely, the Operation Experience (OE) program, which can be applied to all nuclear installations. It turns out that the OE program is a sharp tool to seek ways of improving the performance of nuclear power station. Besides the experience feed back 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

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

(2) Investigation of Reactor Scram and Forced Outages

All of the six operating nuclear power units in Taiwan, including four BWRs and two PWRs, are designed and fabricated by the United States vendors. Therefore, all activities essential to nuclear power station, 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 USNRC. For this reason, the permission to restart the unit after refueling outage was not necessary for AEC to approve in earlier years of operations. However, for reducing the frequency of nuclear unit scrams and forced outages, AEC has decided to regulate the unit restart after refueling outage to assure the maintenance quality of structures, systems, and components (SSCs) of the facility and to improve plant performance since 1987.

Besides, in case of a reactor scram, TPC must report to AEC about the consequence and probable root causes of the scram within two hours after its occurrence. 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, TPC has to submit an application in advance with necessary documents about its causes, procedures of modification, safety assessment and so on. 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, TPC has to investigate the root causes, propose remedy measures and submit a report to AEC. AEC will review the remedy actions and dispatch inspectors for field inspection if necessary. The implementation of the measures will be followed up by AEC until the issue is effectively resolved.

(4) Investigation of Plant Equipment Malfunction

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

(5) Development of Severe Accident Management Guidelines (SAMGs)

After the Three Mile Island (TMI) accident, the nuclear industry performed a large-scale severe accident research to understand the phenomena and develop analysis code for improving the prediction capability. The goal of the severe accident research is to develop a Severe Accident Management Guideline (SAMG) for the plant staff to mitigate the severe accident. At the end of 2003, Taipower has established SAMGs specific to Chinshan, Kuosheng and Maanshan nuclear power station, respectively. The development of SAMG includes the evaluation of system status (hardware capability), plant control parameters (Instrumentation availability), interface among EOP and SAMG, verification of SAMG and training. The SAMGs of the three existing NPSs and TSGs of two existing BWRs can enhance the severe accident management capability of the plant staff including the members of Accident Management Team and the operators.

(6) International Technical Evaluation and Peer Review

TPC has invited several international nuclear groups, such as INPO and WANO, for safety review and discussion. For example, during the last few years, TPC had the following activities:

(a) On May 30 to June 1, 2001, two specialists from Kansai Electric Power Company of Japan visited Maanshan NPS to share the experiences gained by the plant in the Station Blackout event, occurred at Unit One.
(b) On August 8-17, 2002, US INPO sent four specialists to Kuosheng NPS for technical exchange visit
(c) On May 27 to June 14, 2002, a team of twenty-one specialists organized by WANO-TC visited Chinshan NPS for WANO peer review.
(d) On November 10-14, 2003, US INPO sent two maintenance experts to TPC to give a “Maintenance Supervisor Professional Development Seminar”, themed on “Equipment Reliability”

10.6 Measures to Enhance Transparency of Nuclear Safety Information

Communication is a very important mechanism for effective regulation. AEC holds periodic regulatory meetings with the operator to enhance reactor safety. Meetings with stakeholders are also held whenever new laws are enacted, regulations promulgated or policies announced. For public outreach, AEC holds press conferences biweekly 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 website.

To enhance the transparency of nuclear safety information, AEC is taking one step further to make selected real-time data available on the web. Currently, several types of plant operational and environmental monitoring data are transmitted to AEC’s

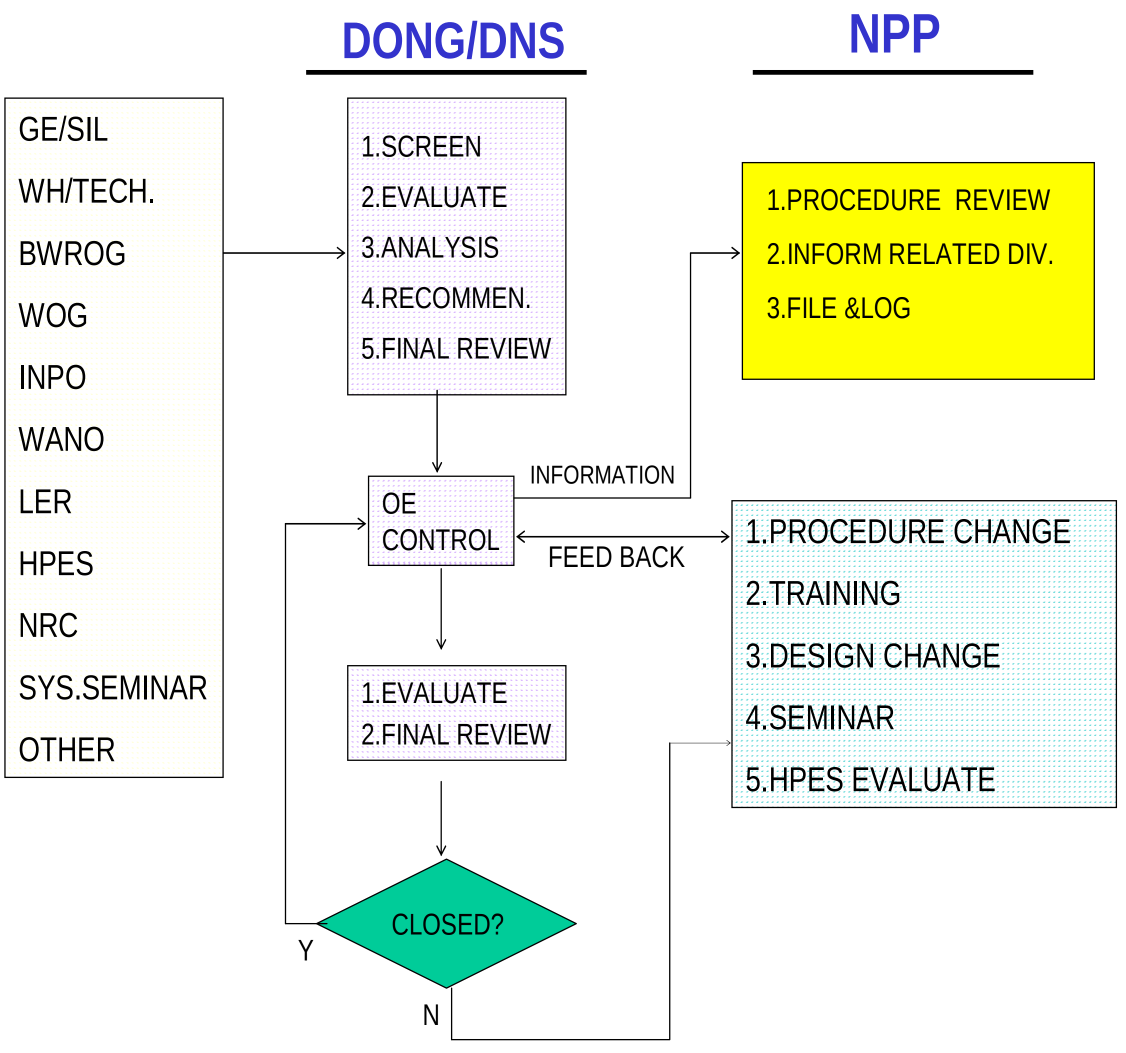
Nuclear Safety Duty Center, a 24-hour working, centralized reporting system for abnormal events and an inter-ministerial communication gateway within the framework of national disaster prevention. During the Year 2004, some of this information will become available to the general public at AEC' s website. First of all, real-time color-coded data of selected parameters from the Safety Parameter Display System (SPDS) will be 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 HPIC readings, updated every hour from data at site boundaries of all nuclear plants. In addition, area gamma radiation, also updated every hour for dozens of sampling stations in the entire Taiwan Area, will be made available－currently such data are on line but only updated twice a day. To help the public to visualize the operation of the nuclear power stations, AEC may post operating data images of each plant on top of the reactor building or turbine building. Images of the control room may also be provided during plant emergencies or drills.

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

AEC also takes part in some of the cooperative activities and training seminars sponsored by the OECD's Nuclear Energy Agency and the International Atomic Energy Agency (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, IAEA conducts safeguards inspections in Taiwan following the spirit of the United Nations' Nuclear Non-Proliferation Treaty and an Additional Protocol with IAEA, United Nations.

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 is sent regularly to the Agency for reporting on "Nonce".

The Nuclear Energy Society, Taipei (NEST), an assembly of representatives of nuclear and radiation related societies and associations, has provided another channel for Taiwan to communicate with international nuclear communities on the subjects of nuclear safety enhancement. 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.



Note:

1. DONG stands for Department of Nuclear Generation of TPC
2. DNS stands for Department of Nuclear Safety of TPC
3. NPS stands for Nuclear Power Station

Figure 10.1 The flow chart of the Operation Experience (OE) program

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.**

This section explains the requirements regarding the financial resources that licensees must have to support the nuclear installation throughout its life, including the resources needed for safety improvements of plant operation, plant decommissioning, and handling of claims and damages from accidents. This section also explains the regulatory requirements for qualifying, training, and retraining personnel.

11.1 Financial and Human Resources of Licensee/Applicant

TPC, a government invested public utility company, is the sole operator of nuclear power stations in Taiwan. TPC is established to strive for the stable supply of electric power through effective energy source development and power management.

TPC is composed of more than 20 departments of general management, operations and business. TPC employs approximately 27,000 personnel with assets worth about NT$1,185 billion（~US$ 35 billion）and total installed capacity of 30,136MWe. The number of employees engaged in the construction and operation of nuclear power stations is approximately 2,500. TPC has 3 nuclear power stations（each with two units）in operation with installed capacity of 5,144MWe and 1 plant（two units）under construction with installed capacity of 2,700MWe.

As shown in Figure 11-1, the nuclear-related organizations in the TPC headquarters include the Department of Nuclear Generation, Department of Nuclear Safety, Department of Nuclear Engineering, Department of Nuclear Backend Management, Department of Fuels, Department of Environmental Protection and Department of Nuclear and Fossil Power Projects. There is also the Power Research Institute, an affiliated research institute; and the Training Center, responsible for the training of nuclear engineers.

In each nuclear power station, as shown in Figure 11-2, there are various Divisions such as operation, mechanical, electrical, instrument, computer, nuclear engineering, chemistry, radwaste, health physics, quality control and simulator center.

The Nuclear Safety Committee (NSC), located in the headquarters of TPC, is an organization for deliberating and decision making on nuclear safety issues. In addition to the Directors of each nuclear-related Department, TPC also invites experts from universities, research institutes, and industries as the Nuclear Safety Committee members. While in each nuclear power station, the Station Operation Review Committee (SORC) is organized to advise the plant superintendent on matters concerning nuclear safety.

Access to adequate funds for the safe construction, operation, decommissioning, and final disposal of spent fuel and waste is necessary to protect the public health and safety. Although there does not appear to be a consistent relationship between a licensee's financial condition and its safety indicator. There is evidence that, at least for some foreign private plants, financial pressure has limited the resources available for corrective actions, improvements, upgrades, and other safety-related expenditures. Further, because a power reactor must provide funds for eventual plant decommissioning and waste disposal, any early shutdown of a plant before accumulating sufficient funds could potentially hinder the safe, expeditious plant decommissioning and waste disposal. The AEC has closely monitored the proposed TPC's privatization program and any possible early closure of the three existing plants.

11.2 Financing of Safety Improvements

TPC established a betterment plan for the safe operation and reliability improvement of each nuclear power station, and planned to secure the required research and development fund on its own. TPC has replaced and/or reinforced its facilities under the Mid- and Long-term Betterment Program. As an example, TPC has completed the replacement or upgrading of the following systems and equipment, such as simulators, feedwater control system, reactor protection system, plant monitoring system, turbine rotors and field instruments for both BWR and PWR units. Significant investment has been made in the betterment plan. Table 11.1 gives the total number of Design Change Requests resulted from the betterment programs of the three nuclear power stations in the recent years.

The AEC also performs necessary regulatory research and development as part of the Mid-Term and Long-Term Nuclear Energy Research and Development Programs for maintaining safe operation of nuclear power stations and revising regulations resulted from the advancement of nuclear technology and ever-increasing environmental requirements. To this end, the Atomic Energy Act of 1968 stipulates specifics on the promotion of nuclear research and development programs.

11.3 Financial and Human Provisions for Decommissioning Program and Radioactive Waste Management

The Radioactive Waste Management Policy of 1988 as amended in 1997 stipulates that the operator of nuclear power stations shall establish a nuclear back-end fund for the decommissioning of nuclear installations and permanent disposal of spent fuels and low-level radioactive wastes. TPC estimates the cost on the basis of the installed capacity, projected quantity of radioactive waste, the commodity price index and the international experiences. The fund is collected on the basis of the electricity generated by TPC's nuclear power station operation, which was NT$ 0.17/kwh in 2003. The aggregate sum of back-end fund reaches 145.3 billion NT dollars as of December 2003. An ad hoc committee, established under the Ministry of Economic Affairs, has been managing the fund, which is comprised of 13 members from the government organizations and research institute. The AEC has closely monitored the fund related activities because of their significant effects on future decommissioning and final disposal programs.

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

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

11.4 Financial Protection Program for Liability Claims Arising From Accidents

This section explains the financial protection program for liability claims arising from nuclear accidents. It covers the governing rules, primarily the Nuclear Damage Compensation Act, and process to implement the respective requirements.

The Nuclear Damage Compensation Act, enacted in 1971, as amended in 1977 and 1997, governs the financial protection program. It provides the financial and the legal framework to compensate those who suffer bodily injury/fatality or property damage as a result of accidents at the nuclear facilities covered, mainly nuclear power stations. The AEC's regulations implement the provisions of Nuclear Damage Compensation Act.

The Act was enacted to meet two basic objectives:

- Remove the deterrent to both domestic and foreign private industry participation in nuclear energy presented by the threat of potentially enormous liability claims in the event of a catastrophic nuclear accident.

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

In enacting the Act, the AEC sought for the balance between the needs of the industry and those of the public. Specifically, AEC required that all reactor licensees (or "operators"), among other licensees, purchase specified amounts of liability insurance (at a maximum level of NT＄200 million in 1971) or possess other equal financial protection against the risk of a nuclear accident. However, it was also agreed to limit the total liability claims for an accident at NT$4.2 billions.

The financial protection, indemnification and liability limit applied not only to the liability of the licensee but also to the aggregate sum of all liability for all persons who might be held liable. This "omnibus coverage" effectively channeled the financial responsibility for all damages up to the liability limit of the licensee. In so providing, the Act indemnified the suppliers, contractors and others in the nuclear power industry, as needed in the event of an accident, and assured the availability of reasonable compensation no matter what caused the accident.

The Act has been revised twice since 1971, with the most recent one in 1997. As revised over the years, the means for providing financial protection for power reactors have changed slightly. Under the current Act, those reactors must have a maximum amount of NT$4,2 billions insurance or financial guarantee. 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 reactor operator to cover its complete liability, but only to the maximum amount of NT$4,2 billions. However, the operator shall indemnify the government for the loan.

The public is significantly benefited by another feature of the Act. Claimants need only to prove that the accident do 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.

The Enforcement Rules of the Act provides greater details such as functions of Advisory Committee on Nuclear Accident Investigation and Evaluation, designation of the district court, application of limits of liability, coverage of other licensees, permissible claims （including injuries caused by sabotage, coverage for precautionary evacuations, settlement expenses, and punitive damages outside government indemnification）.

11.5 Regulatory Requirements for Qualifying, Training, and Retraining Personnel

The Atomic Energy Act of 1968 and the Enforcement Rules stipulate that only the relevant license holder approved by the AEC can operate the reactor or handle radioactive material, radioisotopes or machines that generate radiation. The licenses are classified as follows:

- license for senior reactor operator,
- license for reactor operator,
- license for radiation safety supervisor (Class 1, 2, 3),
- license for radioisotope operator for non-medical use (Class 1, 2, 3),
- license for ionizing equipment operator for non-medical use (Class 1, 2, 3)

Licenses are issued to applicants who have engaged in the relevant fields with sufficient experience and successfully passed an examination administered by AEC. The total number of license holders employed as of December 2001 by TPC is 446, as shown in Table 11.2. At regular intervals, the license holder must take retraining programs that TPC implements for specific types of licenses.

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

In general, personnel technical training for Nuclear Power Station (NPS) can be categorized as follows:

- Reactor operators training
- Training for non-licensed plant technical staffs
- General employee training.

11.5.1 Reactor Operators Training

Generally the AEC regulations, AEC reactor operators licensing and license renewal guidelines for nuclear power station operators, 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 plant staff.

It is the responsibility of each Nuclear Power Station (NPS) of TPC to select qualified operator candidates to attend the plant operators training program. Each NPS is also responsible to develop and conduct the program. After completion of the operators training program, operator candidates have to pass the TPC in-house operator qualification examination to get their certificates, and then pass the AEC operator license examination to get their operator licenses. In accordance with the AEC regulations, a retraining program for the licensed plant operators shall be developed and conducted by the NPS to assure that the licensed operators receive adequate, appropriate, and required training. Then they are qualified to renew their licenses.

The reactor operators/senior reactor operators (ROs /SROs) training program was developed and conducted in accordance with AEC requirements of plant operators and with reference to the content of AEC examination for operator license. The major training items are as follows:

(1) Nuclear fundamentals training

(2) BWR/PWR system technology training

(3) Simulator training

(4) Plant observation

(5) Control room operating practice.

Nuclear Fundamentals Training

In accordance with the AEC's requirements for plant operators, nuclear fundamentals training shall cover theories of nuclear fission and reactor operation. Major course contents include the followings:

(1) Principles of reactor operation: Atomic structure and reactivity, nuclear reaction and radioactivity, neutron behavior and control of the fission process, core and nuclear steam supply system characteristics, thermal hydraulic design.

(2) Design features of the nuclear power station.

(3) General operating characteristics of the nuclear power station.

(4) Reactor instrumentation and control systems.

(5) Radiation control and safety provisions.

(6) Fundamentals of heat transfer, thermodynamics, and fluid flow related to transient analysis.

BWR/PWR System Technology Training

Major course contents include the followings:

(1) Plant instrumentation and control systems.

(2) Safety, fire, and emergency systems.

(3) Primary and secondary mechanical systems (Nuclear steam supply system and balance of plant).

(4) Electrical systems.

(5) Fuel handling systems.

(6) Waste processing systems.

(7) Integrated plant operation, system interactions, and casualty response.

This course shall also include system and component malfunctions, and the results are documented by a written examination.

Simulator Training

Licensed candidates shall practice the controls of the NPS on a plant-specific simulator, which is a full-scope simulator with the panel configuration and control functions identical to the NPS main control room. Instructions during simulator training shall include 1) standard and emergency operating procedures, 2) plant transients, 3) accident identification and analysis, 4) plant control through a simulator under normal, abnormal, and emergency situations, and 5) operating philosophy, use of procedures, shift and relief turnover, and verification of system status. The training duration shall be no less than 3 months. As a minimum, the licensed candidate shall participate in training sessions that include the followings:

(1) Plant startups including feedbacks of nuclear heating.

(2) Plant shutdown.

(3) Manual control of feedwater during startup and shutdown.

(4) Any significant power changes due to manual changes in control rod position or recirculation flow (BWR only).

(5) Any reactor power change of 10% or greater where load change is performed with load limit control.

(6) Loss of coolant that includes (a) Inside and/or outside primary containment (b) Large or small breaks, including leak rate determination.

(7) Loss of instrument air.

(8) Loss of electrical power.

(9) Loss of core coolant flow/natural circulation.

(10) Loss of condenser vacuum.

(11) Loss of service water required for safety.

(12) Loss of shutdown cooling.

(13) Loss of component cooling system or cooling to individual component.

(14) Loss of feedwater or feedwater failure.

(15) Loss of protective channel.

(16) Dispositional control rod or rods.

(17) Inability to drive control rods.

(18) Conditions requiring use of standby liquid control system.

(19) Fuel failure or high activity in reactor coolant or off gas.

(20) Turbine or generator trip.

(21) Malfunction of automatic control system which affects reactivity.

(22) Malfunction of reactor pressure system.

(23) Reactor trip.

(24) Main steam line break (inside or outside containment).

(25) Nuclear instrumentation failure.

Plant Observation

Planned systematic observation training of license candidates shall be conducted on accessible plant equipment. Emphasis shall be on the understanding of system operation, local plant control, system interactions and indication. Documentation of this training will be made of system familiarity check and oral examination.

Control Room Operating Practice

A 3-month operating practice shall be arranged for those who have passed the AEC operator license examination. This training shall take place in the NPS central control room. Trainees shall observe the operating practices and maneuver the power station from a central control room under the supervision of a licensed senior operator.

11.5.2 Licensed Operator Retraining Program

TPC also conducts an annual retraining program for licensed operators, rotating in a six-group three-shift system, to maintain the proficiency of plant operation skill. 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, Licensed Event Reports (LERs), and human errors as applicable to their area of responsibility. This retraining program includes lectures, simulator training, plant tours, and annual re-qualification examinations including written and oral examinations. It is conducted by the plant itself on a regular and continuing basis, with lectures training no less than 90 hours. Content of the lectures take into consideration the fundamentals and operational proficiency topics as described in the Regulation entitled" License Renewal Guidelines of Nuclear Power Station Operators". The following items shall be covered within a 2-year period:

(1) Fundamentals:

- Applied theory and principles of reactor operation
- Applied heat transfer, fluid flow and thermodynamics
- Features of facility design
- General and specific plant operating characteristics, including expected response to equipment and instrument failure
- Plant instrumentation and control systems
- Engineered safety features
- Radiological protections
- Plant chemistry control
- Fuel handling
- Reactor core design parameters and limits

- Transient and accident analyses and mitigation measures, including accident management
- Supervisory skill training
- Teamwork, leadership, communications, and diagnostics.

(2) Operational proficiency:

- Case studies of related plants and in-house operating experiences
- Normal, abnormal, and emergency operating procedures
- Critical safety functions monitoring
- Prevention and mitigation of core damage including severe accident management
- Technical specifications
- Administrative procedure, conditions, and limitations
- Major operational evaluations
- Facility design and license changes
- Modifications that affect plant operations
- Procedure change and update
- Job-related quality assurance and quality control
- Site emergency plan
- Industrial safety
- Purpose, significance, and basis of the key steps in the emergency operating procedures (EOPs) and their effects on the critical safety functions.

The simulator training shall take no less than 30 hours annually. The contents of simulator training take into consideration the modifications and evolutions of the plant control system, as required by the Regulation entitled" License Renewal Guidelines of Nuclear Power Station Operators". The following items shall be covered within a 2-year period:

- Normal plant evolutions, including testing and administrative duties
- Abnormal nuclear steam supply system evolutions
- Abnormal balance of plant evolutions
- Emergency events that challenge the critical safety functions.

11.5.3 Training for Non-licensed Plant Technical Staffs

Non-licensed plant staffs include 1) Non-licensed on-shift operators of plant system/equipment, radwaste, control and process system/equipment, switchyard, pump house, gas turbine, etc., 2) all categories of engineers for maintenance and engineering support, such as engineers for mechanical maintenance, electrical maintenance, instrument and control, nuclear engineering, chemistry/radiochemistry, health physics, quality assurance and quality control, and computer engineering, and 3) all categories of technicians for maintenance and engineering support, such as technicians for mechanical maintenance, electrical maintenance, instrumentation and control, chemistry/radiochemistry, health physics, and quality assurance and quality control. 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 the systematic approach to training (SAT) method, a performance based method containing following elements: analysis, design, development, implementation, and evaluation. Training programs shall be developed after determining job performance requirements through the process of job and task analysis for the personnel of each category. Training program shall be updated to reflect results of program evaluations, changes of regulations, changes in the facility, and lessons learned from industry experiences. A system for periodic review of initial and continuing training programs shall be established to assess instruction and program effectiveness in helping trainees to meet performance requirements.

Initial Training

For every category of plant personnel, an initial 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 may be exempted from that specific training based on their prior education, experience, and training.

Continuing Training

For every category of plant personnel, continuing training programs shall be implemented to maintain and enhance proficiency of the plant. These programs shall include the following as they are important to the employee performance:

(1) Significant plant system and component changes.

(2) Applicable procedure changes.

(3) Applicable industry operating experiences.

(4) Selected fundamentals with emphasis on knowledge and skill necessary to nuclear safety.

(5) 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 NPS who perform safety-related functions remain cognizant of changes to the facility, procedures, governmental regulations, quality assurance requirements as well as industry operating experiences, and personnel applicable to their area of responsibility.

BWR/PWR System Technology Training

All non-licensed plant staffs including engineers and technicians shall also take system technology training with different training period as required. This training will be conducted by the NPS itself, and the training material will be in Chinese and developed by the staffs who have completed the manufacturer's system technology training.

11.5.4 General Employee Training

All employees of the NPS and NPS contractors shall be trained in the following areas commensurate with their job duties:

(1) General description of the plant and facilities

(2) Job related policies, procedures, and instructions

(3) Radiological health and safety program

(4) Station emergency plans

(5) Industrial safety program

(6) Fire protection program

(7) Security program

(8) Quality assurance program.

11.5.5 Training Centers

In 1998, TPC established a comprehensive Nuclear Power Maintenance Training Facility within the affiliated Training Center for the purpose of developing technical manpower. This training facility is fully equipped with large mock-ups of various equipment and facilities, including spent fuel pool, fuel-loading facility, and reactor coolant pump to enhance the maintenance capability of the personnel. At each nuclear power station site, the local training center is equipped with a full-scope simulator and small-scale mock-ups for training.

11.6 Regulatory Related Training

AEC provides its staff with systematic training to maintain their professional capabilities up-to-date as to meet the ever-increasing regulatory challenges. For example, a course of 12-24 weeks on BWR or PWR technology and simulator training is a requisite for resident inspectors at nuclear power stations; advanced technology training courses are followed to enhance the capability of the inspectors. In addition, selected staff members are dispatched to regulatory agencies and research institutes in nuclear advanced countries for on-the-job training.

Table 11.1 Total Numbers of the Design Change Requests that have been finished in the recent four years

	CS0	CS1	CS2	KS0	KS1	KS2	MS0	MS1	MS2
1998	9	50	26	20	21	63	25	125	44
1999	12	58	53	9	47	12	31	52	127
2000	13	22	61	26	42	38	15	103	106
2001	14	56	17	30	31	53	26	127	42

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

Table 11.2 Number of License Holders Employed by TPC

As of December 2001

Type of License	Chinshan	Kuosheng	Maanshan
Senior Reactor Operator	31	31	24
Reactor Operator	18	25	25
Radiation Safety Supervisor	27	56	34
Radioisotope and Ionizing Equipment Operator	22	61	92
Total	98	173	175

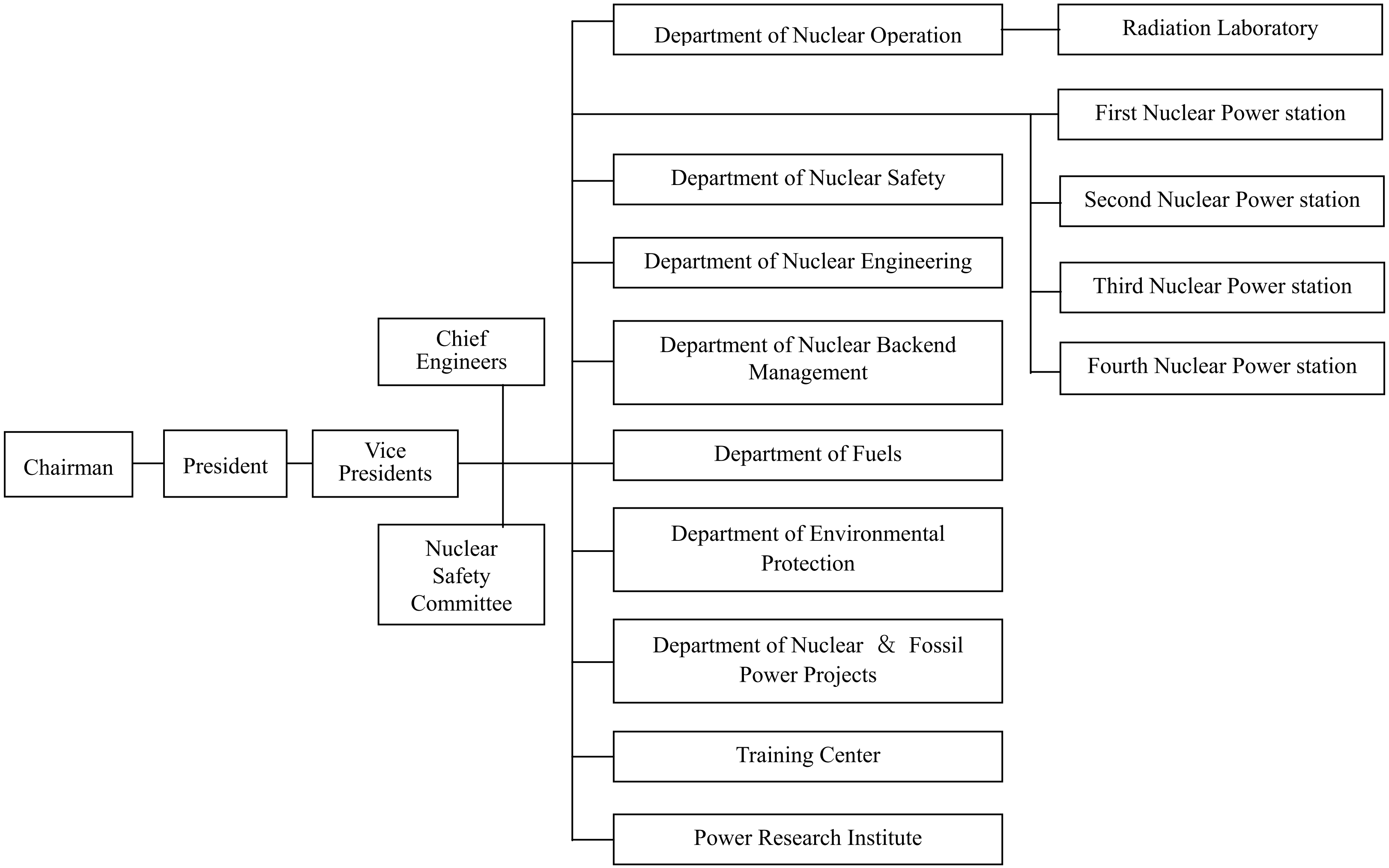


Figure 11.1 Organization Chart of the Nuclear Sector in Taiwan Power Company

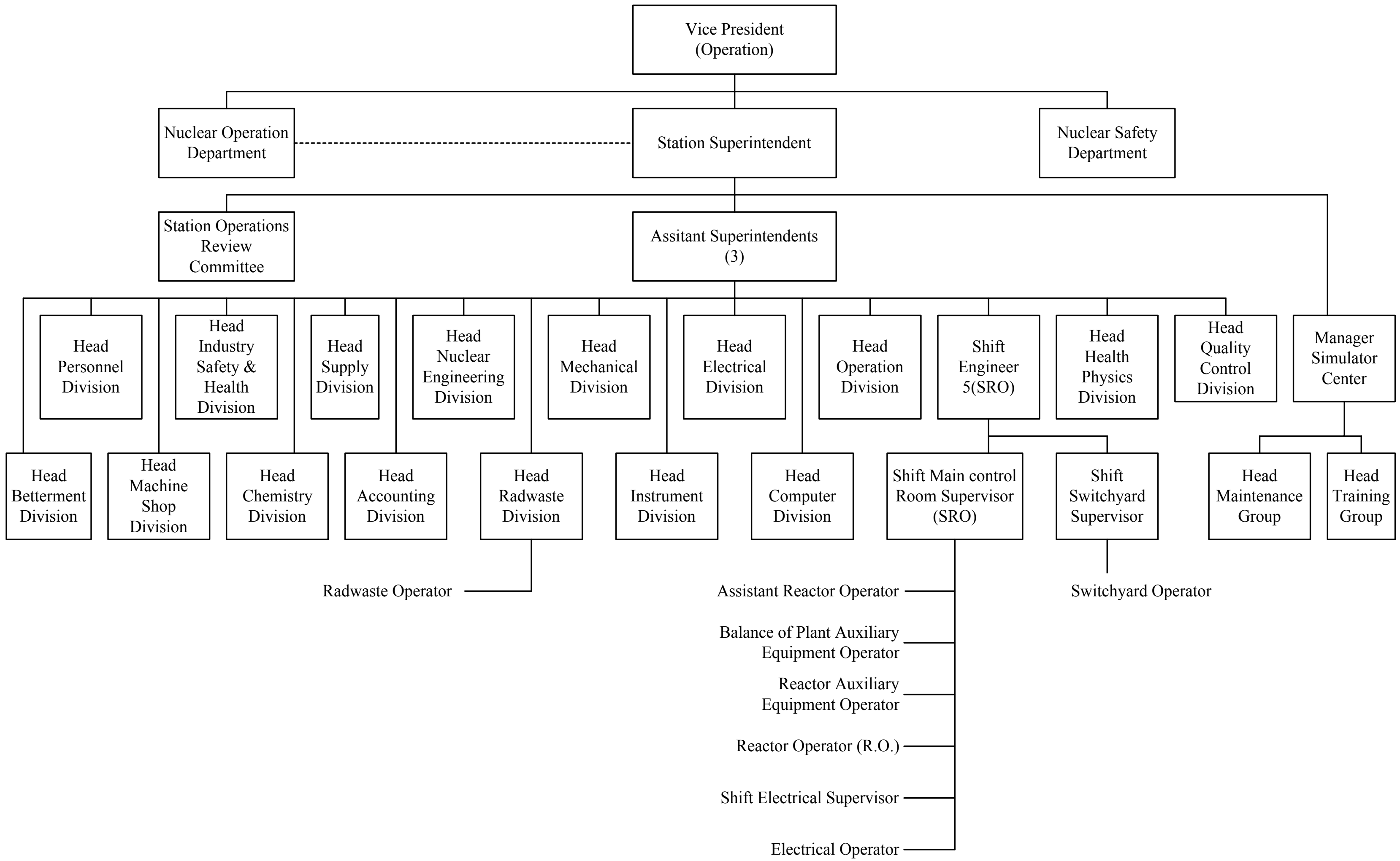


Figure 11.2 General Organization Chart of the TPC Nuclear Power Station

ARTICLE 12. HUMAN FACTOR

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

12.1 Methods to Prevent, Detect, and Correct Human Errors

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

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

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

(4) For the purpose of reducing human errors, ten preventive measures are reiterated in TPC's safety culture enhancement program, in which the operating experiences are regarded to actively prevent occurrence of repeated events in domestic NPSs. 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 TPC's Licensee Event Reports. Through the process of event screening, evaluation, and analysis, the conclusions will be shared by applicable TPC NPSs via operating experience (OE) feedback system. The relevant NPSs will follow the documents and reflect countermeasures into plant procedures, training, or equipment conditions. 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 SOERs are very valuable and the plant operators used to take the advantage of them for event prevention before hand.

(5) Ten Preventive Measures to Reduce Human Errors

- Double check
- Potential risk evaluation
- Tool Box Meeting
- Self Checking (STAR)
- Adherence to procedure
- Conservative decision making
- Enforce the coordination within the O&M Group
- Reduce the human errors of vendors and contractors
- Experience Feedback and training
- Root cause analysis of human error type events.

These preventive measures form the "Barriers for the Prevention of Human Errors" as shown in the following page.

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

(7) In order to evaluate the plants safety, the INER and TPC have collaboratively developed living Probabilistic Risk Assessment (PRA) models on all three operating NPSs in 1996. The Human Reliability Analysis (HRA) is an important issue of the models. According to such factors as man-machine interface, complexity of task, working environment, stress, timing, training, procedure, and 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 miscalibration probability of 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 very valuable for the actual reduction of human errors in plant operation.

(8) To contractors, TPC has long implemented preventive measures as follows to ensure the safety.

A. Plant Orientation Training – Workers hired by contractors should take the following training courses before entering the plant as requested by plant procedures. Contracted workers have to pass the test before qualified for the work. The qualification is valid for one year
   (a) Safeguards and entrance control
   (b) Industrial safety and sanitation
   (c) Radiation protection
   (d) Environmental and radioactive waste management
   (e) Quality control
   (f) Emergency plan.
B. Pre-job training
   The training basically will be based on trainee's work scope to setup the course and usually includes mock up training.
C. Trainee's qualification and license confirmation required.
D. Onsite management
   Contractor should assign a foreman with engineering and management experiences to reside and supervise on-site.
E. Self verification and experience feedback
F. Evaluation of the contracted work by TPC
G. Penalty and warranty terms.

12.2 Managerial and Organizational Issues

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

Whenever a human error event occurs, the plant operator needs to work out a HPES (Human Performance Enhancement System) 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 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 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 the Table on the following page) that were judged to have influences on the maintenance of the nuclear power stations. Questionnaires related to these factors were solicited from 35 managers and 372 engineers on the sites. The results, somewhat to be surprised, showed that substantial differences existed in the choice of factors and their relative importance to the maintenance performance between the group of managers and the group of engineers. Therefore, the plant managers were recommended to pay more attention to those various areas and to foster a more consistent perception among plant employees. 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.

12.3 Role of the Regulatory Body and the Facility Operator

12.3.1 The Role of the Regulatory Body

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

12.3.2 The Role of the Facility Operator

To keep good human performance in nuclear power stations, TPC plays a key role in the prevention, detection, and correction of human errors. The AEC's requirements associated with human factors are the baselines for TPC to follow. In addition, 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 section 12.1 and 12.2.

Table 12.1 O&M Factors With Impacts on Nuclear Power Station Maintenance

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

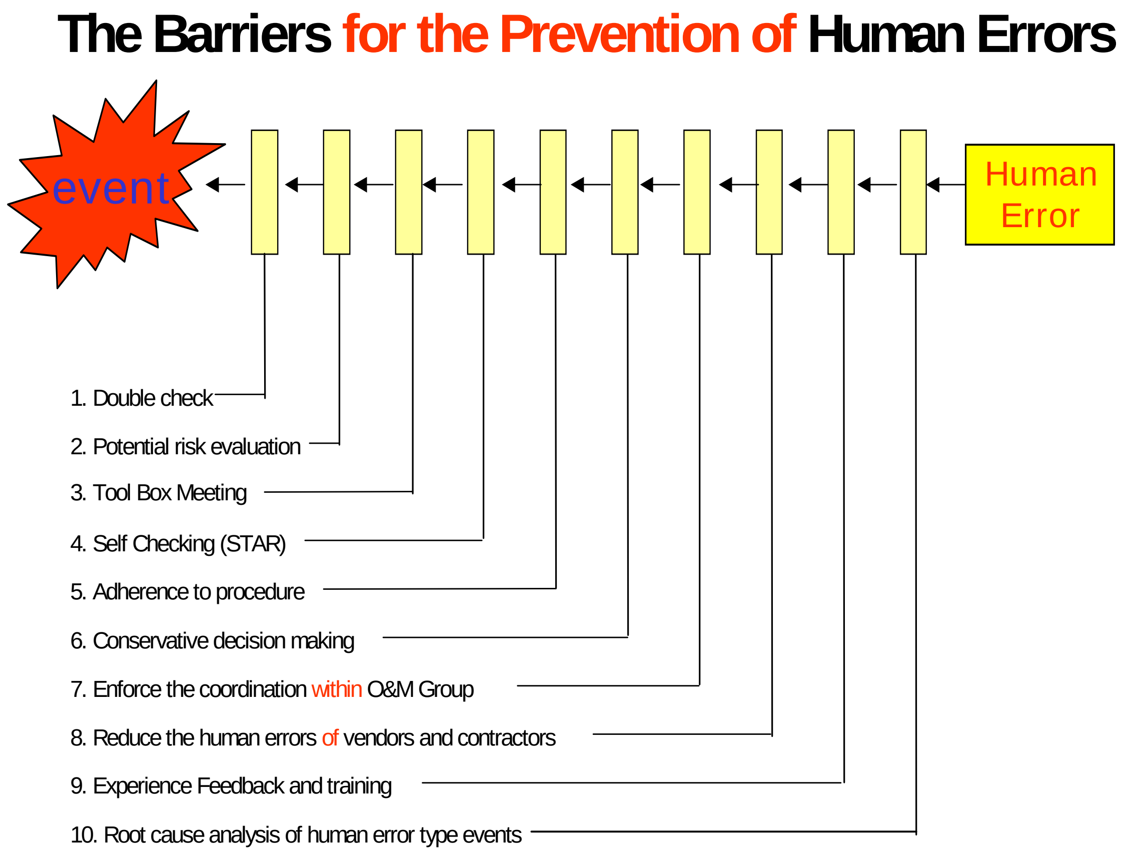


Figure 12.1 The Barriers for the Prevention of Human Errors

ARTICLE 13 QUALITY ASSURANCE

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

13.1 Quality Assurance Programs

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

Each applicant for a construction permit of a nuclear power station 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 permit will be issued after PSAR is reviewed and approved by the AEC. To verify the implementation of QA program during the design and the construction stage, the AEC will perform onsite inspections in accordance with Article 12 of the Enforcement Rules for the Implementation of Nuclear Reactor Facilities Regulation Act.

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

1. Total quality management system shall be established based on the national or international standard. Total quality management shall be undertaken with continuous improvement activities to promote 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 regulation shall be fulfilled to assure the nuclear safety and public health.

Specifically, a nuclear engineering QA program is established before a nuclear facility to be built. A nuclear operation QA program is established for the safe operation of a licensed nuclear power facility. 10CFR50 Appendix B of U.S.A. is adopted in both QA programs.

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, construction, to preoperational testing for all new projects, as well as any specifically nuclear related works.

In 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 10CFR50 Appendix B and ANSI N45.2. In addition, the QA program for the Lungmen Project meets the applicable United States Nuclear Regulatory Commission (USNRC) regulations, and the ASME NQA-1 and NQA-1a requirements. Appropriate and practical quality requirements such as that in the ISO-9001 program will be applied to non-safety yet reliability-related structures, systems and components to meet the TPC's requirements of a total quality management program.

For each nuclear facility with operating license, a nuclear operation QA program is established by the Nuclear Safety Department 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.
- Provide feedbacks to the management on quality of performance.

During 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 will implement its QA program and at the same time supervise the implementation of it into the plant constructors QA program. The latter will be through plant constructor's standard procedures supplemented with approved Lungmen project procedures and procedures addressing unique 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 construction. A more detailed description of the implementation of inspections is provided in Article 14 of this report.

13.3 Configuration Management

13.3.1 Purpose of Configuration Management

Configuration management is an essential tool for managing high quality engineering activities and many requirements contained in 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 station are correctly incorporated in the appropriate documents. The configuration management plan of Lungmen project, first time for TPC's nuclear power stations, was based on the INPO 87-006 report. A computerized Information Management System 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 the Information Management System, required document and information can be quickly and correctly retrieved.

13.3.2 Configuration Management Plan

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

1. Purpose
2. Scope
3. Framework of configuration management
4. Design bases
5. Design and design change control
6. Design document control
7. Evaluation of the configuration management process
8. Interfaces and integration
9. Working procedures for configuration management plan
10. Computer codes for design and configuration management.

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

Design Documents in Configuration Management

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

ARTICLE 14. ASSESSMENT AND VERIFICATION OF SAFETY

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

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

(ii) verification by analysis, surveillance, testing, and inspection is carried out to ensure that the physical state and the operation of nuclear installations continues to be in assurance with its design, applicable national safety requirements, and operational limits and conditions.

14.1 Ensuring Safety Assessment throughout Plant Life

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

14.1.1 Safety Assessment before Operation Stage

The licensing procedures for nuclear installations, pursuant to the Nuclear Reactor Facilities Regulation Act and described in Articles 5 and 6, consist of two steps: the construction permit and the operating license. The applicant for a construction permit 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, preliminary safety analysis report (PSAR) and 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 assessment report to the Environmental Protection Administration in order to fulfill the licensing requirements. More detailed descriptions of the requirements for environmental impact assessment are provided in Article 17 of this report.

Regulations for the Construction Permit Application, enacted pursuant to Article 5 of the Nuclear Reactor Facilities Regulation Act, describe the required contents in the Preliminary Safety Analysis Report (PSAR). Regulation for the Preliminary Safety Analysis Report (FSAR), which is required by the Article 6 of Nuclear Reactor Facilities Regulation Act, is currently underway in AEC. Because the content can be covered by the standard SAR of the country of origin, the contents of PSAR and FSAR for Chinshan, Kuosheng and Maanshan Nuclear Power Stations are essentially the same as that required in the country of origin. For Lungmen Nuclear Power Station (LNPS), the contents of PSAR are based on that of the standard SAR of ABWR. There are two chapters more compared with the standard PSAR and FSAR. In addition to that, five more appendices are added due to the requirements of the AEC. The added contents are:

- Chapter 18 : Human Factors Engineering
- Chapter 19 : Severe Accident Analysis
- Appendix A : Probabilistic Risk Assessment
- Appendix B : Integrated Reliability Analysis
- Appendix C : Emergency Planning
- Appendix D : Backend Program
- 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 Lungmen project.

The severe accident analysis is performed to show that the regulatory requirements and the severe accident policy established by the USNRC (regulatory body of the country of origin) for advanced LWRs can be met. The probabilistic risk assessments (PRA) of LNPS 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 LNPS. 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 LNPS 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 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.

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 stations.

14.1.2 Safety Assessment at Operation Stage

Licensees are required to update their final safety analysis reports periodically to incorporate the revised information and analyses that they submitted to the AEC. The description of the FSAR update is provided in article 6 of this report. In response to the promulgation of the Act of Administrative Procedure, modification of relative regulations required by the Nuclear Reactor Facilities Regulation Act is underway. The proposed requirements related to the FSAR update are described in this section. The current as well as the proposed requirements for the periodic integrated safety assessments are also provided in this section.

In the proposed version of the regulation, the first update of FSAR shall be completed within two years after the operating license is granted. The follow up FSAR updates shall be completed within six months after each fuel reload. If one FSAR is shared by multiple units, the reference date will be set by the latest unit.

According to the Article 6 of the Nuclear Reactor Facilities Regulation Act, the valid period of the operating license shall be forty years at longest, and 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 with the competent authorities within the period prescribed by the competent authorities. The operation thereof shall not be continued without the renewal of license in accordance with the prescription.

Article 9 of the Act further stipulates that after nuclear reactor facilities have been formally operated, one integrated safety assessment at least shall be implemented every ten years and then be submitted to the competent authorities for review and approval.

According to current "Guidelines for the renewal of Nuclear Power Plant Operating License", the applicant shall submit three documents for the license renewal:

- A completed application form
- The most updated FSAR
- The assessment report of the current status of the unit.

The current status assessment report shall include the contents as shown in Table 14.1. Aging management as discussed in Section 14.1.4 is usually included in this report.

As it is mentioned above, modification of the regulation is underway. In the proposed version of the regulation, no renewal of the licensee's operating license is required. However, an integrated safety assessments report with almost the same contents and requirements as those in the current license renewal application is still required.

14.1.3 Design Changes

A nuclear power station is required to be operated in accordance with the license as described in its final safety analysis report. Whenever design change or equipment overhaul is required, all works must be prepared in accordance with Nuclear Reactor

Facilities Regulation Act or relevant regulations. AEC's approval is required for the following design changes before implementation:

- A change to the technical specifications
- A new safety issue is involved with a change
- A change is made to the safety grade equipment which requires the modification of FSAR and its quality is below the original design standard
- Items requested by the AEC.

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

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

For design changes requiring AEC's approval, an assessment report should be submitted to the AEC. The AEC will organize a special task force for review. The change request could be started only if a satisfactory review conclusion has been reached. The inspections are conducted during the work. After the completion of the design changes, proof tests are performed to assure the quality of the work.

14.1.4 Aging Management

With the life extension of nuclear power stations, aging related issues receive more and more attention. Intensive world-wide efforts have been put into the development of Life Cycle Management (LCM) for License Renewal programs. Existing programs such as maintenance, surveillance, testing and inspection can be linked and integrated into the overall LCM strategies. The objectives of the aging assessment for LCM together with the existing maintenance programs are to take control and make timely improvement of the aging related problems to assure the safe operation during the designed service life.

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

The TPC has performed several aging related studies. The first step was to collect and review related documents and to identify the components that are important to system function and subjected to aging. Aging assessment of the entire nuclear power station is very complicated. Therefore, a pilot study on certain selected systems was established. In a three year project, a BWR High Pressure Coolant Injection System, a BWR Residual Heat Removal System, and a PWR Condensate and Feedwater system have been examined for demonstration purpose. The objective of this demonstration project is to select one system from each power station so that a general methodology for aging assessment can be developed and further used in other related systems.

The aging of the reactor pressure vessel (RPV) and the RPV internals are important issues. According to the surveillance program for RPV material test samples are removed from the RPV and assessed for every 10 years. The assessment of the RPV material toughness showed that sufficient margin was maintained. As for the RPV internal components, improvements on the material, water chemistry and stress distribution have been made. The visual inspection procedure for the RPV internal components has also been enhanced to minimize the probability of potential damage.

Currently there are three ongoing aging management related projects:

- Impact assessment of environmental enhanced corrosion of RPV.
- System development and application of technical assessment of aging management.
- Investigation of the stress corrosion cracking of steam generator secondary side for Maanshan Nuclear Power Station.

14.1.4.2 Future Perspective

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

14.2 Verification by Analysis, Surveillance, Testing and Inspection

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

14.2.1 Nuclear Power Station Inspection

The inspection of nuclear facilities is one of the most important tasks of the AEC, with the following approaches for a normal operating plant:

- Resident plant inspectors perform daily monitoring and inspection on site. The resident inspectors should be well informed and in good control of the plant operating conditions.
- Periodic outage inspections are performed to assure the quality of maintenance works by means of group inspection.
- Unannounced inspections are performed without pre-notice to test the alertness of the plant operators.

- Taskforce inspection to review the complete plant operating condition is conducted at each site to assure the nuclear safety. The scope of this inspection includes all primary items related to the safety of nuclear power station.

The inspections at the construction stage are similar to that for the normal operation and are described as in the following:

(1) Resident Inspection

The responsibilities of the resident inspector during construction are (i) to report the daily construction activities to the headquarters, (ii) to monitor the implementation of the quality assurance program, (iii) to survey 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, the AE Company, and the Lungmen site office are inspected using the guidance of the integrated design inspection program published by the USNRC, and the quality assurance requirements.

During the manufacture stage, the inspection activities are focused on the manufacturing quality of the equipment and components significant to safety. Examples are reactor pressure vessel, re-circulation internal pump, fine motion control rod, and reinforced concrete containment vessel liner.

Three categories of inspections: civil and structure, mechanical and piping, and 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, safeguard and administrative control, and quality control and quality assurance functions. The mechanical and piping inspection works are focused on welding, nondestructive 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, experts outside the AEC are invited to join the inspection team. The followings showed the concerned areas for special taskforce inspections：

(i) Civil and Structure

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

(ii) Special Process Control

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

(iii) Human Factor Engineering

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

(iv) 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 to be selected for the key holding point are listed in the following.

i. Reactor building base mat first concrete pouring.

ii. Reactor pressure vessel installation.

iii. Safety related mechanical equipment installation initiation.

iv. Safety related piping system installation initiation.

v. Safety related instrumentation and control equipment installation initiation.

vi. Safety related electrical equipment installation initiation.

vii. Containment integrity functional test.

viii. Reactor protection system functional test initiation.

ix. Cold hydrostatic test.

x. Simulator operator training initiation.

xi. Pre-operational test initiation.

xii. System integration functional test.

xiii. Initial fuel loading.

xiv. Initial criticality and safety margin test.

xv. Turbine rolling and initial synchronization.

xvi. Power ascension test (i.e., 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 initial fuel loading, initial criticality, turbine rolling, generator synchronization, and 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 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 operation and maintenance 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.

14.2.2 Reload Safety Analysis

For each fuel reload, licensees are required to submit a reload safety analysis report (BWR) or reload safety evaluation report (PWR) to the AEC. The reload safety analysis report (or reload safety evaluation report for PWR) is reviewed and approved by the AEC before the restart of nuclear power station for the next fuel cycle. Extensive review in the areas of fuel mechanical design, thermal hydraulic design, neutronic design, transient analysis, and other affected design or analysis will usually be undertaken in any of the following situations:

- a different fuel vendor is selected,
- a new fuel type of the same vendor is introduced,
- a 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 asked. For example, observation of crud thickness, measurement of oxide layer thickness and internal gas pressure of fuel rod are required when a new fuel type (ATRIUM-10) was proposed to be loaded in Chinshan and Kuosheng Nuclear Power Stations.

14.2.3 Preventive Maintenance

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

14.2.3.1 Daily Preventive Maintenance

A computerized maintenance management program called "Maintenance Management Computerization System (MMCS)" has been developed. All the daily preventive maintenance activities such as work assignment, schedule, notice, performance and validation as well as the information storage and tracking of delayed items are all handled by a subsystem of MMCS.

14.2.3.2 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, items that preventive maintenance were planned to be performed in the given outage and items that required preventive maintenance as selected from monitoring results are 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 the case that the system parameters depart far away from the normal or malfunction of SSCs, the licensee is required to report it to AEC and justify the continued operation. Depending on the situation, safety analysis may be required and extensive review be initiated. For example, cracks have been found in the welds of the cover plates of core shroud support access hole at the Chinshan Unit 1 in 1990. This observation is similar to 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 consequently the core bypass flow. As part of the supporting material to justify continued operation, safety analysis has been performed to show that there is no safety concern for the core bypass flow because the postulated event is less severe than that of a recirculation pump seizure, which is covered in the original FSAR.

As for the spent fuel storage, all the fuels are stored in the fuel pools located in the power station. 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.

Table 14.1 Contents of the Current Status Assessment Report of a Nuclear Power Station

1. Review and self-examination of the operating status of the unit

    1.1. Review and assessment regarding the operation safety

    This includes the review of the following operational performance indicators of the unit: the stability of the unit (scrams), water chemistry, fuel operational performance, safety system performance, fire prevention, abnormal events and events that violate regulations, and personnel training.

    1.2. Review and assessment regarding the radiation safety

    The following items shall be included and analyzed: the radiation dose received by the staff, the activity of the released radioactive material, the impact of radioactive material to the surrounding environment, and the trend of the accumulated dose received by the people in the vicinity.

    1.3. Review and assessment regarding the radioactive waste management

    A summary of the management of radioactive waste in the past and the planning for the coming ten years shall be made. Back-end operational planning shall also be included.

2. Self-examination of the pending items to be performed for improving and strengthening the unit

    2.1. Self-examination to identify the important issues

    Based on the operational status of the unit and associated operating experiences, detailed assessment shall be made to identify any issue which may have negative impact on the operation of the unit for the coming ten years.

    2.2. Description of the committed items for improving and strengthening the unit

    Action plans and schedules shall be proposed for each issue described in the previous item.

3. Summary

ARTICLE 15. RADIATION PROTECTION

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

15.1 Act, Enforcement rules, and Regulations

The Ionizing Radiation Protection Act (IRPA) was enacted in 2002 and came into effect on February 1, 2003. The new Act consists of 5 chapters and 57 Articles. At the same time, 18 daughter regulations took effect for the implementation of IRPA.

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

The IRPA and its daughter regulations prescribe the basic radiation protection principles. Those applied to the nuclear power station are as follows:

- 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.

- Training requirements for the persons working in radiation environment.

The Enforcement Rules and Regulations of the Act specify the details necessary for implementing the Act are as follows:

- Detailed provisions necessary for implementing protective measures against radiation hazards, such as the actions needed for the radiation overexposure incident, and relevant reporting requirements.

- Detailed provisions necessary for implementing the radiological control measures such as designation of a controlled area and radiation work permit.

- Detailed provisions of the performance criteria for the personnel dosimetry service.

The Regulations prescribe technical requirements on radiation protection such as the conditions for radioactive effluent release and the dose limits. The dose limits specified in the Regulation are shown in Table15-1.

15.2 Implementation of Act, Enforcement Rules and Regulations

15.2.1 Radiation Exposure Control and Reduction

● **Implementation of ALARA in the Design and Construction of Nuclear Power Stations**

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

- Equipment capable of producing ionizing radiation is to be installed separately in shielded room 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.
- Appropriate radiation zone classification and access control.

● **Criteria for Radiation Exposure Control**

TPC in practice establishes a target dose limit for radiation workers at 90% of the official limit, 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 superintendent and proper measures are taken.

● **Management of Radiation Work**

It is prescribed in TPC's procedures that any person, intended to have access to the controlled areas for radiation works, shall obtain in advance a radiation work permit. 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 safety personnel from Health Physics Division have to evaluate the expected dose, and if necessary, to further impose special conditions on the worker.

● **Reduction of Occupational Radiation Exposure**

TPC has established and implemented respective targets for reducing 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. 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, TPC evaluates 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 radiation workers to utilize the technique after the results of application or experiences from any past work have been evaluated.

● **Personnel Dosimetry Service and its Verification**

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

All organizations with personnel dosimetry service, including TPC, should obtain approval from the AEC. TPC distributes, collects and reads monthly TLD and informs relevant personnel of the results. These are also reported to AEC on a semiannual basis. Accuracy of the reading is maintained by the accreditation from the Chinese National Laboratory Accreditation（CNLA）Program of the Bureau of Standards, Metrology and Inspection and by inter-laboratory comparison.

● **Radiation Protection Training**

TPC prescribes in the procedure that radiation workers and any personnel having access to 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 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,
- Course for any offenders, 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. If passing the evaluation, the trainee is then qualified to have access to or conduct works in the controlled areas.

15.2.2 Requirements on Radioactive Effluents Release

AEC refers US NRC 10 CFR Part-50 Appendix I – Numerical Guides for design objectives and limiting conditions for operation to meet the criterion "ALARA" 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 nuclear power stations and the relevant dose constraints. According to the regulations, TPC is allowed to discharge the gaseous or liquid effluents into the environment after confirming their concentration is within the allowable limit.

The dose constraints for gaseous effluents, as prescribed in the regulations, are as follows:

- Air absorbed dose from gamma rays: 0.1 mGy/yr・unit
- Air absorbed dose from beta rays: 0.2 mGy/yr・unit
- Effective dose equivalent from external exposure: 0.05 mSv/yr・unit
- Skin dose equivalent: 0.15 mSv/yr・unit
- Organ dose equivalent from internal exposure: 0.15 mSv/yr・unit

The dose constraints for liquid effluents are as follows:

- Effective dose equivalent: 0.03 mSv/yr・unit
- Organ dose equivalent from internal exposure: 0.1 mSv/yr・unit

15.2.3 Assessment of Radiation Doses to the Population around Nuclear Power Stations

The radiation dose to and its effect on the population around nuclear power stations are assessed quarterly according to the Regulation entitled "Estimate of Radiation Dose to the Population around the Nuclear Installation from the Radiation Monitoring Results". The assessments model is based on the radioactivity of liquid and gaseous effluents, the atmospheric conditions, metabolism, and social data including agricultural and marine products of the local community within a radius of 50 km.

15.2.4 Environmental Radiation Monitoring

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 ” Environmental Radiation Monitoring “.

The environmental radiation monitors are installed at 5 stations within a 2 km radius of nuclear power stations, 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 monitoring system and the radiation dose levels can be confirmed, on real time basis, in the Radiation Laboratory and the main control room where the monitors are connected on-line. TLDs are installed on 40 posts for assessing quarterly the cumulative gamma radiation dose of the area within a radius of 50 km around nuclear power stations.

The environmental samples are 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 15.2.

15.3 Regulatory Control Activities

The Radiation Monitoring Center（RMC） of AEC installs and operates the nation-wide environmental radiation monitoring network. RMC measures the radioactivity in airborne dust, fallout, rainwater, drinking water, underground water, livestock products, farm products, soil, milk, and background radiation levels throughout the nation. This enables RMC to quickly detect and properly respond to any abnormal situations in environmental radioactivity. The nationwide environmental radiation monitoring network, as shown in Figure 15-1, consists of the following facilities: an environmental radiation monitoring center in RMC, local monitoring stations at five major cities with large population, one monitoring post at AEC Headquarters, and monitoring posts at three nuclear power station sites. RMC has conducted annually national and international inter-laboratory comparisons on environmental radioactivity measurements for quality control.

Two important regulatory activities have taken effects: reduction of dose limits and active implementation of the ALARA principle. AEC had also conducted a series of projects since July 1996, to incorporate ICRP-60 recommendations into relevant national 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 Safety Standards for Protection against Ionizing Radiation 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.

● **Utilization of Radiation Protection Control System (RPCS)**

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 material and equipment capable of producing ionizing radiation, AEC started the use of 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 material and equipment, etc. into a computer-controlled management system.

Table 15.1 Currently Applicable Dose Limits

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

Table 15.2 Environmental Radiation Monitoring in the Vicinity of Nuclear Power Station

Sample Items	No. of Locations			Sampling Frequency	Analysis Items/Analysis Frequency
	First NPS	Second NPS	Third NPS		
Direct Radiation					
TLD	45	36	32	Continuous	Gamma Dose Rate/ Quarterly
HPIC	5	5	5	Continuous	Gamma Dose Rate /hr
Air					
Particulates	21	19	24	Continuous	Gross β、 γ Spectrum [1]/ Weekly、γ Spectrum / Quarterly、Sr-89,90[2]
Iodine	12	12	15	Continuous	I-131/Weekly
Fallout	1	1	1	Continuous	γ Spectrum /Monthly
Water					
Sea Water	9	8	10	Quarterly	γ Spectrum [3]、H-3[3]/ Monthly、Sr-89,90[2]
Drinking Water	7	6	8	Quarterly	γ Spectrum、H-3/ Quarterly、Sr-89,90[2]、I-131[4]
River	2	4	2	Quarterly	γ Spectrum、H-3/ Quarterly、Sr-89,90[2]
Pond	5	3	3	Quarterly	γ Spectrum、H-3/ Quarterly、Sr-89,90[2]
Ground Water	2	3	2	Quarterly	γ Spectrum、H-3/ Quarterly、Sr-89,90[2]
Precipitation I	2	2	1	Monthly	γ Spectrum / Monthly、H-3/ Quarterly、Sr-89,90[2]
Precipitation II	2	2	1[7]	Rain	γ Spectrum、H-3[7]
Agriculture & Marine Products					
Milk：Cow/Goat	1/1	1/1	1/2	Quarterly	I-131、γ Spectrum / Quarterly、Sr-89,90[2]
Rice	2	3	4	Semiannually	γ Spectrum / Semiannually、Sr-89,90[2]
Vegetables	6	5	6	Semiannually	I-131、γ Spectrum / Semiannually、Sr-89,90[2]
Tea	5	—	—	Semiannually	γ Spectrum / Semiannually、Sr-89,90[2]
Fruits	2	2	—	Annually	γ Spectrum / Annually、Sr-89,90[2]
Vegetables（Root）	3	3	2	Annually	γ Spectrum / Annually、Sr-89,90[2]
Sweet Potato	1	1	—	Annually	γ Spectrum / Annually、Sr-89,90[2]

Vegetables（Stem）[5]	1	—	1	Annually	γ Spectrum / Annually、Sr-89,90[2]
Poultry	3	3	4	Semiannually	γ Spectrum / Semiannually、Sr-89,90[2]
Seaweed	2	2	2	Annually	I-131、γ Spectrum / Annually、Sr-89,90[2]
Sea Fish & Shellfish	5	5	6	Quarterly	γ Spectrum / Quarterly、Sr-89,90[2]
Index Organism					
Acacia〈Land〉	1	1	1	Monthly	γ Spectrum / Monthly
Algae〈Sea〉	1	1	1	Annually	I-131、γ Spectrum / Annually、Sr-89,90[2]
Land & Coast					
Beach Sand	9	12	11	Quarterly[6]	γ Spectrum / Quarterly
Soil	18	19	17	Semiannually	γ Spectrum / Semiannually
Sea Sediment	4	4	4	Semiannually	γ Spectrum / Semiannually

Note: 1.conduct γ Spectrum analysis if weekly Gross β > 4 Bq/ft3。

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.First NPS: 600 m from outlet（SS102）; Second NPS: outlet（SS203）; Third NPS: inlet and outlet（SS303 及 SS305） monthly。

7.conduct tritium analysis during rainy period。

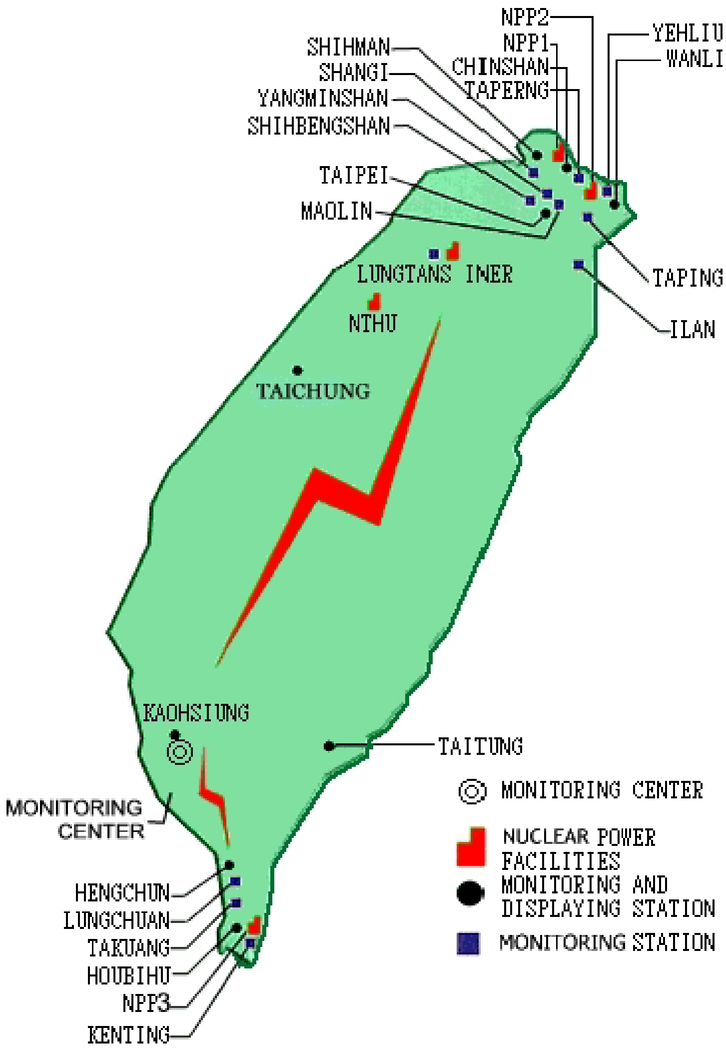


Figure 15.1 Environmental Radiation Monitoring Network

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 Acts, Regulations and Requirements for On-Site and Off-Site Emergency Preparedness

On the basis of Nuclear Reactor Facilities Regulation Act, the safety of nuclear installation is strictly regulated in the stage of design, construction and operation. Nevertheless, to assure the preparedness against very unlikely occurrence of accidents with large release of radioactive material, the Executive Yuan (the highest national administrative authority) promulgated a “Nuclear Accident Emergency Response Plan (NAERP)” in 1981 and revised it in 1994, 1998 and 2002. Nuclear Emergency Response Act was then promulgated on December 24, 2003. In order to carry out emergency response effectively for nuclear accident, following response centers shall be established according to the degree of possible impact of nuclear accident: the central Competent Authority (i.e., AEC) shall set up National Nuclear Emergency Response Center and Radiation Monitoring and Dose Assessment Center; the Ministry of National Defense shall set up Nuclear Emergency Supporting Center; the local Competent Authority shall set up Nuclear Accident Regional Hazard Response Center; the nuclear reactor facility licensee shall set up a dedicated Nuclear Emergency Response Unit and the Nuclear Emergency Response Organization within the Facility. The timing to set up the Nuclear Emergency Response Unit with dedicated responsibility and the Nuclear Emergency Response Organization within the Facility, and the relevant operational procedures and grouping are to be proposed by the nuclear reactor facility licensee and submitted to the central Competent Authority for approval.

The Nuclear Accident Emergency Response Plan (NAERP) covers the responsible organizations, accident categorizations, protective actions and recovery measures for emergency response to nuclear accidents. The major contents of this plan are summarized as follows.

(1) Emergency Response Organizations and Their Missions

In case of a nuclear accident, TPC is responsible for all the emergency response activities inside the plant while the National Nuclear Emergency Management Committee (NNEMC) assumes those activities outside. This Committee consists of heads from the following organizations: the AEC, Ministry of the Interior, Ministry of National Defense, Ministry of Economics Affairs, Ministry of Transportation and Communication, Council of Agriculture, Department of Health, Environment Protection Administration, Taiwan Provincial Government, Taipei Municipal Government, Kaohsiung Municipal Government, Taipei County Government, Pintung County Government, National Police Administration, Army General Headquarters, TPC, etc. Several temporary centers will be organized and under the command of the Committee if a nuclear accident occurred. These centers and their missions are described as follows.

The Near-Site Command and Coordination Center

This center is composed of specialists from AEC and TPC. The major missions are information collection, siren broadcasting, consequence prediction, radiation detection and dose assessment, etc.

The Crisis Directing Center

This center is composed of employees from local government. The major missions are the notification of protective actions to the public, accommodation of refugees, medical cares for the injured people and comforting of the affected inhabitants.

The Supporting Center

This center is composed of experts from military units. The major missions are establishing emergency communication network, decontamination of people, vehicles and environment, providing vehicles for the evacuation of local residents, and controlling the traffic and safeguarding the disaster area.

The Media Relations Office

This center is composed of personnel from AEC and TPC. The major missions are releasing of information associated with accident conditions and rescuing movements, clarification of false messages and notification of government announcements.

(2) Categorizations of Severe Nuclear Accidents

Not all the abnormal incidents occurred in a nuclear power station have a need for emergency response. Even 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 four categories according to their severity.

The First Category: Unusual Event

When a safety parameter slightly exceeds the limit defined in the Technical Specifications, it is recognized to be an unusual event. This kind of event usually involves such situation as reactor scram. According to the regulations, when a plant encounters an unusual event, AEC needs to be informed within 2 hours. While there is no release of radioactive material, there is no need to perform any protective actions for the public.

The Second Category: Alert

When an abnormal event in a nuclear power unit deteriorates to the extent that specific safety parameter(s) seriously exceeds the Technical Specifications limit and probably threatens the safe operation of the unit, it is defined as an Alert. Small amount of radioactive material 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.

The Third Category: Site Area Emergency

When a nuclear power unit encounters a major accident that causes sever damages to the safety systems and endangers the safety of the unit, a "Site Area Emergency" accident needs to be declared. In case of this kind of accident, TPC has to mobilize its whole in-house emergency response structure. At the same time AEC has to inform NNEMC to initiate such actions as assembly of manpower and materials for disaster rescuing movements. Protective actions for the offsite residents may also need to be activated during the accident.

The Forth 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 needs to be declared. In case of this kind of accident, all of the emergency response organizations need to be mobilized. The most proper protective actions for the offsite residents need to be performed immediately as well.

(3) Emergency Planning Zone

In case of a nuclear accident for which preliminary protective measures or evacuation of the local residents are required, how large the affected area should be is a question to be answered. Besides the public safety consideration, cost-effectiveness should be another vital factor. The philosophy and procedures adopted by advanced nuclear countries worldwide have been followed to establish the Emergency Planning Zone (EPZ) for the preparation of emergency responses for all nuclear accidents. The size of EPZ is closely related to the type of the reactor, the population density around the plant, the local topography, and the local weather conditions, etc. Taking the hypothetical meltdown of the reactor core as an example, three most important guidelines would be:

(a) The probability of the public receiving 100 m Sv whole body dose would be less than 3/100,000 per year.

(b) The probability of the public receiving 2 Sv whole body dose would be less than 3/1,000,000 per year.
(c) The probability of the death of the public caused by the nuclear power station accident is less than 1/1000 of the chance of death by human induced accidents per year.

The analysis of the accident dose and risk distribution for EPZ was made by applying the computer code CRAC2 or MACCS2 with the following inputs: the possibility of radiation release, the weather conditions, the population distribution around the plant, and other related parameters. Based on the above mentioned guidelines, methods, and the dose limits of "Nuclear Accident Protective Action Guide", the EPZ for the three operating nuclear plants was set as a circle with 5 km radius from the center of the nuclear power station following detailed analysis and integrated consideration.

Within the EPZ, all the preparedness must be ready for all times, including public information dissemination, gathering or pick-up spot, evacuation route and refugee center, etc. Drills should be conducted periodically to evaluate the feasibility of the preparedness and response arrangements, to see whether the staff reacts according to the procedures, to check the functions of relevant hardware and software, and finally whether nearby residents are used to the practice so as to enhance the efficiency and effectiveness of the NAERP.

(4) Protective Action Guide

In case of radioactive material release to the offsite areas of a nuclear power station, the decision on whether the offsite residents need to take shelter or to evacuate is based on the predicted radiation exposure. On the other hand, the decision of food edibility in the contaminated area is based on the equivalent concentration of iodine in the food. The relevant dose limits for protective actions set up by AEC are listed as follows.

16.2 Implementation of Emergency Preparedness

(1) Mobilization of Emergency Response Organizations

The mobilization of the emergency response organizations is base on the predicted whole body dose rate at plant site boundary. The sequences of mobilization are listed as follows:

- Below 20 μSv/hr: Partial mobilization of TPC emergency response personnel.
- At 20 μSv/hr: TPC sends out engineers to perform offsite radiation monitoring.
- At 50 μSv/hr: AEC and TPC send out personnel to establish the Near-Site Command and Coordination Center and to perform its designated missions.
- At 200 μSv/hr: The Supporting Center has to be established and to execute its missions.

- At 500 μSv/hr: NNEMC starts to take charge of all emergency response movements.

(2) Notification of the General Public

In case of a severe nuclear accident that may affect the residents in the EPZ, the Disaster Rescuing Center is responsible to provide the public correct and complete information. In general, the ways of notifying the public include radio, TV, broadcasting cars and emergency siren systems set up at police stations in the EPZ.

(3) Protective Actions for the General Public

In case of a severe nuclear accident that may affect the offsite environment, the protective actions for the residents in the EPZ include sheltering and evacuation. These actions are performed according to the criteria described in the subsection (4) of section 16.2, this report. Medicines for radiation dose reduction (e.g. potassium iodine) are prepared for all the evacuees. An acceptance center is established to accommodate the personnel evacuated from the EPZ. Personnel and vehicles need to be checked for radiation contamination before entering this center. De-contamination processes will be executed if necessary. The Disaster Rescuing Center is responsible for providing the evacuees water, food, medicines and other necessary assistance.

16.3 Training and Drills

(1) Training

To assure the knowledge and skill of emergency response for the personnel involved in the response actions, periodical training courses are held in each nuclear power station. The training areas include emergency operating 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 central government, local government and military agency who are responsible for 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 NNEMC. Special trainings on the decontamination of personnel, vehicles and roads are performed by the relevant military agency on regular basis as well.

In addition, introductory lectures for emergency response are given to the local residents in the area of EPZ before each off-site drill. 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, audio and video compact discs about emergency response are also dispersed in EPZ area associated with each nuclear power station every year. Besides, in order to promote the capability of medical care in case of nuclear accident, medical doctors selected from medical centers and hospitals nationwide have been sent to the United States to receive special training on therapy of radiation injury.

(2) Drills

To assure the effectiveness of emergency response actions, both on-site and off-site emergency response drills are held periodically. For on-site drill, once per year is required for each plant. The scenario of each drill is designed in TPC Headquarters and kept confidential beforehand. An evaluation group, consisting of scholars, government officials and civilian representatives, is organized to oversee the performance of the drill. Recommendations from this group are documented for TPC to follow.

For off-site emergency response, a full scope drill was held every two year before 2001. However, the frequency has been changed to once a year after 2002 as required by the government. The participating organizations in this kind of drill include all Ministries involved in NNEMC, TPC, the relevant local governments and military units. In addition, about one percent of the residents in the area of EPZ are invited to participate in the evacuation practice of each drill. The performance of each drill is assessed by a group of experts similar to the evaluation group for on-site drill. And the recommendations on further improvements are followed up by AEC.

16.4 International Framework and Relationship with Neighboring Countries

To promote the technologies of emergency preparedness and to enhance the capabilities of personnel involved in the activities, NNEMC actively engages in the cooperation with relevant international organizations. Several important activities in this area, completed or still ongoing, are described as follows.

(1) As a member of the Emergency Notification System of IAEA, regular communication tests between the AEC and the headquarters of IAEA have been performed for several years, and participation of the emergency drills held by IAEA were made several times.

(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 to have training in the areas of emergency medical care, assessment of EPZ, planning of emergency response, etc. On the other hand, many experts from governmental agency, national laboratories and utilities of the United States were invited here to exchange the information of emergency preparedness with local officers and engineers. In addition, participation in the international research projects organized by USNRC on severe nuclear accidents, such as plant analyzer and pipe integrity projects, were proved very fruitful.

(3) To cooperate with neighboring country in the area of emergency preparedness, a bilateral emergency support agreement on nuclear accidents has been signed by the AEC and Japanese Atomic Industrial Forum. A lot of activities, such as information exchange and exchange of experts and governmental officers, have been performed through this agreement.

16.5 Compensation for Nuclear Damage

The financial protection program for liability claims arising from nuclear accidents is described in section 11.4 of Article 11. However, some important requirements associated with compensation for nuclear damage are emphasized in this section. The Nuclear Damage Compensation Act is enacted according to Article 29 of the Atomic Energy Act with the latest version promulgated on May 14, 1997. This Act applies to the compensation for nuclear damage resulting from the peaceful uses of atomic energy. When a nuclear incident occurs in a nuclear installation or during transportation of nuclear materials, the operator thereof shall be liable for the compensation of the resulted nuclear 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 New Taiwan Dollars (NT$ 4,200,000,000). 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 transportation 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 Act, 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 incident. After the occurrence of a nuclear incident, AEC may organize a 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 incident 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 incident.

(4) 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 incident may seek compensation by way of a judicial proceeding, the court may take into account these reports.

Table 16.1 Dose Limits for Protective Actions of Sheltering and Evacuation

Dose Limit	Whole Body 5-50 mSv or Thyroid 50-500 mSv	Whole Body 50-100 mSv or Thyroid 500 mSv-1 Sv	Whole Body 0.1 Sv or Thyroid exceed 1 Sv
Protective Action	1. All residents need to take shelter inside a house. 2. If thyroid exceed 0.25 Sv, all residents need to take KI tablets. 3. Area control	1. Pregnant women and children need to take shelter inside a concrete house or to evacuate from EPZ. 2. Other adults need to take shelter inside a house. 3. Take KI tablets. 4. Area control.	1. All residents inside EPZ need to be evacuated. 2. Take KI tablets. 3. Area control.

Table 16.2 Food Control Standards in Contaminated Area

Food Spices	Drinking Water	Vegetables	Milk
Control Standard	3000 pCi/g	200 pCi/g	6000 pCi/g

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 Licensing Process and Regulatory Requirements

The siting requirements are mainly contained in the Enforcement Rules for the Implementation of Nuclear Reactor Facilities Regulation Act. According to Article 3 of the Enforcement Rules, the area surrounding the nuclear facility shall be divided into two regions based on the possible damage resulted from the design-basis nuclear accidents:

(1) Exclusion area: An exclusion area 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 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 required in Article 3 of the Enforcement Rules of the Nuclear Reactor Facilities Regulation Act, Article 4 of the 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 low population zone.

According to Article 4 of the Act, 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, to a newly establish school, works, jail, hospital, long term nursing institute, 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 competent authorities for the latter's in consultation with the government of municipality under the direct jurisdiction of the Executive Yuan, and the county (city) government for approval, so that the same can be implemented in accordance with the relevant laws/statutes and decrees.

Other major Codes and Standards for the site selections required by the country of origin (here referred to USA) are listed below:

- 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.

17.2 Assessments of Site Related Factors

17.2.1 Assessments of Seismic and Geological Aspects

Descriptions of the characteristics at and around the site shall be included in the applicant's safety analysis report. Basic geologic and seismic data, vibratory ground motion, surface faulting, stability of subsurface materials and slope stability are the major items to be included in the seismic and geological assessments. The assessments 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.

In selecting a site, relevant external initiating events shall also be included in the applicant's safety analysis report. Therefore, the structures, systems and components of safety systems shall be designed to withstand appropriate seismic forces. Seismic forces are evaluated from earthquakes of two different magnitudes: safe shutdown earthquake (SSE) and one-half of the safe shutdown earthquake (1/2 SSE). The SSE is defined as the maximum vibratory ground motion at the plant site that can reasonably be predicted from geologic and seismic evidence.

For the 1/2 SSE loading condition, the station is designed to be capable of continued safe operation. The design for the SSE is intended to assure that:

A. The integrity of the reactor coolant pressure boundary is not compromised;

B. The capability to shut down the reactor and maintain it in a safe condition is not compromised;

C. The capability to prevent or mitigate the consequences of accidents, which could 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.

17.2.2 Assessments of Nearby Industrial Facilities

The applicant should provide information on industrial and military facilities, 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.

17.2.3 Assessments of Meteorology and Natural Phenomena

The applicant is required to provide local meteorology and natural phenomena to the AEC for review. The following phenomena are to be taken into consideration: rain, floods, winds, typhoons, tidal effects, lightning, and thunderbolt. Investigation of the atmospheric diffusion characteristics and provision of the bounding atmospheric concentration ratio (X/Q) are required for evaluating radiological consequences of postulated design-basis accident to ensure that the limits described in Section 17.1 are not exceeded.

17.2.4 Assessments of Radiological Consequences

To evaluate the range of the exclusion area and the low population zone, an applicant should 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.

Besides the dose analyses necessary to support reactor siting, licensees are also required to evaluate 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. For example according to Article 8, if the licensee selects a new type of fuel, prior AEC approval for the safety analysis report associated with the proposed fuel design is required. As part of the accident analysis in SAR, it is required to assess the change in dose resulted from the design basis accidents such as large break LOCA, small break LOCA, and fuel handling accidents, etc., to ensure that this change still complies with the dose criteria.

17.2.5 Experiences and Examples

An example of site selection is presented below.

Originally there were four potential sites selected for the Lungmen nuclear project. Many factors have been considered in the selection, which could be classified into two categories -- most important factors and 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 import siting factors of the four candidates has been made and the results are shown in Table 17.1 with rating from 1 to 4. Beside 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 has been selected as the site of the Lungmen nuclear project as shown in Table 17.2.

17.3 Assessments of Environmental Impact

According to Article 5 of Environmental Impact Assessment Act, 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 EIA. During the planning stage, the project proponent 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 proponent 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 proponent, of these conclusions. The review period may be extended for another fifty (50) days under unusual circumstances. If the conclusions reached by the EPA show that the development activities are likely to have significant adverse impacts on the environment, the project proponent must conduct a Phase II EIA. According to Article 11 of Environmental Impact Assessment Act, the nuclear project proponent shall prepare a draft environmental impact assessment report [("Draft EIA Report")] and submit it to the AEC. The content of Draft EIA Report is 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. Inspection record, public meeting minutes and the draft EIA Report shall be submitted to the EPA.

EPA shall conclude its review within sixty (60) days and provide the conclusions to the AEC and the project proponent. The project proponent 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 of 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 (60) days under unusual circumstances.

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 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 plant incorporates design features for minimizing impact on environment.

Best = 4 Better = 3 Good = 2 Poor = 1

Table 17.3 Contents of Draft Environmental Impact Assessment Report

1. The name and business address or office address of the project proponent;
2. the name, residence or domicile and identification number of the representative of the project proponent;
3. the 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. the 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 proponent);
14. The budget for implementing environmental damage mitigation measures;
15. Summary of measures to prevent and mitigate adverse impacts to the environment caused by development activities; and
16. Bibliography of references.

ARTICLE 18. DESIGN AND CONSTRUCTION

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

(i) the design and construction of a nuclear installation provides for several reliable levels and methods of protection (defense in depth) against the release of radioactive material, with a view to preventing the occurrence of accidents and to mitigating their radiological consequences should they occur

(ii) the technologies incorporated in the design and construction of a nuclear installation are proven by experience or qualified by testing or analysis

(iii) the design of a nuclear installation allows for reliable, stable, and easily manageable operation, with specific consideration of human factors and the man-machine interface

18.1 Licensing Process and Regulatory Requirements

The licensing procedures for the design and construction of nuclear power stations are described in Section.9.2

The criteria for a construction permit of nuclear power stations are specified in the Enforcement rules for the Implementation of Nuclear Reactor Facilities Regulation Act, the Enforcement Rules as follows:

- Technical capability necessary for the construction of nuclear power stations shall be secured.
- The location, structures, and components of nuclear power stations shall conform to the technical requirements prescribed in the Enforcement Rules of the Act, relevant Regulations in such a way that there shall not be any impediment to the protection of human bodies, materials, and the public against radiation hazards caused by radioactive material or materials contaminated by them.
- There shall not be any impediment to the protection of the public health and the environment against danger or injury from radioactive material, which may accompany the lifetime operation of nuclear power stations.

The technical requirements for the location, structures, and components of the nuclear power station consist of 44 articles in the Enforcement Rules.

18.2 Implementation of Defense-in-depth Concept

In order to assure the safety of nuclear power stations, TPC applies a multi-barrier concept, based on the defense-in-depth principle, to the design, construction and operation of nuclear power stations. 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.

Irrespective of reactor types, the design of all systems, components, and structures of a nuclear power station should take into consideration the following internal and external events at the stage of siting, as specified in the Nuclear Reactor Facilities Regulation Act, its Enforcement rules and 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 station is designed by applying the defense-in-depth principle as a safety design concept against internal and external events as mentioned above. Its major contents are as follows:

- Sufficient safety margin is secured in the design so that the probability of any design basis accident is minimized. Safety systems are designed with independency, redundancy, and diversity so that the consequences of accidents are minimized.
- Plants are designed so that even if any abnormal condition occurs due to equipment failures, operator errors or combination thereof, the reactor protection system operates automatically after detecting the abnormal condition and initiates the reactor shutdown system to prevent the abnormal condition to proceed into a severe accident.
- Plants are designed with multiple barriers, such as the fuel pellet, the fuel clad, the reactor vessel and the reactor coolant pressure boundary, and the containment building to prevent the release of any radioactive material into the environment.

18.3 Prevention and Mitigation of Accidents

The followings are reflected in the design of TPC’s nuclear power stations to prevent any accident from occurring:

- 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 reactivity (i.e. negative power coefficient) . The reactor core is also designed to assure that power oscillations, which can result in conditions exceeding specified design limits, are not possible or can be readily suppressed.

- The reactor coolant pressure boundary is designed to have an extremely low probability of abnormal leakage and gross rupture. If any leakage of the reactor coolant system takes place, it is promptly detected to prevent against proceeding to a severe accident. It is also designed to allow periodic inspection and testing of the system to assess the structural integrity and leak-tightness.

- The emergency core cooling system (ECCS) is designed for automatic core cooling following any loss of reactor coolant at a rate such that any fuel damage that could interfere with continued effective core cooling is prevented. Even if the off-site power is lost, the necessary power of the ECCS system is to be supplied from emergency diesel generators installed in the nuclear power station. The residual heat removal system is designed to remove the core decay heat.

The reactor protection system is installed to detect accident conditions and to maintain the reactor at a safe state by automatically initiating the reactor shutdown system and the engineered safety features. The reactor protection system is designed with redundancy, diversity, and independence to assure that no single failure of any equipment or channel of the system results in the loss of the intended safety functions.

The followings are incorporated into the design of nuclear power stations 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 station, 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). Safety parameter display system (SPDS) is installed in the following locations: main control room of the plant, the TSC, and the TPC Headquarters, so that major safety parameters are promptly recognized.

The main control room is designed so that even if any serious accident occurs, the operators can safely remain inside to take the necessary post-accident actions. It is possible in the control room to monitor the operating parameters, the radioactivity inside and outside the reactor containment, the radioactive material releasing passage, and the radioactivity around the nuclear power station in order to get hold of the accident conditions and to take appropriate actions.

18.4 Application of Proven Technologies

TPC requires that technologies incorporated in a design shall be duly proven by experience or qualified by testing or analysis. All nuclear power stations in Taiwan were designed, as elaborated in the bid specifications, with technologies proven by operating experiences inside or outside this country and also licensable in the export country.

18.5 Operation in Consideration of Human Factors and Man-Machine Interface

The Nuclear Reactor Facilities Regulation Act, its Enforcement Rules and Regulations stipulates that the main control room, the safety parameter display system, and the remote control room shall be designed so that the results of analyzing and evaluating the human factors are reflected therein in order to maximize the safety and efficiency of nuclear power stations. According to this provision, the analysis for the feasibility and suitability of the human engineering design are included in the preliminary safety analysis report（PSAR）and in the final safety analysis report（FSAR）. The major contents of the analysis are as follows:

- In the design of the main control room, human factors are considered so that the man-machine interface is suitable for the safe operation of nuclear power stations. The major factors are: working space and the environment around it, alarm and control facility, visual indicating facility, auditory signal facility, nameplates and their positioning, and layout of distributing boards.

- In the design of the safety parameter display system, the human engineering principle is considered so that the system continuously provides important safety information and the reactor operators can easily recognize them from designed location.

- The remote control room is designed in consideration of man-machine interface so that the reactor can be safely shutdown.

From TMI accident it showed that operator performance is crucial to system safety. Human error is one of the factors that affect human performance. Currently, human error mitigation is being considered in the design of Human System Interface of the main control room (MCR) for nuclear power station as follows:

1. Eliminating affordability of errors in the design phase

2. Intelligent decision support systems including the training program improvement

3. Memory aids for maintenance personnel, e.g., portable interactive maintenance

auxiliary

4. Training of error management
5. Ecological interface design.

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

To obtain AEC’s approval to construct or operate a nuclear power station, an applicant must submit safety analysis reports for review. Article 14 of the report describes the safety reports, and the AEC’s reviews and audits that apply to the issuance of an operating license.

19.2 Operational Limits and Technical Specifications

Article 6 of the Nuclear Reactor Facilities Regulation Act requires that the licensee shall state and submit an application, enclosed with the FSAR, the summary on the corrective actions following inspection findings during the construction as well as the approved pre-operation test reports on all the system to the AEC for the approval of initial fuel loading. The technical specifications, being part of the final safety analysis report, are established to ensure safe operation of the nuclear reactor. According to the working guidelines issued by the AEC, the contents of the technical specifications include the safety limits, the limiting conditions for operation, the limiting safety system settings, the surveillance requirements, the design features of the facility, and the administration management. The technical specifications are the important bases for both the operational safety and the surveillance test of a nuclear power station. Because the Technical Specifications were completed before the nuclear power station began to operate, timely revision of Technical Specifications may be required along with the operation of the plant. According to the Administrative Regulations of the design amendment and equipment change of the Nuclear Reactor Facilities, neither of the design amendment nor equipment change, if involved in the revision of the technical specifications, shall be make without the AEC's approval. The AEC encourages licensees to use the Improved Technical Specifications (ITS) as the basis for plant-specific technical specifications. A more detailed description of the Technical Specifications updates and ITS implementation is provided in section 6.4 of Article 6 of this report.

19.3 Operation, Maintenance, Inspection, and Testing Procedures

This section describes the operation and maintenance procedures used by the licensee to ensure plant safety under routine operating, abnormal, and emergency conditions. As required by the quality assurance plan, any activities that affect quality must be documented in appropriate instructions, procedures or drawings. These activities include operation, maintenance, inspection and testing of the facility.

Operation and maintenance procedures are divided into the following categories:

A. General Plant Operating Procedures

   These procedures describe the following steps: bringing the plant from cold shutdown or hot standby condition to power operations, load changing, and finally bringing the plant back to hot standby or cold shutdown condition.

B. System Operating Procedures

   These procedures describe the steps required to take the individual system into or out of service. These procedures also include the manipulation of the system for several normal conditions as required.

C. Operating Procedures for Instrumentations

   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.

D. Operating Procedures for Abnormal Events

These procedures describe the instructions for the operators to respond for abnormal system conditions.

E. 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 alarm system, it is designed to give visual (light) and audible (sound) alarms for each window. The visible alarms are classified into two categories: "Red" for trips and "White" for alerts. Every visual alarm is initiated by unique protective system and accompanied by a high frequency buzz noise alarm to remind the operator for action. When alarm is cleared the annunciator system acknowledges with a low frequency buzz.

F. Emergency Procedures

These procedures provide instructions for the operators to handle plant emergency situations such as:

- Earthquake
- Typhoon
- Loss of all feed water
- Loss of coolant, etc.

A more detailed description of emergency procedure will be given in the following section (section 19.4).

G. Temporary Procedures

These procedures are to provide detailed instructions for specific tests or operations of the safety related systems.

The schedule for the surveillance tests of the safety related systems will be established in accordance with the surveillance requirements in the Technical Specifications.

The operating procedures related to the safety of nuclear installations are to be deliberated by the Safety Operation Review Committee (SORC) and implemented after the plant superintendent's approval. The operational technical specifications require that the same process shall apply in case of any changes to the approved procedures.

19.4 Procedures for Responding to Anticipated Operational Occurrences and Accidents

In the FSAR, transients and accidents are analyzed based on the single-failure criterion. The single-failure criterion is considered to be not appropriate for 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 failures of equipment or operator errors. Examples of multiple failure events include:

(1) Multiple tube ruptures in a single steam generator and tube rupture in more than one steam generator.

(2) Failure of 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, stuck-open relief valve or safety/relief valve, or loss of main feedwater.

(5) Operator errors of negligence.

Symptom-oriented EOPs have been developed and implemented in all the three nuclear power stations after the TMI accidents. Based on generic Emergency Procedure Guidelines (EPGs) provided by the reactor vendors, detailed emergency operating procedures were developed by the TPC. Differences between EOP and EPG have been properly documented and justified. The resultant emergency operating procedures shall comply with the requirements of NUREG-0737, Item I.C.1. To ensure that proper procedures had been developed, the TPC performed the verification and validation of EOPs. In addition, simulators have also been used to make sure that EOPs can be properly simulated. The verification and validation program of EOP as well as the EOP itself were then reviewed and approved by the AEC.

19.5 Engineering and Technical Support

The support for the plant operating organization, i.e., TPC, is available from the initial testing program through the lifetime of the plant in accordance with the following special capabilities:

A. Nuclear, mechanical, structural, electrical, thermal hydraulic, metallurgy and materials, instrument, and controls engineering supports were provided by Nuclear Operation Department (NOD) of TPC, other TPC plants, and the Institute of Nuclear Energy Research (INER).

B. Plant chemistry and health physics support was provided by NOD/TPC and INER.

C. Fueling and refueling operation support was provided by other TPC plants, NOD/TPC, the Department of power equipment repair and maintenance of TPC, and INER.

D. Maintenance support was provided by the Department of power equipment repair and maintenance in TPC or INER.

The TPC has also retained consultants to provide technical services on subjects associated with plant safety and plant operation.

In addition to the direct and daily regulatory activities provided by the AEC, research and development programs have been performed by INER to ensure that adequate engineering and technical supports are available throughout the lifetime of a nuclear installation. INER, under the supervision of the AEC, has been established over 30 years and is the only domestic and specialized nuclear energy R&D institution. The current research and development are focused on the following areas: nuclear engineering technology, nuclear safety, radiation protection & environmental monitoring, radioactive waste treatment, and civilian nuclear applications. INER has conducted many tasks in supporting the operation of nuclear facilities. Some of the examples performed in the Year 2000 to 2004 were: upgrading the Maanshan training simulator for Mid-loop operation simulation, development of remotely controlled weld overlay repair technique, development of LOCA licensing calculation capability, establishment and application of TPC's integrated risk monitor, preliminary study of severe accident management for TPC NPSs, development and application of safety analysis techniques for special transients in the nuclear power stations, promotion of ionization radiation key comparisons in Asia-Pacific region, studies on the wet oxidation of spent ion exchange resin and the solidification of residue generated, commissioning the facility of plasma vitrification for low-level radioactive waste, etc.

19.6 Incidents Reporting

19.6.1 Regulatory Requirements

Requirements for incident reporting are specified in the technical guideline "Immediate Notification of Abnormal Events for Operating Nuclear Power Reactors," issued by the AEC. The licensee for a nuclear power station shall report any abnormal events described in the following conditions within 2 hours after the discovery of the event.

1. Violation of the safety limits in the Technical Specifications.

2. Initiation of power reduction or shutdown as required by the limiting condition of operation in the plant's Technical Specifications.

3. Any event resulted in the condition that may affect the safety of nuclear power station:

   (a) In a condition not analyzed in the safety analysis report;

   (b) In a condition outside the design basis of the plant;

(c) In a condition not covered by the plant's operating and emergency procedures.

4. Any natural phenomenon or external condition that posed an actual threat to the safety of the nuclear power station or significantly hampered site personnel from the activities necessary for the safe operation of the nuclear power station.

5. Any event or condition, which was not part of the planned sequence, that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF) or the Reactor Protection System (RPS).

6. Any event that results in a major loss of emergency assessment capability, offsite response capability, or offsite communications capability.

7. Any event or condition that alone could prevent the fulfillment of the safety function of structures or systems that are needed to:

   (a) Shut down the reactor and maintain in a safe shutdown condition;

   (b) Remove residual heat;

   (c) Control the release of radioactive material; or

   (d) Mitigate the consequences of an accident.

8. (a) Any airborne radioactivity release in the monitored region or the unrestricted region exceeded the alert set point for those regions.

   (b) Any radioactive nuclide concentration in liquid effluents released exceeds the limits in Line 8 of Table 4 of Safety Standards for Protection against Ionizing Radiation, or sums of total curies in one release event (not including tritium and dissolved noble gas) exceeds 3.7×10^9 Bq (0.1 Ci), or quarterly sums of total curies released (not including tritium and dissolved noble gas) exceeds 9.25×10^{10} Bq (2.5 Ci).

9. Any event requiring the transport of radioactively contaminated persons to offsite medical facility for treatment.

10. Any event that is related to the threat of human health, environmental protection, and the feeling of the general public, such as:

    - A unplanned scram, shutdown, or load decoupling.
    - A power reduction of more than 20%, continued for 4 hours due to component failure.
    - Any industry related event occurred onsite that results in human injury or death and requires the transport of a person to an offsite medical facility for treatment.
    - Any plant employee received dose higher than the dose limits in the Radiation Protection Safety Standard.

- Occurrence of any huge sound, smoke, natural hazard, accident onsite or at the neighboring region may result in a concern of the general public.
- Any security related event such as sabotage.
- Any dispute between the employees of nuclear facility and the general public, or a demonstration held by the general public.
- Removal of radioactive material to offsite that violates the radioactive material control requirements.
- A handling accident of radioactive material waiting for treatment, fuel or reactor internal component, occurred during the onsite handling and transporting process.
- Any nuclear fuel, radiation source, or radioactive material waiting for treatment that was lost, stolen or damaged.
- A water hammer phenomenon that results in component damage or impact to the function of the affected system.

19.6.2 Evaluation of Nuclear Power Station Reactor Scram Events

Within two hours of the occurrence of a reactor scram, The TPC must report to the Department of Nuclear Regulation of the AEC about the consequences of the scram and probable causes. If the cause for the scram is unclear or it is with possible safety concerns, the restart of the unit will be under rigorous control. The unit will be allowed to restart only if the root cause is cleared or a satisfactory safety assessment is completed.

19.6.3 Evaluation of Nuclear Power Station Abnormal Occurrence and Equipment Malfunctions

If any abnormal event occurs at nuclear power station that is required to report as specified in the Technical Specifications, a detailed report of the situation, the corrective actions and the measures to prevent recurrence must be submitted to the AEC within 30 days. AEC will review the 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 of TPC's nuclear units during the period of 1988 to 2003 is shown in Table 19.1.

Similar process is applied to the malfunction of major equipment. To assure the safe operation of nuclear power stations, whenever there is a malfunction of any major equipment, AEC will immediately dispatch personnel to the site for on-site inspection, detailed review of TPC's analysis of the root cause, and asking for further improvements if necessary.

19.7 Programs to Collect and Analyze Operating Experience

19.7.1 Regulatory Information Study and Experience Feedback

With the assistance of the INER, a program has been established by the AEC to regularly collect and analyze plant operating experiences since 1993. This includes the collection of generic communications from the USNRC, such as regulatory issue summaries, generic letters, bulletins, and information notices and the abnormal events from both Japan and France. On the other hand, the TPC obtains operating experiences from General Electric Service Information Letter, Westinghouse Technical Bulletins, BWROG, WOG, INPO/WANO Networks and NRC bulletin. Feedbacks of foreign operating experiences 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 required by the AEC as a result of the experience feedback. For example, ASME code does not require in service inspection of the BWR jet pump beam and shroud head bolts, but they were performed in EOC cycle 18 refueling outage of Chinshan plant due to the fact that cracks have been found in foreign plants.

19.7.2 Establishment of a System for the Feedback of Operating and Maintenance Experiences

To share important operating and maintenance experiences among different plants, TPC worked out a program, the Operation Experience (OE) program, which can be applied to all nuclear installations. The detailed description of OE program is shown in Article 10 of this report.

The station operating procedures of a plant have been developed to ensure that plant operating personnel is kept informed of the pertinent improvement information on plant operation. In addition, steps have been taken to ensure that this information is continually factored into training programs. For example, Maanshan station's operating procedures (SOP 109) have been developed to comply with the requirements of operating experience feedback to plant staff.

19.8 Radioactive Waste

The Nuclear Material and Radioactive Waste Management Act was enacted on December 25, 2002, which replaced all administrative orders enforced upon licensees in the past decades. For radioactive wastes, the Act sets regulatory requirements for all licensing and enforcement activities on their treatment and storage as well as repository construction, operation, closure, decommissioning and institutional control. The AEC with its subsidiary agency, FCMA, is the regulatory authority. Pursuant to the Act, TPC has submitted the management plan for LLRW and will submit HLW management plan by the end of 2004.

Low-Level Radioactive Waste

AEC's LLRW management strategies are to reduce the waste volume, renovate the waste treatment technology, ensure the safety of storage and actively promote the final disposal program. Till November 2003, 182,440 drums (55-gallon each) of low level radioactive waste (LLRW) are stored in Taiwan. More than 90 percent of the LLRW was produced by the three nuclear power stations, while hospitals, research institutes and industry wastes accounted for the remaining. The Lanyu storage facility, located on an offshore islet Lanyu, provides interim storage for solidified LLRW since 1982. The facility, designed to store 98,172 drums of LLRW in 23 semi-underground engineered trenches, reached its full capacity in 1996. New storage facilities have been constructed at each nuclear power station to accommodate the newly generated LLRW.

A new Act on site selection for the LLRW repository has been drafted and submitted to the Legislative Yuan for approval. According to the bill, the Ministry of Economic Affairs (and TPC) is responsible for selecting an LLRW disposal site within five years after the Act is enacted. Then it shall be submitted to the Executive Yuan for approval. The Enforcement of the Act, when passed, is expected to expedite the ongoing siting process.

Spent Nuclear Fuel

Till December 2003, 11,929 spent fuel assemblies (2,427 MTU equivalent) were discharged from three NPSs in Taiwan. As for the spent fuel management, on-site dry storage is recognized as a favorable option before implementing final disposal. Commissioning of the storage facilities at Chinshan and Kuosheng plants are anticipated in 2008 and 2009, respectively. On the other hand a long-term investigation plan is being carried out by TPC to select a suitable site with geological formations of preferred characteristics for a final repository of spent nuclear fuel. Preliminary results submitted by TPC to the AEC show that some potential host rocks in certain regions of Taiwan are worth further investigation.

19.9 The transparency of Nuclear Information

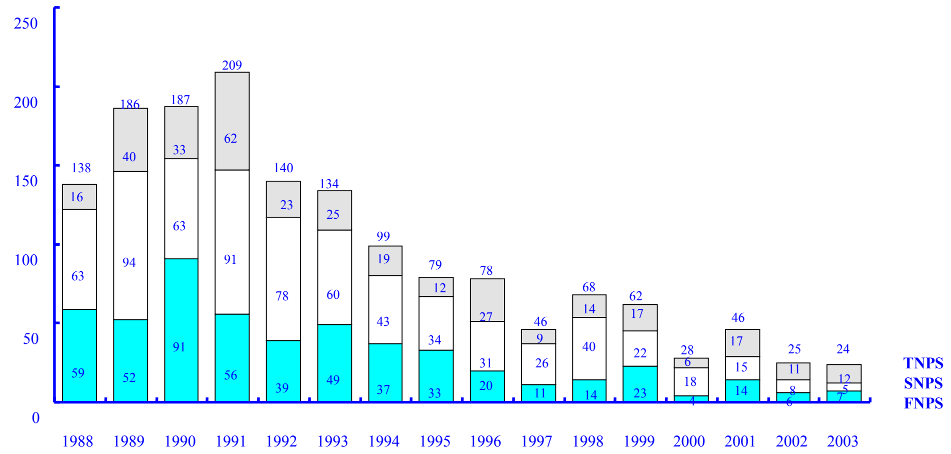
To make the public aware of the fact that nuclear power is an indispensable source of energy in Taiwan and to provide the public with the most updated information on nuclear operations, TPC has made considerable efforts to demonstrate its sincerity and openness in the following manners:

- The public has been well informed of all nuclear events at all times through the release of countrywide newsletters. The local residents surrounding nuclear power stations have been given the first priority to access the most updated nuclear information through media or special/weekly reports. As more details become available, it will be put on TPC's Web site or through news media.

- The public can retrieve the nuclear operations information, including plant status and event reports, by logging into TPC Web site.
- TPC has issued newsletters/bulletins and distributes them to local residents, in which status including plant construction/operation, radwaste disposal, and environmental protection, etc. is explained.
- Toll-free service telephones are provided to the public for enquiries.
- The nuclear power station has invited reporters to visit the plant site for the understanding of the good working practice during plant outage. The plant manager answers the enquiries himself.
- The nuclear power stations host two-way communication meeting with residents committee or community groups around the sites.
- Delegates of the nuclear power stations have been sent to participate in the community activities.
- Informative materials have been prepared and released according to the requests of legislators, reporters, scholars and experts, antinuclear groups, students and even the general public.

Table 19.1 Number of Reportable Events of TPC's Nuclear Units

	Chinshan	Kuosheng	Maanshan	Total
1988	59	63	16	138
1989	52	94	40	186
1990	91	63	33	187
1991	56	91	62	209
1992	39	78	23	140
1993	49	60	25	134
1994	37	43	19	99
1995	33	34	12	79
1996	20	31	27	78
1997	11	26	9	46
1998	14	40	14	68
1999	23	22	17	62
2000	4	18	6	28
2001	14	15	17	46
2002	6	8	11	25
2003	7	5	12	24



Note:

FNPS means First Nuclear Power Station (Chinshan NPS)

SNPS means Second Nuclear Power Station (Kuosheng NPS)

TNPS means Third Nuclear Power Station (Maanshan NPS)

Figure 19.1 Number of Reportable Events of the TPC Nuclear Power Stations

APPENDIX A：ABBREVIATIONS

AEC	Atomic Energy Council/Executive Yuan
ALARA	As Low As Reasonably Achievable
ASME	The American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
CNLA	Chinese National Laboratory Accreditation
CRIEPI	Central Research Institute of Electric Power Industry
DGBAS	Directorate General of Budget, Accounting and Statistics
CLA	Council of Labor Affairs
DNE	Department of Nuclear Engineering/TPC
DNS	Department of Nuclear Safety/TPC
DOH	Department of Health/Executive Yuan
DONG	Department of Nuclear Generation/TPC
DRPI	Digital Rod Position Indicator
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EIA	Environmental Impact Assessment
EIS	Environmental Impact Statement
EPA	Environmental Protection Administration/Executive Yuan
EPRI	Electric Power Research Institute/USA
ERF	Emergency Response Facility
ESF	Engineered Safety Feature
FCMA	Fuel Cycle and Materials Administration/AEC
FSAR	Final Safety Analysis Report
GE	General Electric Company
HSI	Human-System Interface
IAEA	International Atomic Energy Agency/United Nations
IEEE	The Institute of Electrical and Electronics Engineers
INER	Institute of Nuclear Energy Research/AEC
INPO	Institute of Nuclear Power Operations
INSAG	International Nuclear Safety Advisory Group
IRA	Integrated Reliability Analysis
ITRI	Industrial Technology Research Institute
ITS	Improved Technical Specifications
LCM	Life Cycle Management

LCO	Limiting Condition for Operation
LERs	Licensee Event Reports
LNPS	Lungmen Nuclear Power station/TPC
MMCS	Maintenance Management Computerization System
MOEA	Ministry of Economic Affairs/Executive Yuan
MOFA	Ministry of Foreign Affairs/Executive Yuan
NAERP	Nuclear Accident Emergency Response Plan
NBMD	Nuclear Backend Management Department/TPC
NFA	National Fire Fighting Administration/Ministry of the Interior//Executive Yuan
NDCORE	National Database Center of Occupational Radiation Exposures
NNEMC	National Nuclear Emergency Management Committee
NORM	Naturally Occurring Radioactive Material
NPS	Nuclear Power Station
NSC	Nuclear Safety Committee
NSSS	Nuclear Steam Supply Systems
NTHU	National Tsin-Hwa University
OSC	Operating Support Center
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
PWRs	Pressurized Water Reactors
QA	Quality Assurance
RCCA	Rod cluster Control Assembly
RCS	Reactor Coolant System
RMC	Radiation Monitoring Center
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SAT	Systematic Approach to Training
SER	Significant Event Report
SOER	Significant Operating Experience Report
SORC	Station Operation Review Committee
SPDS	Safety Parameter Display System
SSC	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
TEPCO	Tokyo Electric Power Company/Japan

TPC	Taiwan Power Company
TS	Technical Specifications
TSC	Technical Support Center
USNRC	United States Nuclear Regulatory Commission
W	Westinghouse
WANO	World Association of Nuclear Operators

APPENDIX B：ACKNOWLEDGMENTS

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Annex 1

Major Design Characteristics
of
Nuclear Power Stations in Taiwan

MAJOR DESIGN CHARACTERISTICS OF CHINSHAN PLANT

SYSTEMS	Units 1 and 2
THERMAL AND HYDRAULIC DESIGN	
Rated Power, MWt	1,775
Design Power, MWt	1,864
Steam Flow Rate, lb/hr	7.62×10^6
Core Coolant Flow Rate, lb/hr	53×10^6
Feedwater Flow Rate, lb/hr	7.597×10^6
Feedwater Temperature, ^{O}F	420
System Pressure, Nominal in Steam Dome, psia	1,020
Average Power Density, KW/liter	50.5
Maximum Thermal Output, KW/ft	13.4
Average Thermal Output, KW/ft	5.449
Maximum Heat Flux, Btu/hr-ft^2	349,900
Average Heat Flux, Btu/hr-ft^2	144,000
Maximum UO_2 Temperature, ^{O}F	3,300
Average Volumetric Fuel Temperature, ^{O}F	1,080
Average Fuel Rod Surface Temperature, ^{O}F	565
Minimum Critical Heat Flux Ratio (MCHFR)	≥1.9
Coolant Enthalpy at Core Inlet, Btu/lb	525.6
Core Maximum Exit Voids Within Assemblies, %	76
Core Average Exit Quality, % Steam	14.3
Design Power Peaking Factor	
Maximum Relative Assembly Power	1.4
Local Peaking Factor	1.24

Axial Peaking Factor	1.4
Total Peaking Factor	2.43
<u>NUCLEAR DESIGN (First Core)</u>	
Water / UO_2 Volume Ratio (Cold)	2.59
Reactivity with Strongest Control Rod Out, keff	<0.99
Moderator Void Coefficient	
Hot, no voids, Δk/k - % Void	0.6×10^{-3}
At Rated Output, Δk/k - % Void	1.2×10^{-3}
Fuel Temperature Doppler Coefficient	
At 68 ^{O}F, Δk/k –^{O}F Fuel	1.3×10^{-5}
Hot, No Voids, Δk/k –^{O}F Fuel	1.2×10^{-5}
At Rated Output, Δk/k –^{O}F Fuel	1.3×10^{-5}
Initial Average U-235 Enrichment, W/O	1.90%
Fuel Average Discharge Exposure, Initial Fuel Load MWD/Ton	14,800
<u>CORE MECHANICAL DESIGN</u>	
<u>Fuel Assembly</u> (initial core)	
Number of Fuel Assemblies	408
Fuel Rod Array	8×8
Overall Dimensions, inches	175.88
Weight of UO_2 per Assembly, pounds	465.28
Weight of Fuel Assembly, pounds	676
Number per Fuel Assembly	63

Outside Diameter, inch	0.493
Clad Thickness, inch	0.034
Gap-Pellet to Clad-diametric, inch	0.009
Length of Gas Plenum, inches	14
Clad Material	Zircaloy-2
Fuel Pellets	
Material	Uranium Dioxide
Density, % of theoretical	95%
Diameter, inch	0.416
Length, inch	0.5
Fuel Channel	
Overall Dimension, inches (length)	167.1
Thickness, inch	0.100
Cross Section Dimensions, inches	5.438×5.438
Material	Zircaloy-4
Core Assembly	
Fuel Weight as UO2, pounds	189,880
Zirconium Weight, pounds (Z-2+Z-4 Spacers)	≈78,870
Core Diameter (equivalent), inches	136.8
Core Height (Active Fuel), inches	146
Reactor Control System	
Method of Variation of Reactor Power	Movable Control Rods & Variable

	Coolant Pumping
Number of Movable Control Rods	87
Shape of Movable Control Rods	Cruciform
Pitch of Movable Control Rods, inches	12.0
Control Material in Movable Rods	B_4C granules Compacted in SS Tubes
Type of Control Rod Drives	Bottom entry, Locking Piston
CONTAINMENT	
Type	Pressure Suppression
Leak Rate	
(% vol/day)	0.5
Containment	
Construction	Mark I, Steel Dry-Well and Suppression Pool
Drywell	
Construction	Light bulb shape;
	Steel Vessel
Internal design temperature (^oF)	340^o
Max. internal pressure (psig)	56
Free (air) volume total (cu ft)	130,000
Suppression Pool	
Construction	Torus, Steel Vessel
Internal design temperature (^oF)	340^o
Internal design Pressure (psig)	56

Water volume (cu ft)	78,000
Free (air) volume total (cu ft)	87,200

MAJOR DESIGN CHARACTERISTICS OF KUOSHENG PLANT

SYSTEMS (BWR/6, 218-264)	**Units 1 & 2**
THERMAL AND HYDRAULIC DESIGN	
Rated power (MWt)	2894
Design power (MWt) (ECCS design basis)	3039
Steam flowrate (millions lb/hr)	12.453
Core coolant flowrate (millions lb/hr)	84.5
Feedwater flowrate (millions lb/hr)	12.426
System pressure, nominal in steam dome (psia)	1040
Average power density (kw/liter)	52.4
Maximum linear heat generation rate (kw/ft)	13.40
Maximum heat flux (Btu/hr-sq ft)	361,600
Maximum UO_2 temperature (F)	3,435
Average volumetric fuel temperature (F)	1,100
Average cladding surface temperature (F)	560
Minimum critical power ratio (MCPR)	1.20
Coolant enthalpy at core inlet (Btu/lb)	527.8
Core maximum exit voids within assemblies (%)	76
Core average exit quality (% steam)	14.95
Feedwater temperature (F)	420
Maximum relative assembly power	1.40
Axial peaking factor	1.40
NUCLEAR DESIGN (First Core)	
Water/UO_2 volume ration (cold)	2.70
Reactivity with strongest control rod out (keff)	<0.99
Dynamic void coefficient (¢/%)	-7.40
Core average voids (%)	41.3
Fuel temperature doppler coefficient (¢/%)	-0.410
Initial average U-235 enrichment (wt%)	1.88
Initial cycle exposure (MWd/short ton)	9004
CORE MECHANICAL DESIGN (initial core)	
Fuel Assembly	
Number of fuel assemblies	624
Fuel rod array	8×8
Overall length (in.)	176
Weight of UO2 per assembly, (pellet type) (lb)	456
Weight of fuel assembly (lb)	697
Fuel Rods	

Number per assembly	62
Outside diameter (in.)	0.483
Cladding thickness (in)	0.032
Diametrical gap, pellet to cladding (in.)	0.009
Length of gas plenum (in.)	9.48
Cladding material (a)	Zircaloy-2
Fuel pellets	
Material	UO_2
Density (% of theoretical)	95
Diameter (in.)	0.410
Length (in.)	0.410
Fuel Channel	
Overall dimensions, length (in.)	167.4
Thickness (in.)	0.120
Cross-section dimensions (outside-in.)	5.45×5.45
Material	Zircaloy-4
Core Assembly	
Fuel weight as UO2 (lb)	284,988
Core diameter (equivalent-in.)	160.2
Core height (active fuel-in.)	150
Reactor Control System	
Method of Variation of reactor power	Movable control rods and variable forced coolant flow
Number of movable control rods	145
Shape of movable control rods	Cruciform
Pitch of movable control rods	12.0
Control material in movable rods	B_4C granules compacted in SS tubes
Type of control rod drives	Bottom entry；locking piston
Type of temporary reactivity control for initial core	Burnable poison；gadolinia-urania fuel rods
CONTAINMENT Type	Mark III. Reinforced concrete containment, as for PWR plants, but with pressure suppression. Reactor building encloses drywall and suppression pool.

Leak rate (%/day)	0.1
Reactor building construction	Reinforced concrete cylindrical structure (not pre-stressed) with hemispherical head；steel lined.
Internal design temperature (^{o}F)	200
Design pressure (psig)	15
Free (air) volume (cu ft)	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
Free (air) volume (cu ft)	238,00
Suppression Pool[(a)]	
Construction	Reinforced concrete, steel lined. Cylindrical.
Internal design temperature (F)	200
Design pressure (psig)	15
Water Volume at High-Water Level (cu ft)	113,950
Break Area/Total Vent	0.008

MAJOR DESIGN CHARACTERISTICS OF MAANSHAN PLANT

SYSTEMS	Units 1 & 2
CORE MECHANICAL DESIGN	
Fuel assemblies	
Design	RCC canless
Number of fuel assemblies	157
UO_2 rods per assembly	264
Rod pitch, in.	0.496
Overall dimensions, in.	8.426×8.426
Fuel weight (as UO_2), lb	164,900
Clad weight, lb	33,746
Number of grids per assembly	2-Type R 6-Type 2
Loading technique	3 region nonuniform
Fuel rods	
Number	41,448
Outside diameter, in.	0.360
Diametral gap, in.	0.0062
Clad thickness, in.	0.0225
Clad material	Zircaloy-4
Fuel pellets	
Material	UO_2 sintered
Density, % of theoretical	95
Diameter, in.	0.3088
Length, in.	0.507
Rod cluster control assemblies	
Neutron absorber	B_4C (with H_f or Ag-In-Cd tips)
Cladding Material	Type 304 SS-cold worked
Clad thickness, in.	0.0185
Number of RCC assemblies	52
Number of absorber rods per cluster	24
Core structure	
Core barrel, ID/OD, in.	133.85/137.875
Thermal shield	Neutron pad design
Structure characteristics	
Core diameter, in.(equivalent)	119.7
Core height, in. (active fuel)	144
Reflector thickness and composition	
Top-water plus steel, in.	~10

Bottom-water plus steel, in.	~10
Side-water plus steel, in.	~15
H_2O/U molecular ratio core, lattice (cold)	2.73
<u>Feed enrichment, w/o</u>	
Region 1	1.6
Region 2	2.4
Region 3	3.1
THERMAL AND HYDRAULIC DESIGN	
Reactor core heat output, MWt	2,775
Reactor core heat output, 10^6 Btu/h	9,471
Heat generated in fuel, %	97.4
System pressure, nominal, psia	2,280
System pressure, minimum steady state, psia	2,250
<u>Minimum DNBR for design</u>	
Typical flow channel transients	≥1.49
Thimble flow channel	≥1.47
<u>Coolant flow</u>	
Total thermal flow rate, 10^6 Btu/h	111.9
Effective flow rate for heat transfer, 10^6 Btu/h	106.7
Effective flow rate for heat transfer, ft^2	44.0
Average velocity among fuel rods, ft/s	15.3
Average mass velocity, 10^6 $lb_m/^{h\text{-}ft}$	2.42
Coolant temperature, ^0F Nominal inlet	557.9
Average rise in vessel	61.4
Average rise in core	64.0
Average in core	591.7
Average in vessel	590.2
<u>Heat transfer</u>	
Active heat transfer, surface area, ft^2	46,800
Average heat flux, Btu/h-ft^2	197,200
Maximum heat flux for normal operation, Btu/h-ft^2	457,500
Average thermal output, kw/ft	5.44
Maximum thermal output for normal	12.6

operation, kw/ft	
Peak linear power resulting from overpower transients/operator errors (assuming a maximum overpower of 118% power), kw/ft	18.0
Heat flux hot channel factor, F_Q	2.30
Peak at peak linear power for prevention of center-line melt ^0F	4,700
RCS DESIGN AND OPERATING PARAMETERS	
Plant design life, yr	40
Nominal operating pressure, psig	2,235
Total system volume including pressurizer and surge line, ft^3	9,410
System liquid volume, including pressurizer water at maximum guaranteed power, ft^3	8,833
Pressurizer spray rate, maximum gal/min	700
Pressurizer heater capacity, kw	1,400
Pressurizer relief tank volume, ft^3	1,300
RCS THERMAL AND HYDRAULIC DATA	
NSSS power, MWt	2,785
<u>Thermal design flows, gal/min</u>	
Active loop	97,600
Idle loop	
Reactor	292,800
Total reactor flow, 10^6 lb/h	109.1
<u>Temperatures, ^0F</u>	
Reactor vessel outlet	619.9
Reactor vessel inlet	557.0
Steam generator outlet	556.7
Steam generator steam	540.2
Feedwater	440.0
Steam pressure, psia	964

Total steam flow, 10^6 lb/h	12.3
CONTAINMENT TYPE	Steel-lined pre-stressed post-tensioned concrete cylinder, hemispherical dome roof
Leak rate, %/d	0.1 (24 h), 0.05 (after)
Design pressure, psig	60
Free volume, 10^6 ft^3	2.0
Diameter/height, ft	130/195

MAJOR DESIGN CHARACTERISTICS OF LUNGMEN PLANT

SYSTEMS	Units 1 & 2
THERMAL AND HYDRAULIC (SECTION 4.4)	
Rated power (MWt)	3926
Design power (MWt) (ECCS design basis)	4005
Steam flow rate, Mlb/hr at 420^0F (FW Temp)	16.843
Core coolant flow rate (Mlb/hr)	115.1
Feedwater flow rate (Mlb/hr)	16.807
System pressure, nominal in steam dome (psia)	1040
Average power density (kw/l)	49.2
Maximum linear heat generation rate (kw/ft)	11.8
Average linear heat generation rate ((kw/ft)	4.16
Maximum heat flux (Btu/hr/ft^2)	380,775
Average Heat flux (Btu/hr/ft^2)	134,239
Maximum UO_2 temperature (^{0}F)	2529
Average volumetric fuel temperature (^{0}F)	1645
Average cladding surface temperature (^{0}F)	565
Minimum critical power ratio (MCPR)	1.31
Coolant enthalpy at core inlet (Btu/lb)	527.7
Core maximum voids within assemblies	75
Core average exit quality (% steam)	14.6
Feedwater temperature (^{0}F)	420
NUCLEAR (FIRST CORE) (SECTION 4.3)	
Water/UO_2 volume ratio (cold)	2.90
Reactivity with strongest control rod out (k_{eff})	<0.99

Initial average U-235 enrichment (%)	2.00
Initial cycle exposure (MWd/short ton)	9600
FUEL ASSEMBLY (SECTION 4.2)	
Number of fuel assemblies	872
Fuel rod array	10×10
Overall length (inches)	176
Weight of UO_2 per assembly (lb) (pellet type)	403
Weight of fuel assembly (lb) (includes channel)	656
FUEL RODS (SECTION 4.2)	
Number of fuel rods per assembly	92
Outside diameter (in.)	0.404
Cladding thickness (in.)	0.026
Diametral gap, pellet-to-cladding (in.)	0.007
Length of gas plenum (in.)	9.64(full length UO_2 rod)
	10.94(part length UO_2 rod
	15.64(full length gad rod)
Cladding material	Zircaloy-2
FUEL PELLETS (SECTION 4.2)	
Material	UO_2
Density (% of theoretical)	97
Diameter (in.)	0.345
Length (in.)	0.35
FUEL CHANNEL (SECTION 4.2)	
Thickness (in.) corner/wall	0.120/0.075
Cross section dimension (in.)	5.278×5.278
Material	Zircaloy-2

CORE ASSEMBLY (SECTION 4.2)	
Fuel weight as UO_2 (lb)	351,262
Core diameter (equivalent) (in.)	203.3
Core height (active fuel) (in.)	150
REACTOR CONTROL SYSTEM (CHAPTERS 4 AND 7)	
Method of variation of reactor power	Movable control rods and variable forced coolant flow
Number of movable control rods	205
Shape of movable control rods	Cruciform
Pitch of movable control rods	12.2
Control material in movable rods	B_4C granules compacted in SS tubes and hafnium
Type of control rod drives	Bottom entry electric hydraulic fine motion
Type of temporary Reactivity control for initial core	Burnable poison；gadolinia-urania fuel rods
CONTAINMENT	
<u>Primary</u>	
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 volume (ft^3)	259,563
Pressure suppression chamber free volume (ft^3) (HWL)	210,475
Pressure suppression pool water volume (ft^3) (LWL)	126,426
Submergence of vent pipe below suppression pool surface (ft) (HWL)	11.8 to 20.8
Design temperature of drywell (^0F)	340
Downcomer vent pressure loss factor	2.5-3.5
Break area/total vent area	0.01
Calculated maximum drywell pressure after blowdown (psig).	39
Pressure suppression chamber (psig)	26
Initial pressure suppression pool temperature rise (^0F) during LOCA	50
Leakage rate (% free volume/day)	0.5
Secondary	
Type	Controlled leakage
Construction	
Lower levels	Reinforced concrete
Upper levels	Reinforced concrete
Roof	Reinforced concrete
Design in leakage rate (% free volume/day at 0.25 in. H_2O)	50