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Addendum

(Paraguay)

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PARAGUAY

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/ 25 March 1987 /

Paraguay has, in addition to its contribution contained in A/CONF.108/NP/5, submitted two reports entitled: "Memoria de la Comisión Nacional de Energía Atómico del año 1984" (Report of the National Commission for Atomic Energy for 1984) and " Memoria de la Comisión Nacional de Energía Atómica del año 1985" (Report of the National Commission for Atomic Energy for 1985). These reports may be consulted in the Conference secretariat.

The general licensing procedure may be divided into the following main steps:

- Filing of application
- Consultation of the parties concerned
- Granting of license
- Granting of construction permit
- Granting of operating permit

Licensing of new reactors in Sweden is prohibited by a recent amendment of the Act on Nuclear Activities (Swedish Code of Statutes 1987:3). The facilities remaining to be constructed are those for the handling and final storage of nuclear waste and spent fuel. I will give a brief description of the licensing procedure and the steps mentioned earlier.

The application for a licensee to build a nuclear installation has to be submitted to the SKI, accompanied by a description of the proposed site and all particulars enabling the safety of the proposed installation to be assessed. The applicant sends a preliminary safety report to SKI and SSI. This report outlines the concept of the safety of the installation and analyses the effects of possible incidents.

SKI reviews the application in depth from a safety point of view and communicates the application to a number of national and local bodies for their opinion. Such bodies are the National Environment Protection Board, the national board for urban planning, the Swedish Metheorological and Hydrological Institute, the Board of Fishery and the SSI. The application is also transmitted to the country council and the commune council. There are no special provisions in Swedish legislation concerning public enquiry, on safety and radiation protection measures, only on general environmental impact on land and water resources etc. However, all recommendations and review documents are public and available for inspection and comment. The commune council may veto the siting of a nuclear reactor or facility for handling and storage of nuclear waste, as well as any industrial plant if it is the first one in the area. Following the grant of a license, the licensee must give a local safety committee information and opportunity to get insight into the safety and radiation protection work at the plant.

When SKI has got the opinions of the various bodies consulted and completed its own review, SKI transmits the application with its comments to the ministry. Decision is taken by the Government. The license granted by the Government covers site approval, construction and then commissioning of the installation, subject to the granting of permits for each of these stages by the SKI and the SSI.

When the operator has obtained the governmental license he must transmit to the SKI evidence demonstrating that he is able to meet the conditions laid down in the license and the testing programs. During construction the final safety analysis report must be transmitted, at least 6 months before the date planned for fuel loading. SKI closely supervises the work and evaluates the final safety analysis report. During the construction period and before granting permit to operate the plant, a careful control of the quality is made. Swedish Plant Inspectorate has an important role to play in this connection.

5. Inspection and regulation in practice

As mentioned above Swedish regulatory work is based on the concept that the owner of a nuclear installation bears the full responsibility for the safe design, construction, operation and maintenance of the installation. The rôle of the inspectorate is mainly to inspect and audit that the owner fullfills this responsibility in accordance with the conditions specified in the license. In some areas, SKI has issued formal rules valid for all nuclear power reactors in Sweden. These formal rules concern pressurized systems and components, assessment of operator education and training, quality assurance and plant physical security. The conditions for operating and maintaining a specific plant are laid down in technical specifications and a number of other requirements valid for the plant in question and stated in the plant license. However, the role of SKI is not only to review audit and inspect; according to its charter SKI is also committed to work for improved safety in all its activities; inspection, regulation and research.

As the construction period has now come to an end in Sweden, the focus of reactor safety work is shifted from design to safe operation and maintenance. The cornerstone in the safety work of SKI is the continuous daily contact between utilities and the inspectorate. SKI has no resident inspectors, but there is a small group of inspectors assigned to each nuclear site, ensuring in-depth knowledge of plant design and performance. The operational state of each plant is reported to SKI daily by telex. Typically there are also daily contacts by telephone between SKI and the nuclear power plants and the plants are visited by SKI inspectors several times per month. In this manner a continuous dialogue is kept running between the inspectorate and the utilities on both short and long time safety issues.

In this dialogue the inspectorate carefully avoids to mix into the process of solving problems. It is important that utility responsibility for safety is kept unbroken and complete. However, SKI will express their views on the safety importance of various issues, set time limits for addressing them and review the solutions proposed by the utility.

Under the 1984 act SKI inspectors have full powers in performance of their duties as regards access to buildings and documents. The inspectorate has also authority to give order of plant shut down if, according to the judgement of the inspectors, plant safety is severely threatened. If the utility opposes such a desicion they have the possibility to appeal the Government. The plant would, however, be shut down with writing for a Government decision on the appeal issue. It should be underlined that such a conflict between utility and the inspectorate so far has not appeared in Sweden.

The work on improving reactor safety includes:

- Incident reporting and analysis, including both SKI's own analysis and SKI auditing industry work in this area.

- Review of technical improvements to plants proposed by industry.

- Review and modernization of SKI formal rules and regulations. For example, a major revision of the SKI code for pressurized components and systems has just been completed to take into account operating experience within service inspection and maintenance. Work on extending the competence assessment system from operators to other key plant personnel, e.g. working in maintenance has started.

- A recurrent safety analysis programme (the ASAR programme) according to which each plant is subject to a thorough safety review every 8 to 10 years. The utilities are asked to prepare an "As-Operated Safety Analysis Report" (ASAR), which is reviewed by SKI. The subject coverage of the ASAR is specified by SKI and includes, inter alia, a detailed plant specific probabilistic safety analysis (level 1, i.e. upt to core damage), a thorough analysis of operating experience and of safety improvements made so far, as well as a proposed safety improvement programme for the coming years. There is a substantial effort involved in completing each ASAR and reviewing it - several tens of highly qualified man-years.

- An 8-year research, development and implementation programme decided by Government and Parliament in 1981 to mitigate releases that can cause ground contamination in case of a severe accident.

Characteristic for the Swedish system is the cooperation between authority and utility in developing nuclear safety. For example joint research and development projects are not unusual. This cooperation, however, does not influence the independent judgement of the authority. A condition to reach such a situation is that the authority has a high competence to assess and specify issues and review proposed solutions and that the authority always carefully watches its integrity. The fact that all SKI documents containing decisions and safety review are public documents open for inspection and debate works as a control mechanism.

7. The Organization of SKI

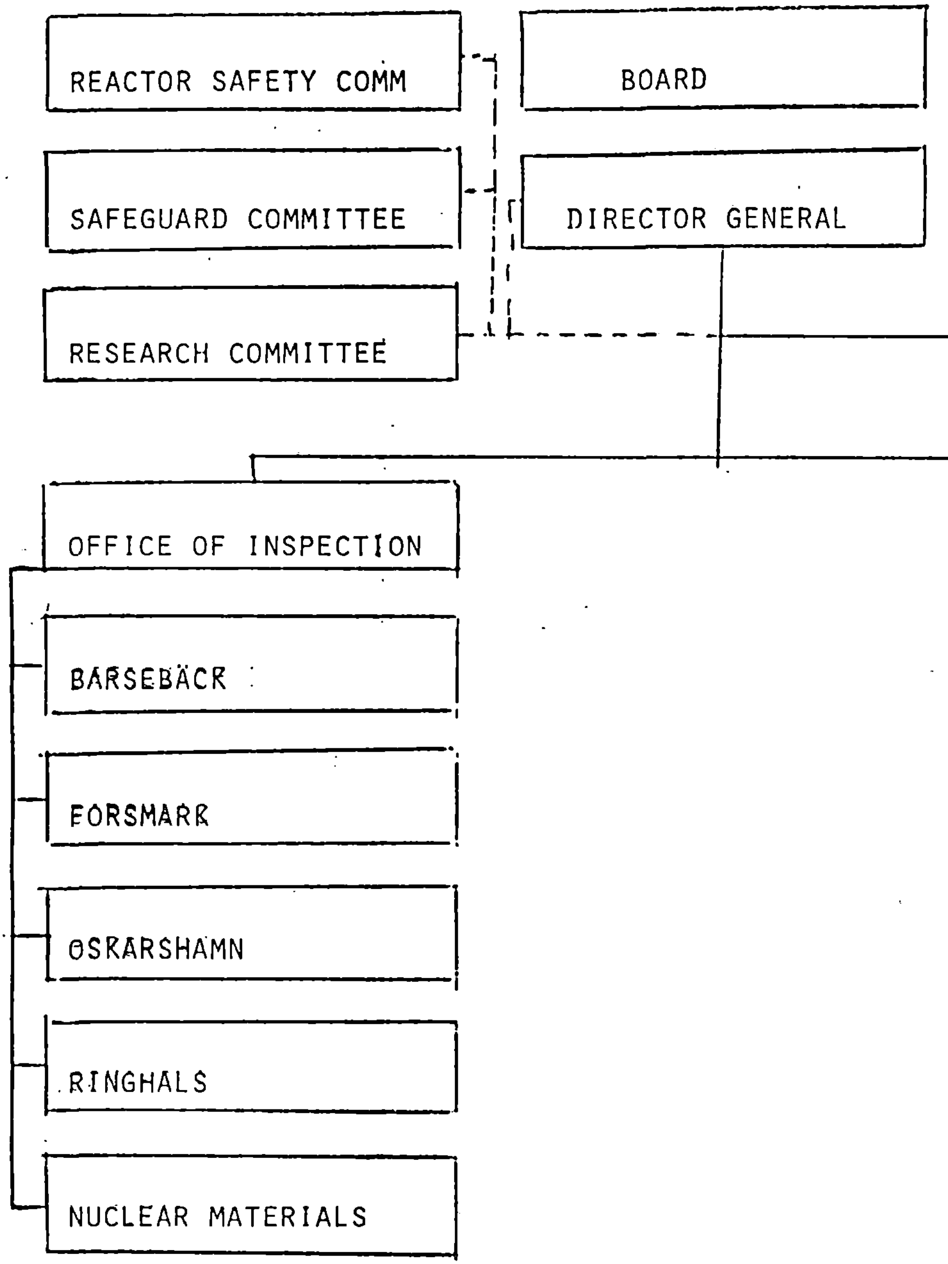
The SKI is governed by a board with representatives from the public. The Director General is chairman of the board. There are three advisory committees,one of them for reactor safety, one for safeguard and one for research. The inspectorate is organized in two offices, the office of inspection and the office of regulation and research (attachement 1). The office of inspection is responsible for measures directly related to operation of nuclear installations. The office of regulation and research is responsible for licensing reviews and safety assessments, reliability analysis, systems for waste handling and storage and for research and development. The research program concerns aspects of safety with the aim of improving the ability of the inspectorate to carry out its duty, of improving the safety of the installations and of contributing to the national competence in general in the field of nuclear safety.

7. Concluding remarks

The guiding principles for the Swedish system of regulation is that the primary responsibility for safety should be put as close as possible to those who are directly operating and maintaining the plant.However, the utility responsibility for safety must also be exercised through internal safety auditing procedures. Thus SKI requires each utility to have a safety committee for internal review of all important safety issues; this committee reporting directly to the top management of the utility. As the Government safety authority, SKI has to supervise how the utility organization for operation and maintenance and safety audit are performing so that safety as far as possible is guaranteed in all situations; SKI not only checking that rules and regulations are met, but also and most important by exercising a qualified and independent technical judgement.

So far, the Swedish model for nuclear safety, with its specific roles assigned to utilities and authority as described above, appears to have performed well. 12 nuclear power plants have been built to essentially the same safety standards as in other western countries, in some areas even stricter. There have been few delays in plant construction due to regulatory action (but some due to political action). Plant safety and performance indicators, such as availability and reliability, frequency of scrams and of incidents of major safety significance, as well as occupational and environmental radiation exposure rate well in international comparison.

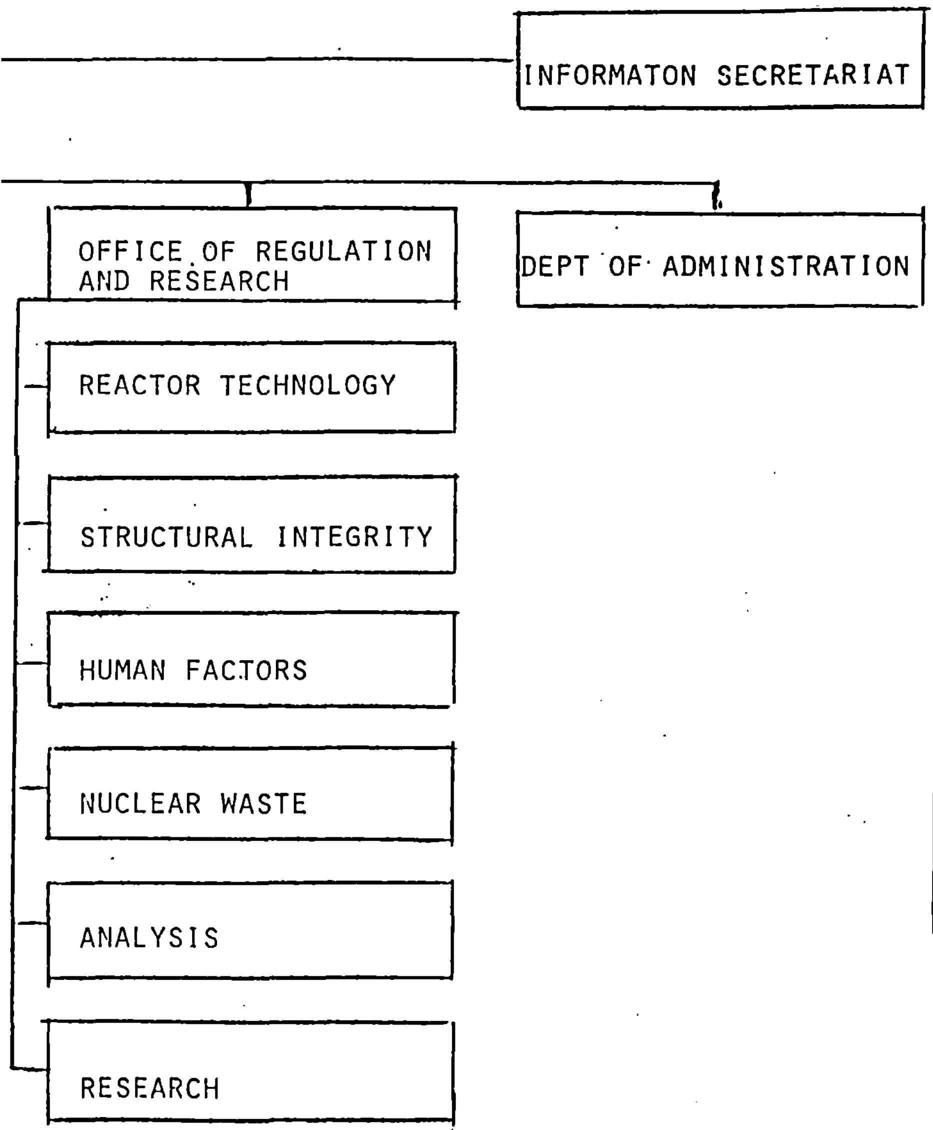
In conclusion I would like to stress the importance of competence, both within the authority and the utility, the dialogue between the parties, the trust, independence and the integrity of the safety authority, and the general safety culture in the country as conditions for high safety in operating complex systems such as nuclear power plants.



REACTOR SAFETY COMM
BOARD
SAFEGUARD COMMITTEE
DIRECTOR GENERAL
RESEARCH COMMITTEE
OFFICE OF INSPECTION
BARSEBÄCK
FORSMARK
OSKARSHAMN
RINGHALS
NUCLEAR MATERIALS

SKi
STATENS KÄRNKRAFTINSPEKTION
Swedish Nuclear Power Inspectorate

ORGANIZATION SCHEME



Attachement 1

REGULATORY ASPECTS FOR NUCLEAR SAFETY IN BRAZIL

R.N. Alves, J.E.L. Salvatore, B.C. Pontes (CNEN)

(Brazil)

1. Introduction

According to the specific Brazilian federal law, activities concerning regulatory procedures are under the Comissão Nacional de Energia Nuclear (CNEN) responsibility. These activities include:

- Issuance of regulations, rules and authorizations related to:
    - Reactors and other nuclear installations;
    - Use, storage and transportation of nuclear materials;
    - Trade of nuclear materials and nuclear ores.
- Issuance of regulations and safety protection guides for:
    - Use of nuclear materials in any kind of installation;
    - Processing and disposal of radioactive waste;
    - Construction and operation of facilities designed for the production of nuclear materials and nuclear energy.
- Licensing and inspection of nuclear installations.

According to CNEN's organization, the activities pertaining to nuclear safety are carried out under the Executive Directory I. This Directory is responsible for three departments (Fig. 1).

It is the task of the Department of Standards and Specifications to propose regulations, standards, specifications, methods and systems to assure the safe use of nuclear energy.

The Department of Reactors is responsible for the safety evaluation, inspection and enforcement during the construction, pre-operational and operational phases of the nuclear power plants and research reactors.

The Department of Nuclear Materials and Installations handles the safety evaluation, inspection and enforcement of the nuclear installations. It qualifies and controls users of radioisotopes, ionizing radiation and nuclear materials.

The Institute of Radiation Protection and Dosimetry is responsible for the development of techniques, the control and standardization in the fields of radiological protection and dosimetry of ionizing radiation, aimed at reducing dose exposures to radiation to acceptable levels in accordance with the regulation in use.

The regulatory activity and licensing of nuclear installations is a multidisciplinary work in view of the necessary safety review that must be made of the Preliminary and Final Safety Analysis Reports (PSAR and FSAR). The degree of specialization required for the performance of these tasks is very high.

When a nuclear programme is initiated, its implementation must be carefully considered from the point of view of the necessary manpower, not only for the construction and operation of NPP's by the licensee, but also for the licensing process and inspections by the regulatory body.

It must be considered that the safety of a nuclear power plant depends much on the way it is designed, constructed and operated. Considering plant operation, there must be an adequate training programme and effective training supervision in order to prepare the operating staff for safe operation and for CNEN's licensing examinations. It is, however, the specific task of the regulatory body to assure that the organizations involved in all the above phases have a high commitment to safety.

The regulatory body must have the necessary technical competence appropriate to its responsibilities and, as such, must have suitable training programmes for its personnel to gradually produce that competence. Since the beginning of implementation of the Brazilian nuclear power programme, CNEN has realized the magnitude of its tasks and has striven constantly to acquire all the necessary technical competence.

2. Methodology

The activities pertaining to licensing (Fig. 2) can be divided into two major areas, namely:

- Safety review;
- Quality assurance.

Licensing steps during NPP licensing encompass:

- Site approval;
- Construction licences;
- Operating licences:
  - low-power physics tests and power ascension;
  - commercial operation.

2.1 Safety Review

The safety review is performed by CNEN and takes into consideration four aspects, namely:

- Comparison with the reference plant;
- Independent calculations;
- Fulfillment of applicable norms and regulations;
- Observation of world experience.

The country's infrastructure has a strong effect on each of these aspects.

Reference Plant

The tendency to standardize the construction of power plants helps, to a certain extent, to simplify the licensing process. With this in view, CNEN decided to define a "Reference Plant" in its resolution CNEN 2/1976 which reads as follows:

- The project for a proposed nuclear power plant must be based on a similar plant in the same capacity range. For this purpose, the organization applying for the construction licence must indicate a "reference plant" with the following characteristics:

  (a) It must be located in the country of the principal supplier;

  (b) It should be licensed, or be in the final stage of the licensing process; in the latter case the project must have been approved;

  (c) A plant can also be chosen which has already been commissioned, thus allowing the new project to profit from its experiences in the pre-operational, commissioning and start-up phases.

- The applicant must justify selection of the reference plant and indicate the differences it presents in comparison with the projected plant with regard to capacity and design characteristics, analyzing the effects on nuclear safety;
- The application documents must contain - in English or Portuguese - all clauses and standard specifications referring to all parts of the project;
- The applicant must submit to CNEN all technical information necessary to prove the safety of the project.

The utilization of a reference plant in a licensing process has the advantage of:

(a) Proving access to comparative data that are needed for the three other processes applied;

(b) Compensating (via transfer of technology) for our lack of experience in some fields of advanced technology. The use of a reference plant has to be considered as a stage in the entire process. Simply dominating its technical problems is not the entire answer to the technology transfer.

Independent Calculation Methods

In the licensing process for nuclear installations, it is indispensable to have recourse to independent calculation methods permitting, by means of computer simulation, analysis of the normal operating conditions of a plant, as well as accident situations.

Three types of resources have to be integrated to assure effective utilization of the codes:

- A team of experts in the areas to be treated, able to analyse and select with regard to the defined objective the most appropriate codes and to continually develop new ones adjusted to Brazilian requirements;

- A team of experts in calculation and processing, able to operate the codes selected and assist the specialists of the former group in data processing;

- The capacity for data processing which consists of efficient equipment and disposition of adequate peripherals, allowing for increased efficiency in performance.

This independent capacity is based on a library of over 100 computer codes, most of which have been obtained from the world market and adapted to Brazilian standards so as to fulfil specific conditions.

Some of the codes referred to have been developed by CNEN alone or by Brazilian institutes and universities under the sponsorship of CNEN's technical programmes.

Standards and Regulations

The establishment of standards is one of the last steps in technological development as they tend to consolidate the "status quo".

Nevertheless, their importance as a disciplinary instrument makes them indispensable from the very beginning, when safety conditions must be guaranteed.

A systematic inspection of the installation project in order to verify whether or not standards and regulations stipulated in the licence have been respected represents an important part of safety analysis. This process includes control over observance of safety and general design criteria.

The licensing authority must have at its disposition many of the applicable standards, codes and regulations in effect within the industrialized nations and must establish with these countries a system of information exchange, taking into account the most recent regulations and their impact on related industries and the governments of the countries involved.

In Brazil, particularly in the nuclear sector, the system of standardization is influenced by the standards issued by international organizations and by the country from which the technology has been transferred. Consequently, the standardization process has to be dynamic, allowing a constant adjustment to new requirements. In certain cases, it is advantageous to adopt temporarily international standards or those of the country of provenance of the transferred technology. In other cases, even to elaborate standards to which the organizations working in the area concerned have consented. These are subject to a two-year test period. These two systems render possible an exact evaluation of the effect which the adopted standards will have on Brazil. At present, there are 34 basic rules already developed by CNEN. There are several others under development.

The hierarchy of standards in Brazil does not differ greatly from that in other countries. In Fig. 3 this hierarchy and the role played by both governmental and private entities is shown. It is important to observe that rules issued by CNEN have, by law, a mandatory characteristic while those issued by private institutions are voluntary by nature.

World Experience

The CNEN staff strives to keep itself informed of the latest developments relating to nuclear power in the world. This includes not only familiarity with technical publications and conference proceedings, but also abnormal occurrences related to safety which may happen to NPPs and other nuclear installations operating in other countries. In such cases, a comparison is made between the foreign plant and those which are being licensed in Brazil, attempting to determine what corrective steps should be taken if such an incident should occur.

2.2 Quality Assurance

One of the requirements for the licence of a NPP in most countries is the approval by the licensing body of a programme which can guarantee the quality of systems and equipment relating to safety.

Two basic philosophies have been used to guarantee the quality required, namely:

- the systems-related philosophy; and
- the object-related philosophy.

According to the systems-related philosophy, organizations involved in the design, manufacture and operation of the NPP must have an approved Quality Assurance Programme. The licensing body must guarantee, through a system of audits and inspections, that the approved programmes are being executed adequately.

The object-related philosophy calls for a more comprehensive system of independent inspections, stressing the quality control criteria from the Quality Assurance Programme.

In Brazil, the Quality Assurance review is performed by CNEN in accordance with the "Code of Practices on Quality Assurance" from the IAEA upon which the Brazilian rule is based.

In addition, an Independent Technical Supervision Organization (IBQN) was created outside CNEN as a means of optimizing system- and object-related philosophies of Quality Assurance.

In order to carry out this system of audits and inspections, CNEN established, from the beginning of construction at the site of its first NPP, an office where resident engineers conduct routine inspections, as well as give support to headquarters inspection teams that visit the site almost weekly to conduct inspections and/or audits. This site office is also responsible for the routine inspections during plant operation and plays a central role in keeping headquarters informed of plant conditions during an abnormal situation. This resident inspector system has proved to be quite effective in conducting inspections and giving the necessary assistance and feedback to the licensing activity.

Inspection of Construction and Operation

CNEN's participation all along the construction, commissioning and operational phases of nuclear power plants is quite active.

The inspection team of resident inspectors (3 for operation, 3 for construction); besides them, headquarters' technical groups personnel enhances the efficiency of the resident inspectors whenever necessary.

In Angra 1, over 40 safety-related tests have been selected by CNEN for surveillance. In addition, a complete audit was conducted at the end of the commissioning phase to verify if all tests had been carried out and the results were reviewed and approved by the utility personnel. A selective analysis was made of the results of the tests.

During construction, CNEN conducts a programme of inspections. The resident inspectors have proved to be quite effective in conducting routine inspections, keeping headquarters informed of all site activities and supporting, as well as participating in, inspections and audits carried out by other CNEN inspectors (see Appendix A).

In addition, in relation to environmental protection programmes, an independent programme to collect baseline data for the pre-operational out-of-plant monitoring programme was conducted and an operational programme is being conducted through the Institute for Radiation Protection and Dosimetry (IRD) independently of the programme carried out by the utility.

Conclusion

The Code of Practice for the Safe Operation of Nuclear Power Plants states that:

> "In discharging its responsibility for public health and safety, the government should ensure that the operational safety of a nuclear reactor is subject to surveillance by a regulatory body independent of the operating organization".

In Brazil, this task is being carried out by CNEN in accordance with the best international practice.

For this, CNEN has exchanged extensive collaboration with regulatory bodies and safety institutions of other countries and the IAEA, not only in the training of people but also in the area of specialized inspections.

The experience so acquired is already being exchanged with regulatory bodies of other fellow developing countries. International co-operation is very important in the development of a good regulatory capacity.

BRAZILIAN NUCLEAR ENERGY COMMISSION

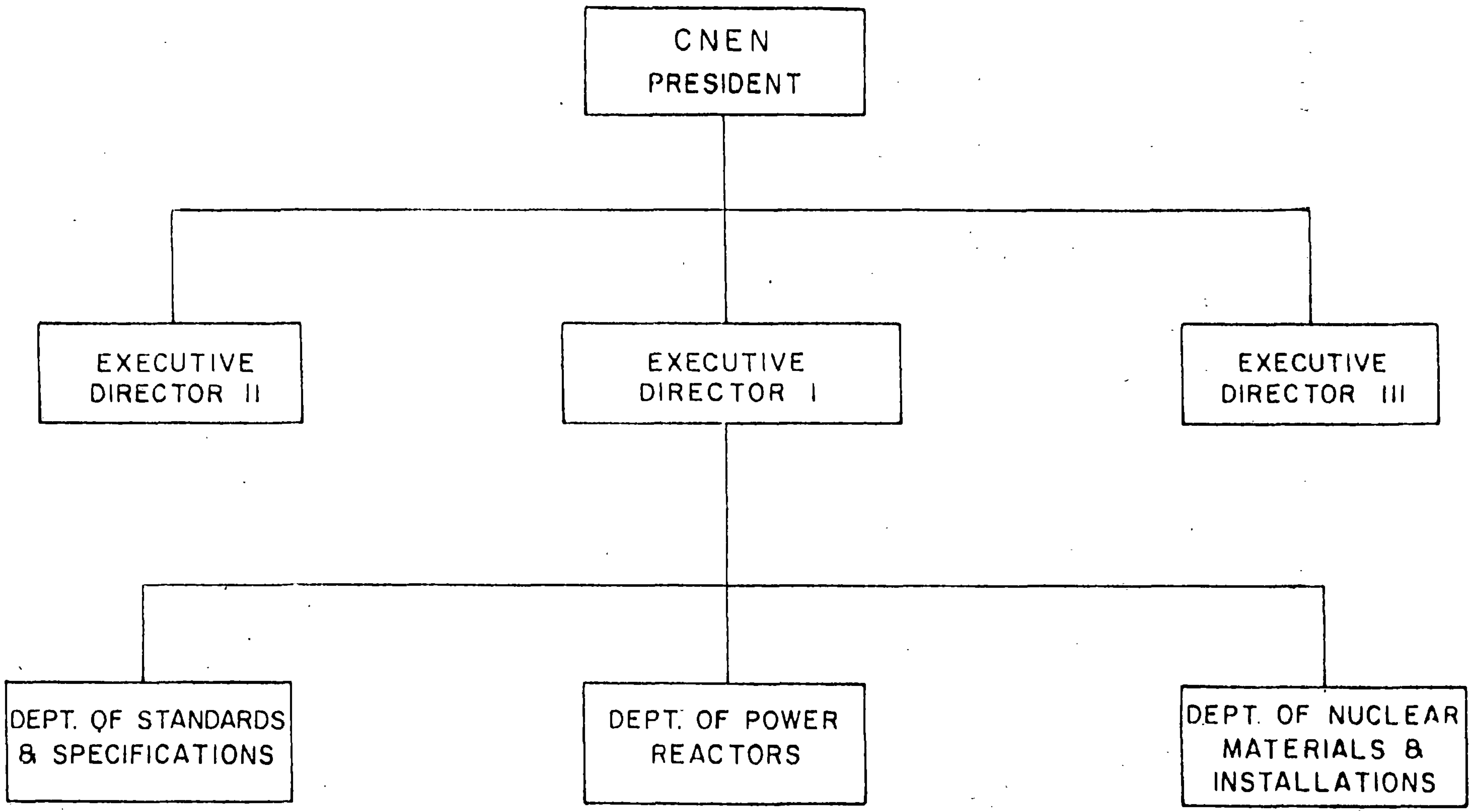
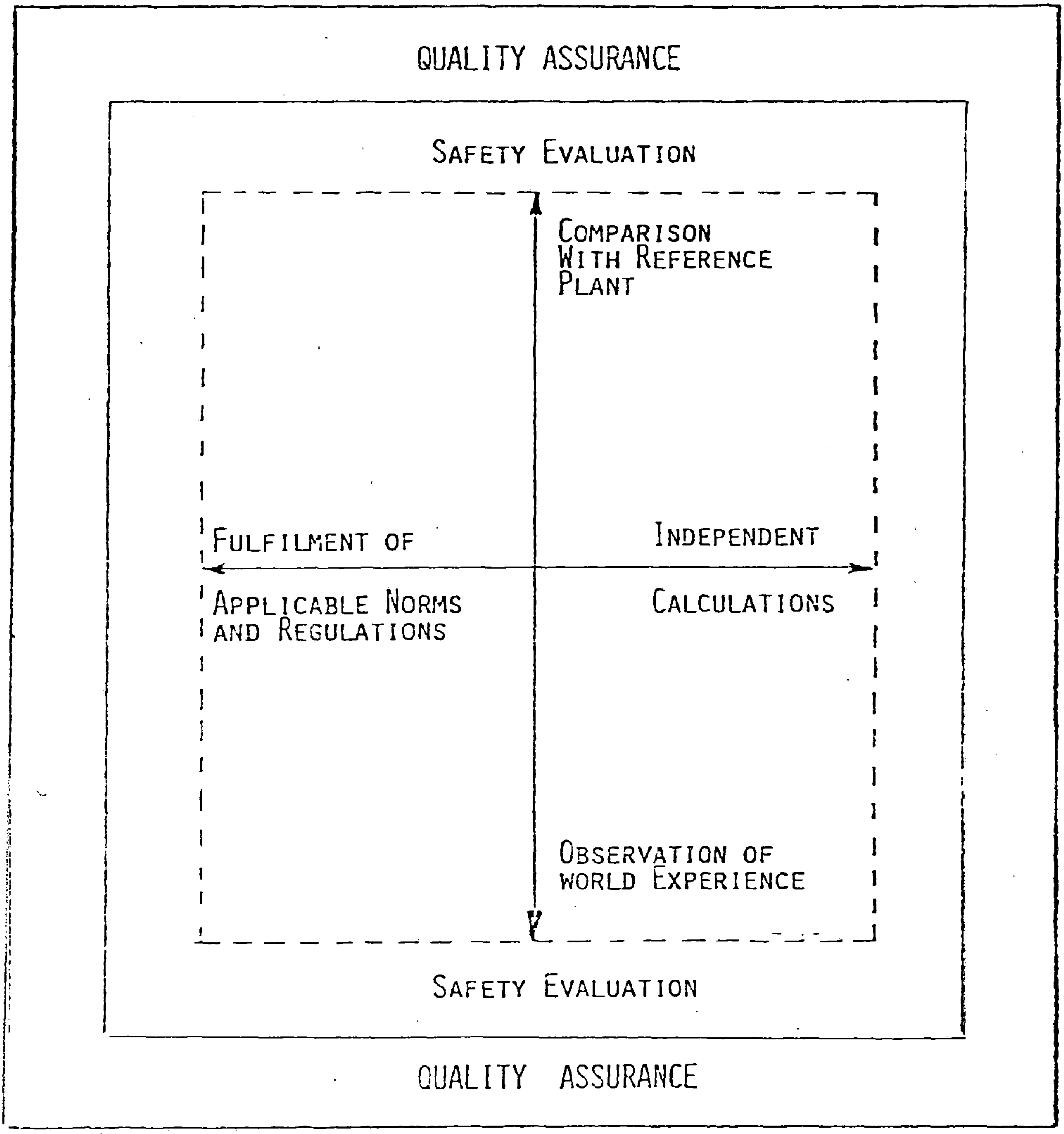


FIG. 1



LICENSING ACTIVITIES

FIG. 2

HIERARCHY OF STANDARDS

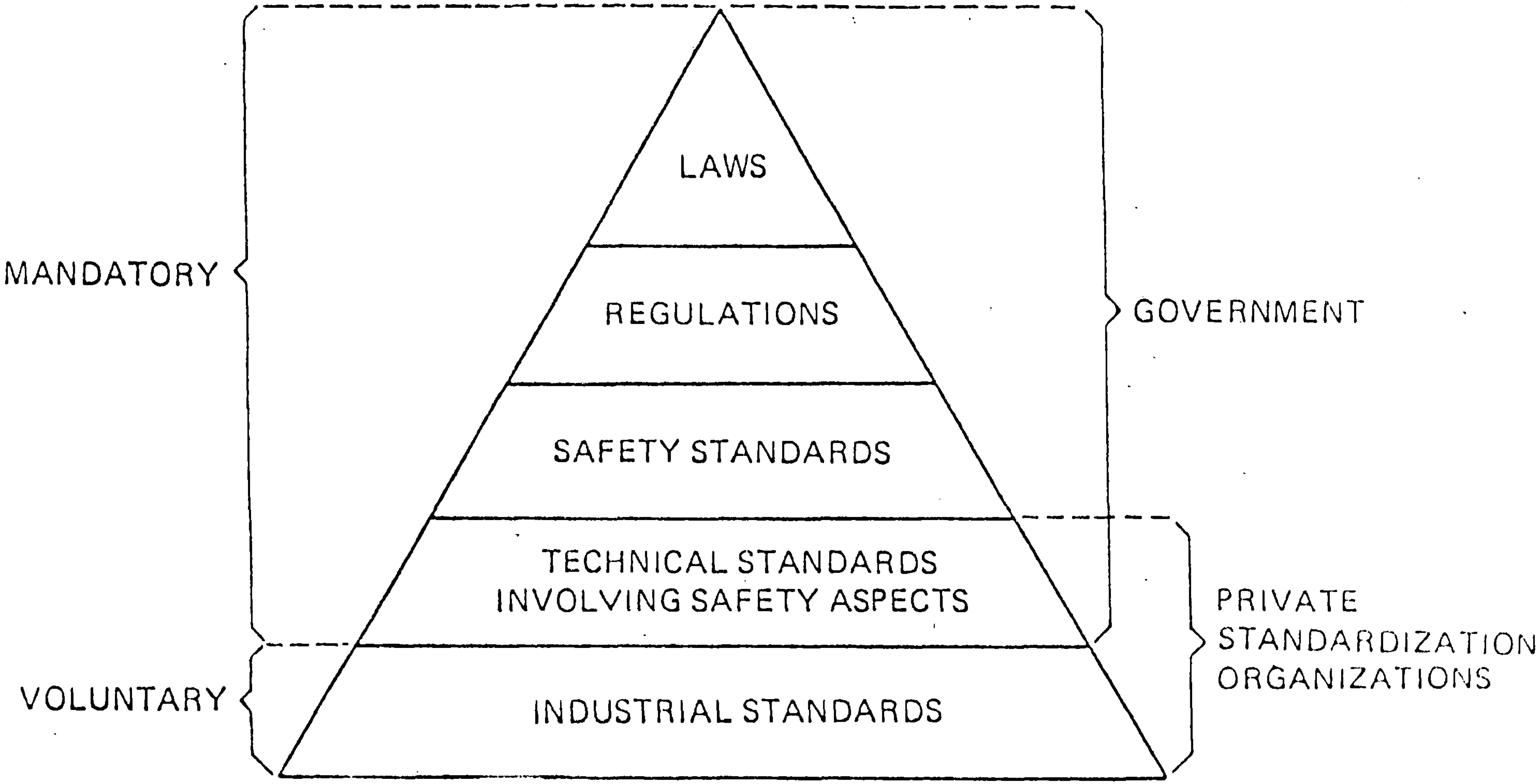


FIG. 3

APPENDIX A

Major Inspections and Audits conducted by CNEN at the Angra site

AUDITS

- o Welding-procedure specifications, material control and welder qualifications
- o Nondestructive examination
- o Civil work - turbine building
- o Operating staff training
- o Calibration and control of measuring and test equipment
- o Electrical systems: cables, raceways and containment penetrations
- o Electrical components
- o Document control
- o Civil work - procedures
- o Record control - QA records collection and storage
- o Fuel elements - manufacturing documents
- o Start-up master file

(inspections - cont.)

- o Hydrogen control system
- o Fuel building - release activities - follow up
- o Electrical system - cables and raceways
- o Chemical and volume control system
- o Emergency power sources cooling system
- o Safety injection system
- o Residual heat removal system
- o Hangers
- o Instrumentation - sensing lines and transmitters
- o Process safety related instrumentation
- o Nuclear instrumentation system

INSPECTIONS

- QA Program - receiving, storage and handling of equipment and material
- Electrical systems
- Document control - civil work
- Foundations of Angra 2 - concrete quality control
- Residual heat removal system
- Civil work - auxiliary building
- Document control - Westinghouse - EBE
- Civil work - shield building
- Mechanical installations
- Civil work - concrete aggregates
- Housekeeping
- Welding - joint fittings
- Painting - auxiliary and safety buildings
- Civil work - auxiliary building
- Welding - procedures - visual examination of welds - UT
- Fuel building - visual examination
- Angra 2 - foundations
- Fire prevention and protection
- Civil work - observation of work activities
- Motor control centers

(Inspections - cont.)

- Civil work - procedures PQE-1 and PQE-5
- Mechanical components and systems - safety related components - work activities
- Welding - material control and visual examination
- Civil work - construction release activities
- Nondestructive test
- DC electrical systems - batteries and chargers
- Electrical systems - inverters and AC panels
- Containment penetrations - observation of work activities
- Angra 2 - pile foundations
- Water systems
- Containment spray
- Nuclear steam supply system - observation of work activities
- Spent fuel pit cooling system
- Auxiliary feedwater system
- Fuel building - release activities
- Civil work - pile foundation, Angra 2
- Main steam system
- Feed water heating system
- Containment cooling and ventilation system
- Emergency diesel generators Nº 1 and 2.

THE SAFETY SUPERVISION OF NUCLEAR POWER IN THE PEOPLE'S REPUBLIC OF CHINA

Ms. Oian, Jingjing

(China)

Introduction

This article gives a brief account of the nuclear power safety supervision system in China.

China's policies and guidelines on the development of the nuclear power industry aim to develop a positive and appropriate approach and place safety and quality first in the construction of nuclear power plants. This applies, to be specific, to siting, design, construction, commissioning, operation and decommissioning of nuclear power plants.

China's nuclear power programme is still at its initial stage. It was only in 1985 that the construction of its first nuclear power plant started. From the very outset, the Chinese Government gave full importance to nuclear safety. The research work started in the 1960's and an institute of radiation protection was established. In 1980, the relevant ministry began to draft safety regulations. The establishment of a nuclear safety supervision system was also initiated when the design and construction of China's first nuclear power plant began.

1. The National Organization

When a country is to proceed with its nuclear power programme, an important measure to ensure safety is to establish a nuclear safety supervision organization. China's National Nuclear Safety Administration (NNSA) was established in October 1984 with the approval of the State Council. In exercising its supervisory authorities, the Administration, reporting directly to the State Council, is independent of any departments or industries dealing with nuclear power development, application and operation.

At present, the Administration has a staff of 50 and 7 divisions:

- division of regulations;
- division of technical review;
- division of nuclear power stations;
- division of support equipment;
- division of radiation protection;
- division of inspection;
- division of research.

In addition, there are two regional inspection offices located in Shanghai and Guangdong Province respectively, working under the Administration with a total of about 20 staff members.

The main responsibilities of the Administration are as follows:

(a) To formulate and enact regulations, rules and guidelines for nuclear safety and to conduct reviews on technical standards related to safety;

(b) To review and assess the safety of nuclear installations and capabilities of the applicants to ensure safety and, in accordance with safety regulations, to issue or revoke nuclear safety licences;

(c) To perform regulatory inspection on nuclear installations, nuclear materials and radiation protection;

(d) To assess and provide information on any nuclear accident and its radiological consequences, and to settle disputes related to nuclear safety, if any;

(e) To organize research in nuclear safety management and technologies and co-ordinate important national research programmes;

(f) To formulate policies related to nuclear safety and to disseminate information, promote public education and organize training in the area of nuclear safety;

(g) To develop and co-ordinate international activities in the area of nuclear safety, and to negotiate and implement nuclear safety agreements with other countries and international organizations.

There is an Advisory Committee under the Administration composed of 26 well-known Chinese experts in the fields of nuclear engineering, safety and so on. The Committee holds meetings irregularly on major scientific and technical issues related to nuclear safety and gives recommendations to the Administration.

2. Nuclear Safety Legislation

The first group of nuclear power reactors are to be built in the economically prosperous and densely populated areas. Therefore, safety is an issue of vital importance. To ensure safety and protect the environment, necessary rules and regulations must be set up, defining clearly safety procedures, requirements and the responsibility of each organization concerned in nuclear safety.

The Administration, as a focal point, organizes relevant authorities to enact nuclear safety regulations and codes. The classification of this code system and a working plan have already been worked out. China's nuclear safety code system, based on other countries' experiences and relevant IAEA regulations with respect to the technical content will be similar to those of advanced nuclear power nations and consists of two parts: administrative regulations and standards.

(a) Administrative Regulations

Regulations on Nuclear Safety Management - specifies such important issues as the scope of regulations, regulatory body and its functions, principles and procedures of surveillance, etc. It is issued by the State Council and has full legal force.

Implementing Rules - specifies in detail the implementation requirements of the regulations on nuclear safety management. It has legal force and is issued by NNSA.

Nuclear Safety Codes - specifies technically the safety goals and basic safety requirements. It is also issued by NNSA and has legal force.

Nuclear Safety Guides - recommends methods and procedures in the enforcement of nuclear safety codes. It is issued by NNSA as a supplement to the safety codes but does not have legal force. When using methods other than those given in the guides, however, the safety of the new methods has to be proved to NNSA before they can be applied.

(b) Standards and Criteria

All standards and criteria are to be reviewed and approved by NNSA. Following international practices, China permits the application of valid standards and criteria issued by a supply country to the imported nuclear plant or equipment as long as they are in accordance with China's nuclear safety regulations and codes. For nuclear power plants designed and constructed by Chinese companies, the decision on the types of standards is left to the users with the approval of NNSA.

So far, the first regulations and four codes have been promulgated. Four sets of regulations and five codes are under compilation. In addition, China plans to issue 49 safety guides, of which 15 are expected to be issued by the end of 1987.

3. Licensing

China has established a licensing system for nuclear installations. Before a nuclear power plant can be built or operated, the party involved has to apply and obtain a construction permit and operation licence. Operators of a nuclear installation must pass an examination in order to obtain an operation licence. The safety licensing procedure involves the following stages:

(a) Prior to construction, the applicant shall submit an "Application for the Construction of Nuclear Installations", "Preliminary Safety Analysis Report" and other relevant documents. After careful review, NNSA may approve the application and grant a construction permit. Only when the licence is obtained can the construction begin;

(b) Prior to the fuel loading and commissioning, the applicant shall submit "The Final Safety Analysis Report". The fuel loading permit will be granted only after satisfactory review and approval. Commissioning and trial operation can then proceed;

(c) Prior to the actual operation, the applicant shall submit "The Revised Final Safety Analysis Report". Operation licences will be granted only after satisfactory review and approval.

A group of 120 experts from major universities and research institutes work either full-time or part-time on these safety reviews.

The safety review and assessment for Daya Bay Nuclear Power Plant started in January 1987. The safety review for the construction phase is expected to be completed by the end of 1987.

4. Inspection and Enforcement

Nuclear safety inspection and enforcement are of vital importance in ensuring that all activities - siting, design, construction, commissioning, operation, and decommissioning of nuclear installations - are carried out in conformity with the regulatory requirements and licence commitments.

NNSA is authorized to despatch inspectors to manufacturing workshops, construction sites or operating nuclear installations. The inspectors are entrusted with the following duties:

(a) To review whether the information submitted is real;

(b) To supervise whether the construction meets designed specifications;

(c) To ensure that quality and safety requirements are met;

(d) To ensure that the construction and operation of the nuclear installations is in keeping with the provisions of relevant nuclear safety regulations and codes;

(e) To make sure that the operator is competent in safety operation and implementation of emergency measures.

Nuclear safety inspection is conducted on the basis of Safety Analysis Reports submitted by applicants, the conditions specified for obtaining safety licences and Safety Codes for Quality Assurance in Nuclear Power Plants. In July 1986, NNSA convened expert meetings to conduct preliminary reviews on the quality assurances of Qinshan and Guangdong Nuclear Power Plants.

The execution of nuclear safety inspection requires a set of documents for guidance, such as guidelines, handbooks and inspection programmes. The development of these documents and programmes is now well under way.

5. Scientific Research

The Chinese Government attaches great importance to research and development in the field of nuclear safety and has identified it as one of the key programmes in the Seventh National Five-Year Plan. This will also guarantee the necessary financial support. Over 100 projects will be conducted during this period.

A number of research projects in connection with nuclear safety have been carried out in China, such as:

(a) The establishment of a computer software system for safety analysis;

(b) Probabilistic Safety Analysis for Guandong and Qinshan Nuclear Power Plants;

(c) Inspection technology research;

(d) Research on radiation protection and emergency management;

(e) Simulation in emergency control.

Conclusion

According to our practices, we feel that the following five points related to nuclear safety are of great importance in the development of nuclear energy:

(a) In spite of the small probability of a serious nuclear accident in a nuclear power plant, a major accident is not impossible, as demonstrated by the Chernobyl Accident. It is, therefore, of paramount importance to give predominance to the safety issue in developing the nuclear power industry and to implement the quality control system in the construction and operation of power plants;

(b) In the construction and operation of a nuclear power station, the responsibility system must be thoroughly instituted, with the role of every unit specified. This is an effective way of guaranteeing nuclear safety;

(c) Nuclear safety inspection authorities must be assigned full power to supervise the safety issue independently;

(d) Emergency measures for possible nuclear accidents must be worked out before a nuclear power plant is allowed to go into operation;

(e) Publicity should be given to nuclear science and technology and to policies pursued on nuclear safety, in order to dispense public misgivings and scepticism over nuclear safety.

The development of a safety supervision system for nuclear power in China is still in the initial phase. A great deal remains to be done before a comprehensive and effective safety system is established.

CURRENT INSIGHTS ON THE RISKS ASSOCIATED WITH U.S. LIGHT WATER REACTORS

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Introduction

Since the time of the accident at Three Mile Island, the United States Nuclear Regulatory Commission has been involved with a spectrum of issues relating to severe reactor accidents. These issues have related to modifications to plants, to the development of more specific regulatory policies on such accidents, and the performance of extensive research on accident phenomenology and radioactive "source terms." Within the past few years, this work has resulted in the publication of severe accident and safety goal policy statements (Refs. 1 and 2), and the development of a new source term analysis technology (Ref. 3).

Within the past two months, this work has also has also led to the publication of the Reactor Risk Reference Document, NUREG-1150 (Ref. 4), for public comment. In my talk today, I will focus on the methods, results, and future plans of this most recent work.

NUREG-1150 provides a current assessment of the likelihoods and risks of severe core damage accidents in five operating nuclear power plants. The plants studied--Surry, Sequoyah, Zion, Peach Bottom, and Grand

Gulf--are related to the principal containment design types in the United States. For these plants, studies were made of: the frequencies of internally-initiated accidents leading to severe core damage; the physical processes such as containment loadings and source terms resulting from these accidents; the offsite dispersion and consequences of radioactive releases; and the overall risks associated with these accidents. In addition, study was made of ways to reduce the estimated level of risk. That is, a series of accident prevention and mitigation features were assessed for their "risk-reduction" potential, as well as the range of associated implementation costs.

NUREG-1150 provides a summary of this risk information, as well as insights of possible importance to regulatory decision-makers. Supporting this report are a series of more detailed reports developed by NRC's principal risk assessment contractors--Sandia National Laboratories, Brookhaven National Laboratories, and Battelle Columbus Laboratories (Refs. 5, 6, 7, 8, and 9). NUREG-1150 itself (including appendices) is roughly 1000 pages in length; supporting contractor reports sum to approximately 10000 pages.

Advances in Risk Methods in NUREG-1150

The NUREG-1150 risk study provided the vehicle for the use of a number of new sources of information on severe accident risks. First, these studies considered plant design and operational features in place in the summer of 1985. As such, many of the plant modifications required as a result of the Three Mile Island accident were in place at the studied plants. These included both hardware and procedural modifications.

NUREG-1150 also has been the first NRC risk study to use advanced containment and source term analysis methods. With respect to the former, the contractor analyses underlying the report (Refs. 5, 6, 7, 8, and 9) made use of very detailed containment event trees to describe and probabilistically quantify the progression of a severe accident. The NRC's Source Term Code Package (Ref. 10) was used for risk analysis for the first time, providing a benchmark for the set of source term calculations needed for the study.

In a related and important matter, NUREG-1150 is the first NRC risk analysis to quantitatively assess uncertainties in containment loadings and source terms. That is, the risk results provided in NUREG-1150 include quantitative estimates of the uncertainty in risk due to uncertainties in a set of accident frequency, containment response, and source term issues. The issues considered number roughly thirty per plant, and include such items as common cause equipment failure rates, extent of containment pressurization due to hydrogen combustion, reevolution of radioactive species from reactor coolant system surfaces, and failure pressure of the containment structure. Since many of these issues are incompletely understood, interpretation of the available data by experts and the use of expert judgement when hard data were not available were necessary steps in the process. Using these methods, the NUREG-1150 risk estimates appear as ranges or distributions of values, rather than single estimates.

The NUREG-1150 analysis of offsite consequences was performed with a recently-developed model known as MACCS (Ref. 11). This computer code is a descendent of the widely-used CRAC2 code (Refs. 12 and 13) also developed for NRC. While it differs in a number of areas from CRAC2, the most noteworthy advances in the MACCS model include the ability to model multiple releases of radioactive material from the plant, and the use of health effects models based on the most recent analyses of, for example, human exposures. The latter models have been developed for NRC by a group of medical experts, under the overall guidance of Harvard University (Ref. 14).

General Results of NUREG-1150

Using the methods I have just described, analyses of the frequencies and risks of severe accidents have been performed. I will now turn to a brief discussion of the results of these studies.

The results of the NUREG-1150 core damage frequency analyses of five plants show distinct qualitative and quantitative differences. Mean core damage frequencies vary by roughly a factor of ten, with the highest mean value being approximately 1 in 10,000 per year.

The kinds of accidents leading to core melting vary considerably among plants. For example, while station blackout accidents were important to both boiling water reactors studied, the important failure modes of the power supply system were quite different. In one, diesel-generator failures were important; in the other, battery failures were most prominent. Among the pressurized water reactors studied, the most important accident types also varied. For one, station blackout was most important, while in the second, dominant failures involved the component cooling water system. The third plant was most affected by small loss-of-coolant accidents with failure of emergency core cooling recirculation systems.

These results clearly point out the importance of plant-specific design and operational features to the frequency of accidents. As I have described, many of the important plant features are less related to the nuclear steam supply system design than to the "balance of plant" design. In the United States, such designs vary widely. As a result, the NRC is now developing methods for individually analyzing all existing plants. In addition, the NRC is supporting the concept of standardization of future designs.

Like the results of the core damage frequency assessments, the risks of the five NUREG-1150 plants also show considerable qualitative and quantitative differences. In addition to differences in accident frequencies, these risk differences result from varying importances of particular severe accident phenomena, as well as from different siting characteristics.

In NUREG-1150, data are provided on the relative importance to risk of accident types and containment failure modes. For the former, these data show that accidents causing failure of both core and containment engineering safety features are most important to risk. Station blackout is a well-known example of such an accident.

The relative importance of containment failure modes in the studied plants varies with the containment loading phenomena important to a particular plant. For example, for the studied pressurized water reactors with "large, dry" containments, "direct containment heating" has been identified as a major source of uncertainty. This process involves the ejection of molten core material from the reactor vessel while the reactor coolant system is at high pressure. Under some conditions, this molten material could rapidly mix with the containment atmosphere, transferring sensible heat and undergoing exothermic chemical reactions. If such events were to occur, containment pressure could exceed failure pressures near the time of reactor vessel breach.

The overall risk ranges estimated in NUREG-1150 can best be understood in the context of two risk "standards"--the Reactor Safety Study (Ref. 15) and the NRC's safety goals (Ref. 2). In NUREG-1150, comparisons are made between the Reactor Safety Study's assessments of the Surry and Peach Bottom risks and the present assessments. In general, the former results lie near the upper end of the NUREG-1150 range. This appears to be the result of the somewhat lower present estimates of accident frequencies, as well as the now-recognized plausibility of lower containment failure probabilities and source terms.

As I have previously noted, the NRC has developed and approved general safety goals (Ref. 2). In NUREG-1150, the risk estimates for the five plants have been compared with these goals. In each case, the estimated range of risk falls beneath (that is, meets) these goals.

Plans for Completing NUREG-1150

With this brief discussion of results from the draft version of NUREG-1150, I will now turn to a description of a portion of the work now planned for the completion of the final report. Today I will discuss work relating to review of the draft report and the incorporation of new information.

As I have previously noted, NUREG-1150 has been published and released for public comment. This comment period extends through August of this year. In parallel with this, NRC is planning to support an "expert" review of the document. The review group is expected to consist to roughly ten to fifteen people knowledgeable in severe accident issues and risk assessment. Experts both from the United States and abroad are expected to be involved.

As many of you are aware, the NRC has a large research program on severe accident physical processes. This program is being carried out in cooperation with and with the support of a number of other countries, as well as representatives of the United States nuclear industry. The NUREG-1150 risk results reflect information from this program through roughly the end of 1985. For the final version of the document, updates to the data base used for important issues will be generated and incorporated into the risk estimation process. For example, the final version of the report will reflect recent experiments and code calculations related to "direct containment heating." As I noted previously, the uncertainty in this phenomenon is important to the overall risk uncertainty assessed for some of the studied pressurized water reactors.

The final version of NUREG-1150 will also incorporate risk estimates for a sixth plant. This plant will be the LaSalle boiling water reactor, which uses the Mark II containment design type. These risk results will include both internally- and externally-initiated accidents. In addition, external events studies will be incorporated into the present risk estimates of the Surry and Peach Bottom plants.

Detailed planning of the schedule for completing NUREG-1150 is now underway by the NRC staff and its contractors. It now appears that the document will be released early in 1988.

Before closing, I should also note that other projects are underway at NRC to make the NUREG-1150 and other risk information more accessible to and usable in the rest of the technical community. In one, work is now underway to use this information to assist inspectors at power stations in assessing the risk importance of "day-to-day" changes in plant configuration. In a second, a model is being developed to perform more general sensitivity studies on plant risk information. This model could then be used to, for example, assess the risk reduction potential of a possible hardware modification to a plant. Both of these projects are oriented to application on personal computers, increasing their degree of availability and potential utilization.

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INTERNATIONAL CO-OPERATION IN REACTOR INCIDENTS

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Thank you for this opportunity to discuss with you the importance of international cooperation in the peaceful uses of nuclear energy. It has been our perception that the concept of international cooperation in notification and assistance during radiological incidents is an important step forward in addressing the common interest of worldwide nuclear safety. I would like to share with you our activities in this area and how they fit with my understanding of the objectives of this conference.

In the United States, by law, use of source by-product and special nuclear material for any peaceful purpose is regulated by the Nuclear Regulatory Commission (NRC). The NRC's primary mission is to protect the health and safety of the public. The level of awareness of public health and safety was certainly heightened after the 1979 accident at Three Mile Island.

As a direct consequence of that accident, the NRC proposed to amend its regulations to enhance onsite and offsite emergency planning to ensure the continued protection of the public in areas around nuclear power plant facilities. The final rule requires that emergency planning considerations be extended offsite to significant distances from the facility. The first of the emergency planning zones encompasses an area of about 10 miles (16 km) in radius and considers potential population exposure to a radioactive plume that might result from an accident in a nuclear power reactor. The second emergency planning zone covers an area of about 50 miles (80 km) in radius and considers the foodstuffs that might become contaminated.

Over six years have elapsed since adoption of the upgraded emergency planning regulatory program. Appraisals, inspections, exercise observations, and response to actual events indicate that nuclear power reactor licensees and State and local governments have implemented emergency planning and preparedness programs in accordance with the NRC regulations.

The U.S. Government also has upgraded its response capability for coping with a radiological emergency. It has developed and implemented the Federal Radiological Emergency Response Plan to augment the capabilities of State and local agencies. This plan brings together, in a coordinated fashion, the full complement of resources of the U.S. Government to assist in mitigating the consequences of a major radiological accident to the public and environment. A full-scale field test of the plan was conducted in conjunction with an exercise at the St. Lucie Nuclear Power Plant site in Florida during March 6 through 8, 1984. Lessons learned from that exercise were implemented and another full-scale field test is scheduled for June 1987 at the Zion Nuclear Power Plant site in Illinois. In keeping with our mission to protect the public health and safety, we expect to exercise the full-scale Federal plan approximately every three years.

While the United States was strengthening our internal emergency planning we remained concerned with the potential impact of a radiological incident on neighboring countries and the health and safety of other populations. We recognized the need for international cooperation.

As a result of the United States initiative, the International Atomic Energy Agency (IAEA) Board of Governors adopted a resolution in which it requested the Director General to convene a group of experts -- open to all member nations -- to study the most appropriate means of responding to the need for mutual assistance in connection with radiological accidents and of facilitating international cooperation in the area of nuclear safety. This group of international experts recommended the prompt development of a single set of provisions setting forth, in the form of an Information Circular, the terms and conditions that could be applied to giving emergency assistance. These provisions could serve as a model for the negotiation of bilateral or regional agreements, which certainly are to be encouraged. The provisions should be such that they could readily be agreed upon between a requesting and an assisting nation at the time of a radiological emergency.

Another recommendation of the international experts related to the need for prior arrangements among nations to cope with transboundary impacts of a radiological emergency. In the experts' view, such arrangements would have to cover matters such as establishing a threshold for reportable events, integrated planning, and exchange of information. The Board of Governors of IAEA approved these recommendations and authorized the Director General to implement them.

Subsequent meetings held by this group of international experts produced two IAEA publications. Information Circular 310, "Guidelines for Mutual Emergency Assistance Arrangements in Connection With a Nuclear Accident or Radiological Emergency" was published in January 1984 and Information Circular 321, "Guidelines on Reportable Events, Integrated Planning and Information Exchange in a Transboundary Release of Radioactive Materials," was published in January 1985.

The NRC has also worked with individual countries in order to develop arrangements for notification and assistance in the event of an accident. Formal program guidance, outlining the scope, application and limits, of the technical cooperation which NRC would provide, upon request, to a foreign regulatory agency, was published in April of last year, and was based upon our experience in establishing two specific arrangements in this field.

The NRC arrangements with the Korean Ministry of Science and Technology and the Taiwan Atomic Energy Council, through the American Institute in Taiwan, include the provision for cooperation during an emergency situation should the need arise at any of their U.S.-supplied facilities. In essence, this assistance would take the form of sharing technical and analytical expertise by telephone and telefacsimile. Our efforts are focused on supplementing domestic expertise, not replacing it. Such preplanning affords the opportunity for the NRC to be in a position, upon request, to provide assistance during the early, most critical phase of an event; without preplanning, cooperation could be inhibited by the constraints of time and distance.

The Republic of Korea and the Taiwan Atomic Energy Council through the American Institute in Taiwan are in the process of providing NRC with emergency plan information and other plant-specific information for the facilities to be covered under these arrangements; this information will be maintained at the NRC Operations Center. Successful tests of communications systems have already been conducted in the case of Korea.

Arrangements with Canada are being completed which will cover threshold and timing of mutual notifications, points of contact, and detailed procedures relating to the implementation of the response to an event. This mechanism for early notification and candid technical discussions has been tested twice; each time during emergency response exercises -- one Canadian and one U.S. Additional notification tests will be conducted as a matter of course during future exercises where either government is participating. Clearly, each country recognizes the imperative of early notification of events which may have a transboundary impact. Similar discussions are envisioned with Mexico as well.

The accident at Chernobyl on April 26, 1986, reaffirmed our concerns for the potential impact of such an event on neighboring countries -- as well as world wide. In the wake of this accident, at the Tokyo Economic Summit in May 1986, the United States proposed, and the other heads of government agreed, that it was important to develop a convention dealing with notification of radiological accidents with potential transboundary consequences.

In May and June 1986, the IAEA Board of Governors decided to convene a representative group of experts in international affairs and nuclear power to draft two conventions -- one dealing with notification and the other dealing with emergency assistance. The subject matter for these conventions would closely parallel the two published IAEA Information Circulars. From July 21 to August 15, 1986, experts from 62 countries met in Vienna, Austria; that meeting resulted in consensus agreement on texts of the two conventions.

The first convention, on early notification of a radiological accident, requires a country to promptly notify other affected countries and the IAEA of any accident involving certain specified facilities or activities from which a release of radioactive material occurs or is likely to occur and has resulted or may result in an international transboundary release that could be of radiological safety significance for another country. Thereafter, that country -- the one that has suffered a radiological incident -- is required to provide, and update as appropriate, specified basic information about the accident which would be relevant to minimizing the radiological consequences in any affected country.

The convention covers accidents involving both civil and military facilities and activities. The convention does not contain reporting obligations for accidents involving nuclear weapons, nuclear weapon components, capabilities, operations or plans, or plans for testing, storage, transportation or recovery of such weapons or components. However, the convention provides that the notification of any nuclear weapons accident may be made at a nation's discretion with a view to minimizing the radiological consequences.

The second convention establishes a framework under which a country may provide assistance to another country in the event of a radiological accident or emergency. It does not require any country to offer, provide, or accept assistance. This convention deals with such topics as the direction and control of assistance, the establishment of points of contact, and the functions of the IAEA. The convention states that when assistance is provided, wholly or partly on a reimbursement basis, the requesting country incurs a legal obligation to reimburse, but the assisting country may consider waiving reimbursement in certain specified circumstances. Functional privileges and immunities also are provided for the personnel of an assisting country. Finally, the convention provides certain immunities from claims and legal proceedings for an assisting country and its personnel.

In the fall of 1986, the IAEA convened a special session to deal with the two conventions. After several days of discussion the delegates were eager to sign them; over 50 countries signed on the first day alone. It was quite apparent that the spirit of cooperation that existed among the delegates who drafted the conventions continued through this special session.

Lando Zech, the Chairman of the Nuclear Regulatory Commission, attended the IAEA conference. Chairman Zech enthusiastically told an interviewer later, "Everyone seemed pleased to be signing the conventions." Chairman Zech said: "There was a strong feeling of support for the concept of international cooperation in the very important areas of notification and assistance, and those who signed it seemed to convey the impression that they feel they were doing something important not only for their own country but for all the people of the world."

I am not aware of any other related issue that has received such a prompt international response and resolution in such a short period of time. The concept of international cooperation in notification and assistance is an example of how we can successfully work individually and collectively at technical and political levels.

However, much work still needs to be done to effectively implement these conventions. For example, more discussion is necessary to reach an understanding on the threshold for reporting events. While this issue was heavily debated by the experts who drafted the two conventions, a clear resolution was not reached. Additionally, the conventions presume that the IAEA will have an administrative responsibility to assure effective implementation and continued viability. While no new significant role was contemplated for the IAEA, their involvement is an important contribution to success.

The benefits of nuclear technology can be realized by all nations, but we must recognize the continuing need to be vigilant to assure safety in design, construction, and operation. We must be prudent in our actions to protect the health and safety of all peoples. This conference is an appropriate forum to approach these critical subjects.

THE CANADIAN APPROACH TO NUCLEAR POWER SAFETY

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(Article published in the July-August 1983 issue of "Nuclear Safety")

Introduction

The approach to nuclear power safety in Canada has evolved in a continuous manner over almost three decades. From the outset, the safety objective has been to ensure that the risk to the public presented by nuclear power plants is substantially lower than that from alternative sources of electrical energy. Although the expressed criteria have changed somewhat with experience over the years, this basic objective has remained. An underlying principle has been that the licensee (owner/operator) bears the basic responsibility for safety while the regulatory authority (the Atomic Energy Control Board) primarily sets safety objectives and some performance requirements and audits their achievement. As a consequence, regulatory requirements have emphasized numerical safety goals and objectives and minimized specific design or operational rules.

This paper traces the evolution of this approach and its application with some specific examples illustrating not only the overall effectiveness of the approach but also some of the practical difficulties encountered.

The conclusion is one of confidence that the approach to achieving safety of nuclear power plants which has been followed over the years in Canada is both flexible and effective. This approach could be adopted by any country wishing to develop indigenous regulatory rules which could be applicable to more than one design of nuclear power plant.

CANDU Characteristics

Canada has concentrated on heavy water moderated reactors using natural uranium as fuel. The power reactor design employs pressurized heavy water as the coolant, plus pressure-tubes and on-power fuelling. All nuclear power plants built or planned in Canada are of this CANDU-type.

The combination of heavy water and natural uranium tends to result in reactors having relatively high fuel power rating, high flux, and small excess reactivity. The reactivity constraint, coupled with small temperature-reactivity-coefficients, requires constant control and has led to the extensive use of automatic (in recent plants, digital computer) control.

Automatic control relieves the operator of the need to make quick decisions under stressful conditions. Adjustments required by transient conditions are made automatically by the regulating system which can also bring the plant from shutdown to the demanded power at a safe and controlled rate without intervention by the operator. The operator is, therefore, free to make full use of his diagnostic abilities. As a corollary, the training of operating staff has emphasized a sound understanding of the principles involved.

The pressure-tube design presents some safety considerations which are different from those of other designs while obviating any concern about reactor pressure vessel failure. These include such factors as the heat-sink capacity of the moderator, flow stability questions, and the possibility of the fuel coming into contact with the pressure boundary, all of which bear on the requirements for emergency core cooling systems.

The safety characteristics of the CANDU design have had, inevitably, an influence on the safety criteria developed by the AECB although the safety criteria have, in turn, strongly influenced the design.

Evolution of Approach

A serious accident to the NRX research reactor at Chalk River in 1952 was the catalyst for much of the Canadian reactor safety approach which still prevails today. The essential principles which evolved were derived from the recognition that even well-designed and built systems fail and, therefore, there was a need for separate, independent safety systems which could be tested periodically to demonstrate their availability.

In 1957, a paper by E. Siddall (which had an extended foreward by W.B. Lewis), proposed setting safety standards for nuclear power plants by comparing their economic and accidental death consequences with those of the coal-fired power plants to be displaced. This approach was taken for the design of the small Nuclear Power Demonstration (NPD), Canada's first nuclear power plant which began operation in 1962. The target proposed for NPD from the above approach was a frequency of 10^{-5} per year for serious accidents, based upon an overall risk of 1 death per 100 reactor years (10^{-2} deaths/year).

Concurrently, G.C. Laurence, who had been named chairman of the Reactor Safety Advisory Committee (RSAC) which the AECB had created in 1956, also proposed, on similar arguments, that the likelihood of a "disastrous" accident at a nuclear power reactor should be less than 10^{-5} per year. Laurence further proposed that this target could be achieved with realistic designs if there was adequate separation between the operating equipment, the protective devices, and the containment provisions. On this basis, he proposed that the rate of failure of equipment that could lead to a serious release of fission products should be less than 10^{-1} per year and the probability that the protective devices would be inoperative or the containment provisions ineffective should be each less than 10^{-2}:

In the mid-1960's, these concepts were formalized for the first time into a set of criteria commonly called the Siting Guide. These criteria were based on the separation of plant systems into two categories: the "process", or normally operating equipment; and what later came to be known as the "special safety systems", designed to prevent or mitigate the consequences of failures of the process systems. The "special safety systems" include the reactor shutdown systems, emergency core cooling systems, and the containment provisions.

The basic requirements set limits on the frequency of "serious process failures"*/ and on the unavailability of the "special safety systems". They further stipulated maximum values for the calculated dose of ionizing radiation to members of the public for any serious process failure ("single failure"), and for any combination of a serious process failure and failure of a special safety system ("dual failure"). A corollary is that the special safety systems must be sufficiently separate and independent of the process systems and of each other that the likelihood of a "cross-linked" failure will be less than that calculated for coincident events (dual failure).

Although the "single failure"/"dual failure" approach, as practised, adequately defined the required effectiveness of the special safety systems, some concerns in coverage became evident.

These concerns pointed to a need for a more comprehensive approach to safety evaluation. This was identified not only by staff of the utilities and of the AECB but also by advisory groups set up by the AECB.

In 1975, the designers proposed using a "safety design matrix" (SDM) to deal with matters of interdependency and longer-term actions requiring operator intervention. In its present form, the SDM is a record of a systematic "what-if" investigation. The analyst selects an event which is a potential safety concern, and the possible causes of this event are identified by a fault tree analysis. Various postulated consequences are then represented by event sequence diagrams accompanied by a narrative.

The use of SDM's has contributed significantly to a better understanding of system behaviour and system interactions under abnormal operating conditions, and has the potential to identify proper operator actions, desirable design modifications, and, in certain cases, contradictory design requirements. It still depends, however, on visual inspection by the analyst for identifying interdependencies between systems. Nevertheless, it is currently a major tool used for accident analysis.

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*/ A "serious process failure" is a failure of a process system or equipment that, in the absence of special safety system action, could lead to fuel failure or the release of radioactive material to the environment.

Standards

The AECB has issued only a few regulatory documents related to nuclear power plants. Three proposed regulatory guides have been produced covering the special requirements for the three main safety systems: shutdown system, emergency core cooling system, and containment.

The policy has been that while written statements concerning some basic regulatory requirements are necessary and proper for nuclear power plant design, construction and operation, the establishment of detailed requirements should be handled in other ways. Two methods have developed. The first is a long-standing one which reflects the principle that the primary responsibility for safety rests with the licensee. Nuclear power plant designers have been allowed a very substantial degree of freedom to design plants to meet the basic regulatory criteria. The designs are then submitted to the AECB for approval. This approach has led to the gradual establishment of acceptable safety-related design features. While these features are not formally identified as requirements, AECB staff keep them very much in mind in reviewing each new plant design and further discussions are held with the designers if the features are not in evidence.

The second way of establishing detailed requirements is the more traditional one of developing consensus nuclear standards for particular topics. Such standards are produced in Canada by the Canadian Standards Association (CSA).

Summary and Discussion

With the lessons learned from the 1952 accident to the NRX research reactor vivid in the minds of many, the approach to power reactor safety in Canada embodied numerical safety goals from the outset. While the objective was to limit risk to a defined value, the analytical tools were not available to demonstrate compliance with the objective. Consequently, a simplified approach was adopted in the mid-1960's.

This approach (single/dual failure) was first used in the design and safety evaluation of the Pickering 'A' Generating Station and has continued to evolve since that time. A comparison of the operation of reactors against these design requirements confirms that the approach has been sound, and that only evolutionary, rather than revolutionary, changes were required. The frequency of serious process failures has been consistent with early predictions. Some shortcomings in the availability of special safety systems have been encountered but the necessary corrective actions have been taken to meet the numerical safety goals.

In the process of applying the single/dual failure approach, a number of additional requirements related to reliability objectives have been adopted, e.g. any serious process failure should be detected by two diverse parameters. The need for, or adequacy of, such requirements cannot be rigorously defended in the absence of appropriate component failure data and comprehensive probabilistic risk assessments. However, since adequate tools for doing such assessments are not yet in common use, such requirements will remain. It is, nonetheless, an objective in Canada to improve the capability to do probabilistic safety evaluations. The primary purpose for using fault trees and event trees at the present time is to aid the design and decision-making process. In the longer term, as analytical capabilities and the data bases improve (particularly for the effects of human intervention), it will be possible to assess better the risk posed by nuclear power plants. This will permit a better comparison with the numerical safety goals adopted almost three decades ago in the Canadian risk philosophy.

ROLE OF THE REGULATORY AUTHORITY IN ASSURING NUCLEAR SAFETY

Remarks by Chairman Lando W. Zech, Jr.
U.S. Nuclear Regulatory Commission

(United States of America)

It is indeed a privilege and an honor to address this distinguished gathering. Certainly, it is timely and appropriate that international attention be focussed on the peaceful and safe uses of nuclear energy. The U.S. Nuclear Regulatory Commission exists to regulate the peaceful or commercial uses of nuclear energy and, in doing so, protects the public health, safety and the environment.

Today the knowledge and tools of nuclear science are routinely put to use in the educational, industrial, medical, and commercial activities of most countries around the world.

I believe that nuclear technology will only continue to be accepted by the public if there is general confidence that the technology can be safely managed on a world-wide basis. There are new demands to limit or prohibit nuclear power, if its safety cannot be assured. Vigorous steps are being taken by individual nations, the international community, the IAEA and elsewhere, to promote more effective safety measures, to clarify national obligations to report future accidents promptly to neighboring countries, and to provide requested emergency assistance.

During my visits to nuclear power plants, it has been clear to me that if there is a significant key to success in any nuclear reactor operation, that key is strong and competent leadership -- senior management involvement. People are the key to achieving excellence. There must be a competent management organization where authority, responsibility and accountability are clearly in evidence and fully understood. The organization must operate in a professional manner where discipline, attention to detail in job performance and a degree of formality are important elements of the plant operations' entire fabric.

In my opinion, the management challenge for those of us with nuclear safety responsibilities is to see to it that this dedication to excellence does not just exist at the top of the nuclear safety organizations but is communicated and understood by all of those involved.

Management authority, responsibility, and accountability are the critical factors in achieving excellence in safety. Reliable and safe nuclear power plant operation is achieved through competent performance. There is no substitute for hard work and a dedication to quality and all-round excellence.

I believe that the key people involved in nuclear power are the licensed plant operators. The people that stand control room watches at nuclear power plants. These are the people that must be able to operate the plants safely. When anything acts in a way that could potentially compromise safe operations, the plant operators are the ones that must take correct actions to maintain plant safety.

I learned from my experience in the Navy that when you go to sea, every member of the crew must know how to perform the tasks assigned to him. The crew must have the necessary skills and training or the ship will not be able to perform its mission. They must "know their stuff" individually and they must work together as a team.

I have visited 75 nuclear power plants in the United States. I have also visted a number of foreign reactors in various countries. During my visits, I have met many control room crews. In general, I have found them to be a well-qualified group who are dedicated to safe plant operations. I know that other countries are working to achieve excellance in reactor operator qualifications. The United States has done a good job of selecting and training the operators. Nonetheless, I believe that there is room for improvement in the operation of our nuclear power plants both in the U.S. and around the world. In my opinion, we should all commit ourselves to the highest standards of plant operational performance.

Since the United States has a nuclear power industry that is not standardized, there are differences among control room layouts and operational features. Thus, there are necessarily differences in training requirements. Plant specific simulator training can play a key role in preparing operators to meet the challenge presented by their particular facilities. One of the ways these individual and team skills for reactor operations can be improved is through simulator training. In my opinion, a simulator is one of the best investments a utility can make to assure safe plant operations.

I realize that some representatives at this international conference are concerned with the research, medical and industrial applications of nuclear technology as well as power reactor activities. So am I. The safe handling and use of nuclear material by universities, hospitals, doctors and industrial workers are important aspects of any integrated regulatory program.

Moreover, radiation sources for these purposes are part of international commerce sometimes requiring cooperation across borders in cases of lost or contaminated shipments or improper uses which pose potential public health problems. As in the case of reactor regulation, public confidence depends on competent national regulations and effective international cooperation to avoid serious risks and to deal effectively with problems which arise.

Since the IAEA was created in 1957, it has performed an important function in relation to the non-proliferation of nuclear weapons. It has also assisted countries in developing programs for peaceful uses of nuclear energy. Technical cooperation activities include advisory missions, conferences, training seminars and fellowships. These activities receive broad support from the member countries.

The IAEA has developed a series of nuclear safety documents (5 Codes of Practice and 55 Safety Guides) to help countries in establishing internationally acceptable safety codes and guides for use in regulating their nuclear power programs. Some countries would like to make the IAEA Codes and Guides mandatory on an international basis. Others prefer to emphasize strong national regulations which may exceed the basic IAEA safety guidance.

Of the newer activities of the IAEA, several are especially valuable in my view because they involve a sharing of operational experience aimed at upgrading safety practices and results. One involves Operational Safety Review Teams (OSART). If requested, IAEA will send a highly competent, international team to a specific nuclear plant to help assess its operational safety and to recommend improvements. The U.S. has participated in several OSARTs by sending technical people to be a part of these teams. The U.S. has requested an OSART visit to one of our nuclear power plants. It is expected this visit will take place sometime in the next year.

It continues to be vitally important that countries with operating power reactors share information on incidents and accidents involving their operating reactors. By fully sharing the technical details of these incidents, other countries are alerted to potential problems with their own reactors and can take appropriate action. The U.S. exchanges such information bilaterally with over 20 countries and participates in the multilateral Incident Reporting Systems of both the Organization for Economic Cooperation and Development's Nuclear Energy Agency and the IAEA.

I would also note here, my belief that the future of the United States nuclear industry would be well served by adopting standardized plant designs. In that connection, I believe that we in the United States could benefit from the experiences with standardization in other countries. Standardization would bring about discipline in design, construction, operation, maintenance and training, as well as in plant performance and management. Standardization should take into account the lessons we have learned in almost three decades of development of commercial nuclear power. It could facilitate expeditious licensing and improved plant operational safety.

I would like to close by reaffirming my strong belief that people are the essential factor in the safe use of nuclear energy. People design, construct and operate power plants - people administer medical treatments - they use radioisotopes for a wide variety of industrial applications. Leadership and management involvement is crucial in achieving the desired results. Management authority, responsibility, and accountability focused on competent performance are critical factors in attaining safety and operational excellence.

Nuclear energy for peaceful purposes can contribute to a better world. Safety can be achieved by knowledgeable, competent people.

Region or country	Project No.	Title	Amount (thousands of United States dollars)
Uruguay	URU/72/001	Assistance to the Institute of Oenology	16
Uruguay	URU/73/003	Estudio de factibilidad de un centro nuclear	31
Venezuela	VEN/72/005	Radiation chemistry	20
Venezuela	VEN/72/016	Pilot plant for production of wood plastics	92
Venezuela	VEN/75/003	Applicación de técnicas nucleares a la agricultura	4
Regional	RLA/71/801	Eradication in Central America of the Mediterranean fruit fly	51
Regional	RLA/72/019	Regional seminar on the use of isotope techniques in water resources inventory, planning and development	13
Regional	RLA/73/038	Advanced training course on radiological health and safety measures	19
Regional	RLA/73/039	Training course on radiommunoassay procedures	34
Regional	RLA/83/T01	Regional non-destructive testing network for Latin America and the Caribbean	1 585
Total	61 projects		24 187
Interregional and global projects			
Interregional	INT/70/802	Training course on maintenance of nuclear electronic equipment	72
	INT/70/802	Training course in preparation and control of radio-pharmaceuticals	27
	INT/70/811	Training course on radiation dosimetry	22
	INT/71/807	Training course on the maintenance and repair of nuclear electronic equipment	50
	INT/72/027	Training course on the use and maintenance of nuclear and related electronic equipment	226

Region or country	Project No.	Title	Amount (thousands of United States dollars)
Interregional (cont.)	INT/73/018	Interregional seminar on the preparation and implementation of nuclear power plants	43
	INT/73/019	Training course on uranium geothermal prospecting methods	79
	INT/81/T04	Applications of modern techniques in physics to development	332
	INT/73/901	Control of Rift Valley fever	17
Total	9 projects		868
GRAND TOTAL	213 PROJECTS		60 124