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NUCLEAR ENERGY AGENCY
COMMITTEE ON NUCLEAR REGULATORY ACTIVITIES

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PROCEEDINGS OF THE CNRA WORKSHOP ON NEW REACTOR SITING, LICENSING AND CONSTRUCTION EXPERIENCE

Hosted by the State Office for Nuclear Safety (SUJB)
Prague, Czech Republic
15-17 September 2010

The enclosed CD-ROM contains all the presentations

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"The Committee on Nuclear Regulatory Activities (CNRA) shall be responsible for the programme of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. The Committee shall constitute a forum for the effective exchange of safety-relevant information and experience among regulatory organisations. To the extent appropriate, the Committee shall review developments which could affect regulatory requirements with the objective of providing members with an understanding of the motivation for new regulatory requirements under consideration and an opportunity to offer suggestions that might improve them and assist in the development of a common understanding among member countries. In particular it shall review current management strategies and safety management practices and operating experiences at nuclear facilities with a view to disseminating lessons learnt. In accordance with the NEA Strategic Plan for 2011-2016 and the Joint CSNI/CNRA Strategic Plan and Mandates for 2011-2016, the Committee shall promote co-operation among member countries to use the feedback from experience to develop measures to ensure high standards of safety, to further enhance efficiency and effectiveness in the regulatory process and to maintain adequate infrastructure and competence in the nuclear safety field.

The Committee shall promote transparency of nuclear safety work and open public communication. The Committee shall maintain an oversight of all NEA work that may impinge on the development of effective and efficient regulation.

The Committee shall focus primarily on the regulatory aspects of existing power reactors, other nuclear installations and the construction of new power reactors; it may also consider the regulatory implications of new designs of power reactors and other types of nuclear installations. Furthermore it shall examine any other matters referred to it by the Steering Committee. The Committee shall collaborate with, and assist, as appropriate, other international organisations for co-operation among regulators and consider, upon request, issues raised by these organisations. The Committee shall organise its own activities. It may sponsor specialist meetings and working groups to further its objectives.

In implementing its programme the Committee shall establish co-operative mechanisms with the Committee on the Safety of Nuclear Installations in order to work with that Committee on matters of common interest, avoiding unnecessary duplications. The Committee shall also co-operate with the Committee on Radiation Protection and Public Health and the Radioactive Waste Management Committee on matters of common interest."

FOREWORD

The Committee on Nuclear Regulatory Activities (CNRA), based on the regulatory actions underway or being considered in different members countries concerning the design and construction of advanced nuclear power plants, established a working group responsible of the regulatory issues of siting, licensing and regulatory oversight of generation III+ and generation IV nuclear reactors. The Working Group on the Regulation of New Reactors (WGRNR) main purposes are to improve regulatory reviews by comparing practices in member countries; improve the licensing process of new reactors by learning from best practices in member countries; ensure that construction inspection issues and construction experience is shared; promote cooperation among member countries to improve safety; and enhance the effectiveness and efficiency of the regulatory process.

The WGRNR has established an initial programme of work which includes: the collection of construction experience and the assessing of the information collected in order to share lessons learned and good practices; the review of regulatory practices concerning the regulation of nuclear sites selection and preparation; and the review of recent regulatory experience concerning the licensing structure of regulatory staff and regulatory licensing process.

The WGRNR organised an international workshop aimed to review and discuss recent and past construction experience lessons learned including perspectives from regulatory authorities, vendors, and licensee; issues associated with project management resources including: a) overall human resources, expertise, experience and organisation available to the licensee, b) capability of each potential vendor (in-house knowledge and skills vs planned subcontracting and subcontractor management). The workshop was intended to also discuss the lessons learned in the regulation of site selection, evaluation and site preparation as well as the review of regulatory practices for the licensing of new reactors, including the regulatory body infrastructure, staffing and expertise needed.

The workshop, held in September 15-17, 2010, in Prague, Czech Republic, hosted by the State Office for Nuclear Safety, provided a forum to communicate recent experience on these topics to a wider audience, to introduce and discuss the current programme of work and products under development in WGRNR, and to gain insights from workshop participants on each of the programme of work areas, and get feedback from participants on additional focus areas. This report documents the proceedings of the workshop.

ACKNOWLEDGMENTS

Gratitude is expressed to the State Office for Nuclear Safety (SUJB) in Czech Republic for hosting the Workshop. To Ms. Dana Drabova, Chairperson of SUJB, for her support and dedicated efforts for the organisation of the workshop, and to Mr. Jan Štuller, who co-ordinate all the organisational efforts.
Special appreciation is given to Ms. Laura Dudes, WGRNR Chairperson, and Ms. Rosa Sardella, WGRNR Vice-Chair for their effort and cooperation. Thanks are also expressed to the Workshop Organising Committee, Session Chairpersons and workshop participants for their support.

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SUMMARY AND CONCLUSIONS

1. Introduction

This report documents the proceedings from the "Workshop on New Reactor Siting, Licensing and Construction Experience", held in Prague, Czech Republic on 15-17 September 2010. A total of 59 specialists from 16 countries and international organisations attended. The Meeting was sponsored by the OECD Nuclear Energy Agency Committee on Nuclear Regulatory Activities and hosted by the State Office for Nuclear Safety (SUJB) in Czech Republic.

The objectives of the workshop were to review and discuss recent and past construction experience lessons learned including perspectives from regulatory authorities, as well as vendors, and licensee. The workshop addressed issues associated with project management resources including: a) overall human resources, expertise, experience and organisation available to the licensee, b) capability of each potential vendor (in-house knowledge and skills versus planned subcontracting and subcontractor management). The workshop also discussed the lessons learned in the regulation of site selection, evaluation and site preparation as well as the review of regulatory practices for the licensing of new reactors, including the regulatory body infrastructure, staffing and expertise needed.

The workshop provided an excellent opportunity to communicate recent experience on these topics to a wider audience, including participants from OECD member countries as well as New Entrants from non-OECD member countries. The workshop allowed the WGRNR group to introduce and discuss the current programme of work and products under development in order to gain insights from workshop participants on each of the programme of work areas, and get feedback on additional focus areas.

The workshop was structured in 4 technical sessions, each followed by ample time for panel discussions. The first technical session was devoted to presentations of the licensing process for new reactors followed by different member countries. The second technical session was intended to discuss and share construction experiences issues and lessons learned from construction projects. The third technical session addressed the human and organisational issues associated with the licensing process of new reactors. And the fourth technical session included presentations on the issues, practices and studies conducted to define the sites for new nuclear power plants.

2. Background of the Workshop

Based on the regulatory actions underway or being considered in different member countries concerning the design and construction of advanced nuclear power plants, the NEA's Committee on Nuclear Regulatory Activities established in 2008 a working group responsible of the regulatory issues of the siting, licensing and regulatory oversight of generation III+ and generation IV nuclear reactors. The working group on the regulation of new reactors (WGRNR) constitutes a forum of experts for the licensing of new and advance commercial nuclear power reactors and should facilitate a cooperative approach to identify key new regulatory issues and promote a common resolution.

The main purpose of the WGRNR and its products are to improve regulatory reviews by comparing practices in member countries; improve the licensing process of new reactors by learning from best practices in member countries; ensure that construction inspection issues and construction experience is shared; promote cooperation among member countries to improve safety; and enhance the effectiveness and efficiency of the regulatory process.

WGRNR has established an initial programme of work which was approved by the CNRA. The initial programme of work includes the collection of construction experience and the assessing of the information collected in order to share the lessons learned and good practices, the review of regulatory practices concerning the regulation of nuclear sites selection and preparation, and the review of recent regulatory experience concerning the licensing structure of regulatory staff and regulatory licensing process.

3. Summary and Conclusions

The workshop was opened by a presentation from the Chairperson of the State Office for Nuclear Safety of the Czech Republich, Dana Drabova, who provided the workshop participants with a brief history of nuclear power and the future in the Czech Republic. She also shared her insights into the importance of harmonisation of standards and practices, and cooperation between regulatory authorities.

New reactor licensing process
This session focused on the challenges and successes that have been predominant in most of the current licensing processes. These included establishing a qualified and effective infrastructure as well as coping with level-of-detail during the different stages of licensing. The panel discussed recent experiences in the review of new applications, as well as the prospect and limits of standardisation.

The presentation topics for the first session included overviews of the regulatory processes in countries with mature regulatory programs, the challenges faced by New Entrants as they develop their regulatory policies and programs, insights on standardization of designs as it relates to regulatory policies, and technical insights into the regulatory licensing process.

The session started with a presentation of the WGRNR on its efforts to develop a report that will describe the regulatory structure, licensing processes, and resources used by regulators. This report will serve as a guide for regulators starting the review of their first new reactor application and provide a benchmark against which more developed regulators can assess their programs.

The International Atomic Energy Agency made a presentation on its licensing related activities, including the development of safety standards, and the conduction of expert missions and training.

The Multination Design Evaluation Programme (MDEP), an initiative taken by national safety authorities of ten countries to leverage the resources and knowledge for new reactor design reviews, presented their main activities focusing on a different approach for harmonising safety goals.

The Finish safety authority (STUK) made a presentation on the licensing process in Finland focusing on the three licensing steps: Decision in Principle, Construction Licensee and Operating Licensee. The presentation included the main parties involved in the process and the kind of information submitted to STUK for their assessment.

The Slovenian Nuclear Safety Administration (SNSA) discussed the preparation of the regulatory infrastructure for the new nuclear build, including the changes in the legal and regulatory framework, the analysis of human resources and external expert support needed.

The Canadian Nuclear Safety Commission (CNSC) discussed the licensing of new nuclear power plants in Canada, including a brief overview of the CNSC's regulatory philosophy, the different stages of the licensing process and activities of the CNSC to prepare for the assessment of the licence application and regulatory oversight.

The Korea Institute of Nuclear Safety (KINS) made a presentation on the licensing experience of the new APR1400 reactor in Korea. The different steps of the licensing process were discussed along with the main design features of the APR1400 and the standard design approval and safety review of Shin-Kori units 3 and 4.

Atomic Energy of Canada Limited (AECL) presented the pre-project regulatory reviews of the ACR-1000 and EC6 designs conducted by the CNSC.

The Federal Authority for Nuclear Regulation (FANR) discussed the development of the national nuclear regulations in the United Arab Emirates, noting that the two steps licensing process, Construction and Operation, will be applied to the four Korean APR 1400 reactors to be constructed in UAE.

Industry was represented in the meeting through the World Nuclear Association (WNA), which discussed the efforts of its working group on cooperation in reactor design evaluation and licensing (CORDEL) in promoting international standardization of reactor designs. CORDEL proposed three phases to achieve the standardisation. These phases range from sharing designs reviews and assessment, validating and accepting designs approvals to issuing international designs certifications.

The last presentation in this session was given by the University of Pisa on the Best Estimated Plus Uncertainty (BEPU) approach used to perform accident analysis for the licensing process of the Atucha-2 NPP in Argentina.

A panel discussion on the licensing process took place with the following highlights:

- It was noted that international organisations and initiatives such as NEA, IAEA and MDEP provide forums for international collaboration, developing guidance documents and sharing regulatory information.

- In general it was agreed that National Authorities have varied level of government consent points in the licensing process. However the design and construction reviews all focus on a high level of quality and management oversight by the operator. Licensees are primarily responsible for all aspects of safety for the plant, and safety starts with the design developed by a NSSS vendor.

- The Regulators have its role in assuring safety and are compelled to make an independent assessment of the design basis safety features of the NPP.

- Workshop participants agreed on the benefits from international cooperation as well as from bilateral and multilateral agreements.

- Pre-project regulatory activities, such as development of legal and regulatory framework, establishing appropriate infrastructure frame for licensing process, identification of resources needed, etc. are essential for assuring nuclear safety through the all phases of a nuclear power programme.

- Participants agreed on the standardization of reactor designs and noted that initiatives such as MDEP are seeking for such harmonisation. However it was noted the need to take into account specificities.

- Recent experiences shows that even for countries with mature nuclear power programs there are challenges to license and construct new nuclear power plants. For New Entrants the challenges faced will potentially be more significant. New safety issues raised from new designs, methodologies, and analysis.

Construction experience
This session was aimed to present the industry and regulatory perspectives on the approaches to share lessons learned from past and actual construction experience, to implement measures aimed to detect and correct such events and prevent them from remaining undetected until the plant becomes operational. Topics discussed focused on design and construction activities and accomplishments, and views on recommendations to achieve positive results of co-operation with the regulatory authorities and to promote international co-operation between licensees or designers.

STUK made a presentation on the experiences of the Olkiluoto 3 project (OL3), a turnkey project that it is progressing but with more than three years behind of its original schedule. Some of the issues discussed included the completion and management of the plant's design, the role of quality management, the licensee's role in turnkey projects, and the safety culture in a construction project.

Mitsubishi Heavy Industries, LTD gave a presentation sharing the experiences with Tomari-3 construction. It was described the vast experiences of Mitsubishi due to the continuous construction of 24 PWRs in Japan. The construction and plant specific aspects were discussed emphasizing the importance of modularisation.

US Nuclear Regulatory Commission presented its programme to collect and share lessons learned from past and ongoing construction projects. The programme focuses on new reactor designs and construction, assessing the adequacy of current regulatory activities or the need for enhancement.

The Nuclear Safety and Radioprotection Institute (IRSN) gave a presentation on the safety assessments and on-site inspections of the Flamanville 3 EPR project in support of the French Safety Authority. The main aspects of the civil design safety assessment were discussed along with examples of the technical issues highlighted by the surveillance of the licensee on its contractors or during the on-site inspection programme carried out by ASN with IRSN's support.

Electricité de France (EDF) presented the industry point of view regarding the experience feedback of the Flamanville 3 project. The project organisation was described noting that EDF acts as architect engineer and that the number of main contracts has been reduced from previous projects. It was noted that the good experience feedback on several construction practices have been already transferred to other EDF projects as Taishan EPR project in China.

The last presentation of this session described the analysis conducted by the European Clearinghouse at the Joint Research Centre of worldwide events related to construction and commissioning of nuclear power plants. The main methodological issues were discussed along with the main trending results and lessons learned.

A panel discussion on the construction experience feedback took place with the following highlights:

- New build is very demanding and challenging

  - Construction of a nuclear power plant is a complex project and requires that contractors, designers, licensees and regulators nuclear specific know-how. This is true even for countries with mature nuclear power programs especially when they have stopped new build for decades.
  - Much of the earlier experience and resources have been lost in Europe.
  - Good preparation to first project in a country is essential - regulatory framework needs to be clearly understood by parties (regulator, license applicant, vendors).
  - Lessons learned from the past should allow avoiding occurrence of events.
  - Construction experience sharing is a leverage for quality and so for a future safe operation of NPPs.

- Workshop participants agreed on the need to promote a large spread of construction experience information and analysis.
  - Regulatory Authorities: WGRNR ConEx Program, EU Clearinghouse reports, US Licensee Event Reports (LERs) analysis.
  - Licensees/Vendors should organize their construction experience sharing and international co-operation with a framework allowing overriding commercial interest.

Human and organisational issues
Session three of the workshop focused on human and organisational issues. The topics presented included managing safety in a new build, human and organisational factors in licensing, developing licensee organisational capability, and lessons learned from current construction projects related to human and organisational factors for new reactors.

Advanced Systems Technology and Management, Inc made a presentation on the US NRC International Regulatory Development Partnership (IRDP) which is aimed to provide assistance to countries with emerging nuclear power programs to develop a regulatory agency capable of licensing and regulatory oversight for operation of a new reactor.

The Swiss Federal Nuclear Safety Inspectorate (ENSI) presented the Swiss approach for dealing with human and organisational factors (HOF) in the licensing process for NPPs. It was noted that ENSI requires from applicants that the HOF be adequately considered from the beginning and in all phases of the project of building a new NPP.

The French Safety Authority (ASN) gave a presentation on the EPR Flamanville 3 experiences in assessing human and organisational factors (HOF). It was noted the assessment of HOF in the safety options assessment and in preliminary safety case assessment before the Authorisation Decree for the Flamanville 3 creation (April 2007) as well as the assessment of HOF before ASN decision for commissioning. Some main first lessons to learn from assessing HOF were discussed.

A panel discussion on the importance to consider human and organisational factors in all phases of a nuclear power plant took place with the following highlights:

- Considerations of HOF need to start from the beginning of the project and develop in a graded way.
- The applicant licensee must demonstrate the capability to operate the facility safely for its full lifetime.
- It is important that the licensee understands the design: use of a “design authority” INSAG 16.
- The licensee must take responsibility; the process of handover from vendor/designer is a key activity.
- The Regulator needs to be able to assess HOF processes and their development.

Siting practices and issues

The fourth and final technical session focused on sharing regulatory authorities, reactor designers, and operators/ licensees perspectives on the various practices used in the regulation of nuclear power plant siting (selection, evaluation and site preparation). This session was also aimed to address issues on sites where a mixture of activities are taking place (e.g., operating units, new construction, decommissioning, etc.) including organisation of the regulators and licensee/engineering organisation, methods, systems, etc.

The WGRNR presented to the workshop participants the results and conclusions of the survey on the regulation of site selection and preparation that were documented in the report NEA/CNRA/R(2010)3.

US NRC discussed insights from siting new nuclear power plants in the central and eastern United States. Activities and issues related to the safety portion of the siting review were discussed, noting that pre-application interactions with the applicant are important especially in sites with unique hydrological, geotechnical or seismic aspects. The environmental review process was presented and the role of public stakeholders was stressed.

Korea Institute of Nuclear Safety (KINS) presented the regulatory framework and issues on the site selection and site evaluation for Korean NPPs. The licensing process for new reactors was presented with emphasis on the early site approval and construction permit. Finally some recent siting-related issues experienced in Korea were discussed.

CNSC gave a presentation on the siting practices and site licensing process for new reactors in Canada. It was noted that site evaluation and site selection is not federally regulated and is performed by the proponent. However the proponent needs to provide certain amount of design information to demonstrate site suitability in both the environmental assessment and licensing processes. Finally the process to grant the licence to prepare the site was discussed.

FANR made a presentation on the potential nuclear power plant siting issues in the United Arab Emirates. The licensing approach was discussed as well as some of the siting issues that are to be dealt with in the design of a NPP in the UAE. Dust/sandstorm, high cooling water temperature, and Sabkha (or Evaporites) ground soil are site-specific characteristics of the region of the proposed NPP.

National Nuclear Energy Agency of Indonesia (BATAN) presented the NPP siting analysis in Western part of Java island Indonesia. The different steps of the site selection process were presented along with details of the regional studies to determine potential sites. The natural external event and man-induced events were analysed resulting in the identification of 2 potential sites.

CEZ, a.s. made a presentation on the Temeling 3 and 4 siting. It was noted the plans to build two more units in the existing site of Temelin as it was the original intension, one more unit in Dukovany and 1 or 2 in Bohunice in agreement with the Slovak government. As part of the licensing process the environmental impact assessment (EIA) was prepared and submitted. Since design is not selected, siting approval as well as EIA process is executed for envelope of PWR generation III, III+.

Energopruzkum Praha Ltd made a presentation on the near regional and site investigations of the Temelin NPP Site. It was noted that although the Temelin site is situated in the area with low seismicity, item of seismicity is a basic argument against Temelin NPP and therefore a detail seismic hazard assessment was performed.

Atomic Energy Licensing Board (AELB) gave a presentation on the regulatory issues and challenges in preparing for the regulation of new reactor siting in Malaysia. The legal and regulatory framework was presented along with the three steps licensing process. The main regulatory issues and challenges were discussed.
A panel discussion on the siting issues, and challenges took place with the following highlights:

- Social acceptability of the new build site is very important. Although political will is needed to start the process, consultation with the public and with public interest groups is needed to obtain their acceptance before a project should proceed.

- Site characterization must be comprehensive for a new site, though for an existing site, an update of previous studies may suffice to demonstrate suitability.

- Some states use a plant parameter envelope in siting; others a site envelope in a design review.

- Multiple permits are required in most, if not all, jurisdictions. Co-ordination between the permitting agencies would simplify data submission by the applicant and avoid duplication or overlap of jurisdiction or requirements.

- An environmental assessment is required in all nation states. Geology, hydrology and meteorology are the standard topics to be addressed, though the manner in which they are addressed varies considerably due to the wide variation in local conditions.

- Interaction and co-ordination between established nuclear states and newcomer states will promote safe new build. For the nuclear safety regulator, this should cover guidance for applicants and procedures and criteria for review of submissions. IAEA Requirements and Guidance documents should play a large role.

Conclusions
In general workshop participants agreed on the need to regularly have these kind of forums to discuss relevant regulatory issues for new builds. One important aspect of this workshop was the participation of “New Entrants”. The interaction between NEA member countries with mature nuclear power plants and newcomers was quite important since it gave newcomers the possibility to benefit of mature international practices in order to focus their regulatory oversight and control. NEA members could also benefit from insights the New Entrants discover as they develop or enhance their regulatory controls. In addition technical exchanges associated with construction experience of New Entrants as they begin to license, build and operate NPP could benefit NEA members.

It is recommended that the WGRNR convenes a second conference in two years time (2012).

Workshop on "New Reactor Siting, Licensing and Construction Experience"

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WGRNR Efforts in Understanding the Different Process and Practices Used by the Regulatory Agencies

Eduardo Sastre, Laura Dudes (NRC, USA)

The Committee on Nuclear Regulatory Activities (CNRA) of the OECD Nuclear Energy Agency (NEA) is an international committee composed primarily of senior nuclear regulators. It was set up in 1989 as a forum for the exchange of information and experience among regulatory organisations and for the review of developments which could affect regulatory requirements. The Committee is responsible for the NEA programme concerning the regulation, licensing and inspection of nuclear installations. In particular, the Committee reviews current practices and operating experience.

The CNRA created the Working Group on the Regulation of New Reactors (Working Group) at the Bureau meeting of December 2007. Their Mandate was to "be responsible for the programme of work in the CNRA dealing with regulatory activities in the primary programme areas of siting, licensing and oversight for new commercial nuclear power reactors (Generation III+ and Generation IV)"

At its second meeting in 2008, the Working Group agreed on the development of a report based on recent regulatory experiences describing; 1) the licensing structures, 2) the number of regulatory personnel and the skill sets needed to perform reviews, assessments and construction oversight, and 3) types of training needed for these activities. Also the Working Group agreed on the development of a comparison report on the licensing processes for each member state. Following discussion at its meeting in March 2009, the Working Group agreed on combining the reports into one, and developing a survey where each member would provide their input to the completion of the report. Because of the magnitude of this survey, the Working Group decided to divide into 4 parts; general, siting, design and construction. This presentation will focus on the results of the general section. Some of the topics covered in this part are licensing process, governing authorities, legal decision, general level of effort and infrastructure.

In the end, the purpose of the full report is to serve as a guide for regulatory bodies to promote safety through the understanding of different ways to review and approve a new reactor application.

Multinational Design Evaluation Programme (MDEP) – Safety Goals

G J Vaughan, Nuclear installations Inspectorate, Health and Safety Executive, UK, Chairman: MDEP Sub-committee on Safety Goals

One of the aims of the NEA's Multinational Design Evaluation Programme (MDEP) is to work towards greater harmonisation of regulatory requirements. To achieve this aim, it is necessary that there is a degree of convergence on the safety goals that are required to be met by designers and operators. The term "safety goals" is defined to cover all health and safety requirements which must be met: these may be deterministic rules and/or probabilistic targets. They should cover the safety of workers, public and the environment in line with the IAEA's Basic Safety Objective; encompassing safety in normal operation through to severe accidents.

MDEP is also interested in how its work can be extended to future reactors, which may use significantly different technology to the almost ubiquitous LWRs used today and in the next generation, building on the close co-operation within MDEP between the regulators who are currently engaged in constructing or carrying out design reviews on new designs. For two designs this work has involved several regulators sharing their safety assessments and in some cases issuing statements on issues that need to be addressed. Work is also progressing towards joint regulatory position statements on specific assessment areas. Harmonisation of safety goals will enhance the cooperation between regulators as further developments in design and technology occur.

All regulators have safety goals, but these are expressed in many different ways and exercises in comparing them frequently are done at a very low level eg specific temperatures in the reactor vessel of a specific reactor type. The differences in the requirements from different regulators are difficult to resolve as the goals are derived using different principles and assumptions and are often for a specific technology. Therefore a different approach is being investigated, starting with the top-level safety goals and try to derive a structure and means of deriving lower tier goals and/or targets that can be seen to be clearly related to the higher level ones and set consistent requirements for different technologies. MDEP has, therefore, established a sub-committee to carry out this work.

This paper is a review of the work of this sub-committee over the last eighteen months or so in attempting to outline a framework within which potential goals can be included, as a move towards harmonization. If the work is successful, and leads to an agreed MDEP approach, it will greatly assist in the process of harmonisation. It is important to emphasise that this work has not as yet attempted to derive specific safety goals per se, but to derive a framework, which can be used to understand how the deterministic and the probabilistic elements can be integrated in establishing reference level of safety.

Nuclear facilities pose a range of safety issues. These may be due to normal operational exposures of workers and discharges to the public and longer term issues of storage of spent fuel and radioactive waste, but much emphasis has been given to the consequences of accidents, particularly to the public. Determining how to balance the safety measures needed to prevent and protect against this spectrum of

unwanted consequences was difficult and led, in many cases, to overly conservative requirements in some areas, whilst possibly too low a level in others. Whatever, it was difficult to be sure what the levels of safety achieved were and whether they were appropriate to the particular situation. Increasingly, the emphasis has been placed on probabilistic assessments to provide risk estimates to support decisions in a more balanced approach. This approach can also incorporate lessons learned from operating experience and research results: it has been especially true in the development of methods to analyse accidents.
Nuclear safety requirements were developed decades ago using deterministic approaches with a defence-in-depth (DID) philosophy as the foundation of the regulations/requirements. However, some explicit and implicit probabilistic considerations were used: these ranged from splitting the design basis faults into groups according to frequency with different acceptable consequences: the use of engineering safety margins, which had been determined heuristically; to overtly conservative requirements such as the single failure criterion. Different approaches were used in different countries, with some using more risk-based approaches than others, but in all cases, a DID philosophy, centred on several levels of protection including successive barriers and conservative considerations to prevent the release of radioactive material to the environment, was, and still is, employed.

Basically, all countries' fundamental objective is to ensure that nuclear facility operation will not lead to significant additional risk to the health and safety of the public, and will not adversely impact the environment. In considering the acceptability of a nuclear facility in relation to safety, Governments and regulatory bodies define a range of legal, mandatory requirements which are supplemented by regulatory requirements which may not have a mandatory nature.

Occupational and public dose limits during normal operation have been established by all countries, and these generally conform to the IAEA Basic Safety Standard (1), which is derived largely from the ICRP recommendations. In addition, many countries have developed deterministic and probabilistic goals, which are frequently expressed as numbers e.g. doses, frequencies of core damage and release quantities.
As noted in INSAG-25 (2), which considers risk-informed decision making (RiDM), the design basis analysis uses deterministic assumptions by postulating a set of initiating events and scenarios against which the plant has to be protected and designed within the limits of specified acceptance criteria. The likelihood of postulated events is estimated in a conservative manner, and the acceptance criteria allow increasingly severe consequences for accidents of decreasing initiating event frequency. Conservative assumptions are made and scenarios bounding other potential accidents of similar nature are looked for to ensure that the results provide a generally robust protection against radiation hazards and other harmful consequences.

In addition, probabilistic risk targets and Probabilistic Safety Analysis (PSA) are being used by many countries to assist in safety decisions by the nuclear industry and by the safety authorities. Many countries have developed probabilistic safety criteria which have generally been defined as targets or goals i.e. not mandatory. Taken together, these criteria range from worker risk, societal risk, offsite releases, core damage, whilst other, lower level criteria, have been used in various risk-informed applications over the last few decades. Some countries have also developed criteria for long–term restrictions on the use of extensive areas of land and water, should a major accidental release of radioactivity occur.

Although PSA methodology has now reached a relatively mature level of sophistication, and provides an integrated and systematic examination of safety aspects of design and operation that can be used for a range of applications, it is recognized that PSAs do not model all aspects of design and operation. This adds an additional level of uncertainty to the bottom line numerical estimates, which by the nature of the modelling, contain aleatory and epistemic uncertainties. Accordingly, in using calculated numerical results it must be remembered these are risk metrics and care must be taken not to assume that they are accurate measures of risk that can be compared with frequentist data from, say, road traffic accidents.

Nevertheless, integration of deterministic and probabilistic elements, using risk as a decision-making paradigm to determine a balanced approach, can lead to more coherent decisions.

Whilst many countries use PSAs as a supplement to deterministic requirements, some countries use PSAs as a complement to the deterministic requirements. The complementary use of PSAs leads to identifying areas for safety enhancement and also supports flexibility in eliminating requirements if the requirement is shown through probabilistic analyses to have very low risk. The supplementary use of PSAs would allow imposing additional requirements based on the PSA results but not eliminate any deterministic requirements irrespective of the risk.

Although the predominant type of reactor that is being used currently is the LWR (and mostly PWR), there are several other types of reactor in use in various parts of the world: in addition, other types of nuclear facility exist in many countries. Future NPP designs, within the Generation IV ambit, cover many different technologies, some with no obvious parallel with current designs. Hence, to ensure that the safety of the population is consistently achieved, high level safety goals which can be applied to all types and designs need to be defined. Thus, the term safety goals must be recognized as covering the whole range of safety issues that need to be addressed, from normal operation, through to major accidents. From this basis, lower level safety goals can be derived for specific types of facility and design, in a coherent manner.
The safety goals that are enshrined in law or are promulgated by regulators are usually expressed in terms of specific technology and at a fairly detailed level. Frequently they are based on a series of assumptions of modelling and/or data relevant to a particular technology, or even design. Thus, although several surveys have gathered the safety goals from a wide range of countries, it has been difficult to do more than say that they are roughly in line. However, notable differences do arise and without an understanding of the reasons for this – and specifically how these different requirements have arisen - trying to harmonisation them is doomed to failure. The proposal was therefore made to the MDEP Steering Technical Committee that a different approach should be tried leading to the sub-committee being established.

The first task of the sub-committee was to collect from the participating countries, the basic safety requirements – we referred to them as High Level Safety Goals. We also held discussions with WENRA, who through their Reactor Harmonisation Working Group are developing Objectives for New Reactors (3) and, through a member of the sub-committee, with IAEA thinking in this area. In addition meetings with the NEA, CSNI WGRisk group and the Generation IV Reactor Safety Working Group were held.
The conclusions were that most, if not all, countries subscribe to the view that NPP should only add insignificantly to the risks which the population is exposed. In many cases this is based on 1% or 0.1% of risks of death of individuals or cancer, respectively. These considerations should cover normal operational exposures of workers, radiation and radioactivity discharges to the environment as well as accidents. Although many safety goals are based on the effects on individuals, all countries recognise that the consequences of a nuclear accident can affect wider aspects such as effects on use of land or food production. For this reason, countries all propose that, for new reactors, offsite releases of radioactivity should be reduced to a low level (i.e. the ALARP concept).

As noted, all countries utilize a DID concept to make safety decisions. INSAG-12 (4), provides an excellent discussion of the DID concept. This concept is also a prime factor in meeting Principle 8, Prevention of Accidents, in the IAEA Safety Fundamentals (5) and a key attribute of the Design Safety Requirements for NPP (6) and the Safety Guide supporting it. Most, if not all, countries generally follow the IAEA Safety Standards and may also consider other IAEA documents. The IAEA suite of the Safety Standards, from the Fundamentals, through Requirements to the Safety Guides, is an important consensus baseline for moving towards harmonisation.

Overall, the DID philosophy has resulted in excellent safety record of nuclear facilities and the sub-committee proposes that the framework for defining safety goals should be based on a DID structure. An extended DID framework has been devised which encompasses an approach for integrating the probabilistic and the deterministic elements in making safety decisions. This is, in effect, a way of developing, in more detail, the INSAG25 RiDM approach. The subcommittee have proposed a hierarchical set of Safety Goals, each level being linked to the higher level, but giving more detailed requirements. This hierarchical scheme can be extended to detailed technology-based goals and targets. Whilst goals are often qualitative and express what is to be achieved, targets are usually quantitative and are a measure of the achievement.

The main MDEP committees are considering these proposals.
One of the proposals, which is generally agreed, is that the goal, and hence the derived target, for core damage states should provide a higher level of safety than previously. According to a survey by NEA relating to new reactors (7), in general, a core damage frequency target of 10^{-5} per reactor year is being applied by most countries for new LWR-type reactors. Requirements for large offsite releases to be either "practically eliminated" or must be of a low frequency, typically, 10^{-6} per reactor year, are also being applied by many countries. However, it is important to remember that when deriving values for comparison with these targets, the assumptions and models used may be different in different countries. Hence, when comparing analyses with targets it is essential to ensure that the underlying assumptions are consistent.

The sub-committee's work is not finished and it is intended to finalise its report to the main MDEP committee in the autumn of this year. Position Papers on Safety Goals and Integrated Risk-informed Decision Making have also been drafted and await endorsement by MDEP later this year and, then, we hope, will be issued via the website. The future work will not be carried in isolation: it is a feature of the work that we see ourselves as a catalyst to bring together other groups. Over the coming months, subject to MDEP approval, we will work in conjunction with other interested organisations, including those with which we have already opened discussions. The longer term future is to close off the work when we are satisfied that other, more appropriate bodies, especially the IAEA, can take it forward in a suitable fashion.

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Licensing process in Finland

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In accordance with the Nuclear Energy Act, the use of nuclear energy constitutes operations subject to licence. The licensing process and conditions for granting a license is defined in the legislation. The licenses are applied from and granted by the Government. This paper discusses briefly the licensing process in Finland and also the roles and responsibilities of main stakeholders in licensing.

Licensing of a nuclear power plant in Finland has three steps. The first step is the Decision in Principle (DiP). Goal of DiP is to decide whether using nuclear power is for the overall good for the Finnish society. The second step is Construction License (CL) and the goal of CL phase is to determine whether the design of the proposed plant is safe and that the participating organisations are capable of constructing the plant to meet safety goals. The third step is the Operating License (OL) and the goal of the OL phase is to determine whether the plant operates safely and licensee is capable to operate the plant safely.

Main stakeholders in the licensing process in Finland are the utility (licensee) interested in using nuclear power in Finland, Ministry of Employment and the Economy (MEE), Government, Parliament, STUK, the municipality siting the plant and the general public. Government grants all licenses, and Parliament has to ratify Government's Decision in Principle. STUK has to assess the safety of the license applications in each step and give statement to the Ministry. Municipality has to agree to site the plant. Both STUK and the municipality have a veto right in the licensing process.

1. Decision in Principle phase

Before a Construction License for a nuclear power plant, nuclear waste disposal facility, or other significant nuclear facility can be applied, a Decision in principle by the Government is needed. A condition for granting the DiP is that the operation of the facility in question is in line with the overall good for society. Further conditions are that the municipality of the intended site of the nuclear facility is in favour of constructing the facility and no factors indicate a lack of sufficient prerequisites for constructing the facility according to the regulations. The coming into force of the DiP further requires that it will be confirmed by the simple majority of the Parliament. The Parliament cannot make any changes to the Decision, it can only approve it or to reject it as it is.

STUK has to make a preliminary safety assessment of the application. In its safety assessment, STUK states whether any factors have arisen indicating a lack of sufficient prerequisites for constructing a nuclear facility that fulfills Finnish safety requirements. To perform a preliminary safety assessment, descriptions of the plant options, proposed site and applicant's organisation have to be submitted to STUK.
In accordance with the Nuclear Energy Decree, the licence applicant shall also submit an assessment report in accordance with the Act on Environmental Impact Assessment (EIA) Procedure (468/1994) when applying for a Decision in Principle. STUK will provide a statement about the environmental impact assessment programme and the assessment report. Ministry of the Employment and the Economy is the contact authority for EIA process.

2. Construction License phase

The Construction Licence for a nuclear facility shall be applied for from the Government. Application is submitted to the MEE, which asks statements from all stakeholders to prepare the license application for the Government decision.

STUK issues a statement on the application for the Construction Licence. The statement is supplemented with a safety assessment. The preconditions for granting the Construction Licence are defined in Sections 18 and 19 of the Nuclear Energy Act. In its safety assessment, STUK takes a stand on the fulfillment of the requirements laid down in the relevant legislation and YVL Guides regarding the issues STUK is responsible for. When preparing the safety assessment, STUK requests from the Ministry of the Interior a statement on the physical protection and emergency response arrangements.

When applying for the Construction Licence, the documents listed in Section 35 of the Nuclear Energy Decree, and other reports considered necessary by STUK shall be submitted to STUK for approval. STUK issues a statement about the Construction Licence application only after having approved essential parts of each of these documents by a separate decision. Documents to be submitted to STUK for review are among others Preliminary Safety Analysis Report (PSAR), Safety Classification document, quality manual for design and construction, design phase PRA, plans for physical protection and emergency response arrangements, plan for arranging safeguards.

The construction of a nuclear facility shall not begin, as far as the structures affecting nuclear safety are concerned, before the Government has granted the Construction Licence required by the Nuclear Energy Act for the facility. Beginning the formwork and reinforcing work of the safety-classified concrete structures at the site is considered to be construction of this kind. If the manufacture of structures and components for the nuclear facility is begun before the Construction Licence is granted, the licence applicant shall apply for STUK's prior approval for commencing the work.

3. Operating License Phase

The Operating Licence for a nuclear facility shall be applied for from the Government. Application is submitted to the MEE, which asks statements from all stakeholders to prepare the license application for Government decision.

STUK issues a statement on the application for the Operating Licence. The statement is supplemented with a safety assessment. When preparing the safety assessment, STUK requests from the Ministry of the Interior a statement on the physical protection and emergency response arrangements. The preconditions for granting the Operating Licence are defined in Section 20 of the Nuclear Energy Act. In its safety assessment, STUK takes a stand on the fulfilment of the requirements laid down in the relevant legislation and YVL Guides regarding the issues STUK is responsible for.

When applying for the Operating Licence, the documents listed in Section 36 of the Nuclear Energy Decree, and other reports considered necessary by STUK shall be submitted to STUK for approval. STUK issues a statement about the Operating Licence application only after having approved essential parts of each of these documents by a separate decision. Documents to be submitted to STUK for review are among others Final Safety Analysis Report (FSAR), Safety Classification document, Operational Limits and Conditions, quality manual for operations, PRA, summary programme for in-service inspections, Report on arrangement of the necessary safeguards to prevent the proliferation of nuclear weapons, administrative rules for operations, physical protection and emergency response arrangements, Environmental radiation monitoring programme.

Licensee is not allowed to load the nuclear fuel in to the reactor before the Operating License is granted. The Operating License is for fixed period. Typically Operating Licenses have been recently granted for 20 years. The first periods may be shorter. Anyhow, Periodic Safety Reviews are conducted typically every ten years.

4. Summary

Finland has a three step licensing process. All licenses in Finland are granted by the Government. Ministry of the Employment and the Economy performs all preparations for Government's decisions, asks statements from different stakeholders and arranges possibilities for public participation. STUK's role is to evaluate safety and give statements on safety in each licensing step.

The goal of the Decision in Principle is to decide whether using nuclear power is for the overall good of Finnish society. Government's decision has to be ratified by the Parliament to ensure political acceptance and commitment. The focus on the following steps of licensing is on the safety of the plant, starting from the safe design and construction in the Construction License phase and on the safe operations in the Operating license phase.

Preparation of the Regulatory Infrastructure for the New Nuclear Build

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Leopold Vrankar, Slovenian Nuclear Safety Administration, Slovenia

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Abstract

Slovenia is seriously considering building a new nuclear power plant. The Slovenian Nuclear Safety Administration (SNSA) is very much aware of the complexity of such a project as well as of the fact that at the moment the SNSA does not have sufficient resources for licensing and overseeing the design, construction and operation of the possible new plant. Likewise, the question arises whether technical support organizations which support SNSA in supervising the existing Krško nuclear power plant have sufficient capacity.

Therefore SNSA established a special project team with the task to prepare the Administration for the possible start of the new nuclear build. In the beginning of 2009, the project team prepared the analysis of licensing process, which is basically an overview of spatial planning, construction and nuclear safety regulation processes. The purpose of the review of the whole process, from spatial planning to the issuance of the operating license, was to identify phases which will require most effort. The next step was to set the strategy for the review process as well as to analyze and establish the basis for resource demands needed for SNSA's and other stakeholders involvements and decision making in the process. This will enable SNSA to establish a qualified and effective infrastructure for a possible new nuclear build.

1. Introduction

The construction of a second unit of Krško Nuclear Power Plant (NPP) is one of the national development projects for the period 2010 – 2023. It is a demanding project from the technical as well as the organizational point of view and the role of the SNSA will be decisive in granting licenses for the construction and operation of the new nuclear facility. For this reason the SNSA launched the project "Preparation of the SNSA on Construction of the second reactor of the Krško NPP" in 2007 [1] to raise the level of Administrations' preparedness regarding the technical and organizational manner in case of a new unit construction.

The initial step of the SNSA' preparation on the new build was to analyze the Slovenian legislation which sets the path for the siting, licensing and construction of a new NPP. On one side, the analysis provides a kind of a road map which would allow better resource allocation and thus greater effectiveness of the SNSA. On the other side it also helps to coordinate the efforts of all stakeholders involved and thus enhances the efficiency of the licensing process.

Secondly, based on the legislation analysis, the appropriate strategy for performing each SNSA's task of the licensing process was set. This included a detailed review of each task, proposing several possible strategies for its implementation and finally deciding on the final strategy for each task. The decision on a final strategy was made by the SNSA senior management and most experienced staff.

The third step is the analysis of human resources that SNSA will need for each task of the licensing process. The purpose was to assess the number of people the SNSA would require to carry out the licensing process of the new NPP and also their needed knowledge, skills and competences. Based on the analysis results, the SNSA will be able to prepare and implement recruitment plan as well as individual training programs for the SNSA staff.

The fourth step of the infrastructure preparation comprehends the review of needed technical support organizations (TSO), suitable communication and preparation for the cooperation as well as proper inspection audit of the TSO's competences.

The above listed steps, as shown in Figure 1, shall assure that the SNSA will be well prepared for successful and efficient execution of all tasks regarding the siting, licensing and supervising the construction of the new NPP.

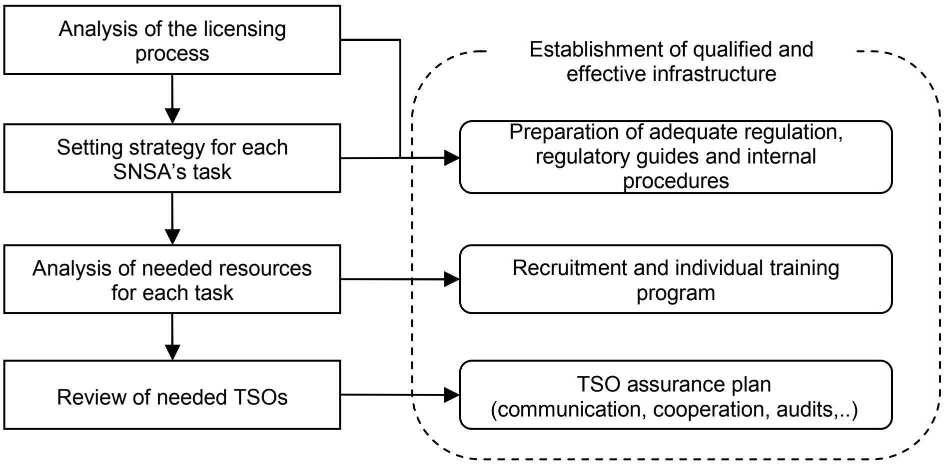

Figure 1: The process of establishing a qualified and effective infrastructure at the SNSA

2. The analysis of the licensing process for the new NPP and the role of the SNSA

The discussed analysis [2] was based on the detailed review of Slovenian legislation that would be used in the licensing process for a new NPP in Slovenia. The legislation includes laws, regulations and other legal documents from areas such as nuclear and radiation safety, spatial planning, environmental protection, construction, energy, public financing and other.

The whole process was partitioned into 10 separate phases, each one dealing with a specific period in the process and/or certain issue. Each phase was analyzed with the objective to determine the inputs and outputs of a phase, connections between different phases, the main players, cooperating parties and interactions between them, legal basis, time schedules and deadlines, needed documents and especially the role of the SNSA. The latter is presented in Table 1.

Table 1: The roles of the SNSA in the new NPP licensing process

PHASE OF THE PROCESS	SNSA's ROLE
National Strategic Spatial Plan	Gives remarks and suggestions on the NSSS draft
Licence to perform energy activity and energy permit	
National Spatial Plan (NSP)	
NSP draft	Prepares the »Guide on the scope and content of the Special Safety Analysis« and issues guidance for NSP
Supplemented NSP draft	Gives opinion on acceptability of Environmental Report and Special Safety Analysis, prepares positions on public and municipality remarks
NSP proposal	Gives opinion on acceptability of plan's impacts on the environment and issues design conditions
Environmental impact assessment (EIA) and environmental protection consent	Gives opinion on the plan's acceptability prepares data that must be included in EIA
Consent for construction	Issues the consent for construction and sets the conditions for trial operation
Review and approval of Safety Analysis Report (SAR) and other documentation	Reviews and approves the SAR, Decommissioning programme, Proposal of scope and duration of preoperational monitoring
Construction license	
Construction	Supervises the construction
Opinion of European Commission (EC)	Prepares and sends data necessary for issuing EC opinion
Trial operation	Issues the Consent for start of trial operation and supervises the trial operation
Review and approval of SAR and other documentation	Reviews and approves the SAR and the Trial operation programme
License for the use of the facility	
Operating license	Issues the operating license
Review and approval of SAR and other documentation	Reviews and approves the SAR and the Report on trial operation

The SNSA has already started preparing the "Guide on the scope and content of the Special Safety Analysis", which is a basic document for the siting process. Because of the tight administrative time limits, the SNSA also plans to prepare several other necessary documents in advance and thus enhance its efficiency.

Roles that will require most of the SNSA's effort and resources are reviews and approvals of documentation related to the plant's design (safety analysis report) in the phases of consent for construction, commissioning and operating license. Supervisions of construction and trial operation, which SNSA performs together with other competent inspection bodies, are also expected to be very effort and resource demanding phases. For proper engagement in these phases, the SNSA will have to recruit and qualify new personnel, assure that adequate TSOs are involved in the process and prepare new internal procedures and regulatory guides for investor and/or vendor regarding fulfillment of nuclear legislation requirements.

3. The SNSA's strategies for the licensing of the new NPP

It is of most importance that the appropriate strategy for realization of each licensing process task is set, since the SNSA is relatively small regulatory body with limited capabilities. Nuclear law requires the use of expert opinions (provided by TSOs) in the regulatory licensing process, meaning that for each application the licensee must provide an independent expert opinion regarding the justification of a licensed subject. There is also a possibility of adaptation of the expert opinion instrument, e.g. the expert opinion can be ordered solely by the licensee or in cooperation with the regulatory body. A regulatory body can also order an independent expert opinion on its own. Furthermore, there are also tasks in the licensing process for the new NPP which the SNSA can and must perform on its own, i.e. for those tasks expert opinion is not required.

In the frame of the prescribed time and extent of needed effort, the project group for the new NPP prepared the proposition of strategies which incorporated several strategy proposals for each task of the licensing process. The main differences among strategies were in the use of the TSO, e.g. whether the expert opinion is required or not and how it is ordered, e.g. the expert opinion is ordered solely by the investor or together with the SNSA, in which case the SNSA has major role on the scope and contents of the expert opinion and also on the choice of the TSO, etc.

The prepared set was overviewed by the SNSA senior management and final strategies were determined. They were used in the next step of the preparation process, the analysis of the required human resources, and will also be used for the preparation of adequate regulatory guides and internal procedures.

4. The analysis of needed human resources

The purpose of this analysis was to assess the resources needed for successful and timely realization of each task based on the chosen strategy. Furthermore, the aim was to evaluate the needed knowledge, skills and competences of each individual SNSA staff member involved in the new NPP licensing process.

The SNSA commenced this analysis with a detailed qualitative and quantitative review of expected contents of each task. Based on that, basic competences and approximate number of people, needed for each task were set. For less demanding tasks, such as issuing a guidance in the siting process, issuing "Guide on the scope and content of the Special Safety Analysis", giving various opinions in the process and other, the assessments were made based on the experience with similar projects. Greater challenge presents assessing the resources needed for major tasks like reviewing and approving the project documentation, e.g. safety analysis report in the phases of consent for construction license, trial operation and operating license, simply because since the existence of the SNSA, the Administration has not handled a project on such a large scale. Also due to the SNSA's atypical licensing process (expert opinion instrument), it is not simple to use the experience of foreign nuclear regulatory bodies.

The preliminary results show that tasks of initial phases, those regarding the siting, can be efficiently and timely accomplished by the existent SNSA staff. The first major and at the same time most extensive task will be the review of the safety analysis report (SAR) in the phase of issuing the consent for construction, in which the SNSA shall review and approve the SAR, design for construction license and other documentation. For the purpose of assessing the resources needed for topics related to the plant design, the required man hours and the number of resources for each specific technical field was determined with regard to the SNSA regulatory approach.

Considering the present coverage of these technical fields, the results show that for the consent for construction phase the SNSA would require additional 10 to 17 new employees in different technical fields (for comparison: for the time being nuclear and radiation safety divisions employ 23 people all together).

Workload distribution on different technical fields is shown in Figure 2.

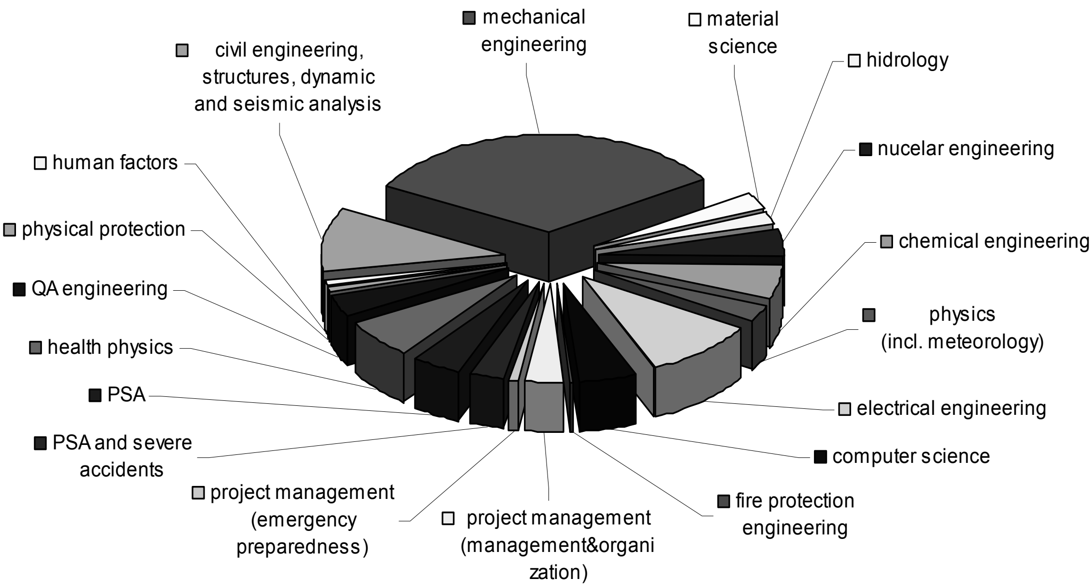

Figure 2: Distribution of workload for SAR review and approval on different technical fields

At the moment this analysis is still in preparation. These numbers are unofficial and need additional review and argumentation before the final number of new employees can be proposed. The analysis will also be supplemented with a detailed description of knowledge, skills and competences needed which will be followed by the preparation and implementation of recruitment plan and individual training programmes.

5. Conclusion

The SNSA has seriously started preparing for the possible new build. It sees the key to the successful, efficient and on-time realization of tasks regarding the licensing process for the new NPP in sufficient number of well trained personnel, supported by well trained and equipped TSOs.

The analysis of the licensing process was the first step by which the SNSA got acquainted with the licensing process for the new NPP and set the base for further analysis and preparation. With the completion of the human resources analysis the SNSA will be prepared to start executing its recruitment plan and detailed individual training programmes.

At the same time, the SNSA will start the review of needed TSOs and continue with implementing appropriate measures to assure the availability of competent and well equipped TSOs.
Besides the above mentioned, the SNSA has started developing necessary documents and guides in advance and will continue with the development of required regulatory guides and internal procedures.

All together shall enable SNSA to await the new NPP licensing process as a prepared, well trained, equipped and competent organization.

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Licensing of New Nuclear Power Plants in Canada

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Executive summary

The regulatory process for new power plant licensing in Canada, from receipt of the initial application to commercial operation, can be divided into three phases:

- Environmental Assessment (EA) and *Licence to Prepare Site*
- Licence to Construct; and
- Licence to Operate

The *Nuclear Safety and Control Act* (NSCA) does not have provisions for combined licences for site preparation, construction, or operation. Separate licences must, therefore, be granted for each phase, and would be issued in sequence. However, applications to prepare a site, to construct and to operate a new nuclear power plant could be assessed in parallel.

The total duration from the application for the Licence to Prepare Site to the issuance of the Licence to Operate (which is a prerequisite for first fuel load) has been established as 9 years subject to certain factors. To help facilitate this timeline, the CNSC has undertaken an aggressive program of documenting regulatory practices, requirements and guidance to assist applicants in submitting complete applications. Working level procedures to assist CNSC staff in their review of submissions are also under development. Extensive program and project management has been introduced to ensure that timelines will be achieved.

In parallel with the above activities, regulatory oversight measures to be employed during site preparation activities and plant construction and commissioning are also being developed.

On the international front, the CNSC is participating in the MDEP program to leverage the resources and knowledge of other national regulatory authorities in reviews the CNSC is undertaking. The CNSC also participates in IAEA and other international activities to utilize/adapt international practices as appropriate in Canada.

1. Background

The development, production and use of nuclear energy and the production, possession of use of nuclear substances is regulated solely at the federal level by the Canadian Nuclear Safety Commission (CNSC) under the mandate of the *Nuclear Safety and Control Act* (NSCA) and supporting regulations under the Act. The NSCA gives the Commission a broad mandate to regulate all activities related to the use of nuclear energy and materials and provides for *the limitation, to a reasonable level and in a manner consistent with Canada's international obligations, of the risks to national security, the health and safety of persons and the environment*. This includes consideration of not only radiological hazards and effects but conventional hazards and effects as well.

The Licensee is the cornerstone of safety and is held accountable by their licence. Under the NSCA, *no licence may be issued, renewed, amended or replaced unless, in the opinion of the Commission, the applicant:*

(a) is qualified to carry on the activity that the licence will authorize the licensee to carry on; and

(b) will, in carrying on that activity, make adequate provision for the protection of the environment, the health and safety of persons and the maintenance of national security and measures required to implement international obligations to which Canada has agreed.

The CNSC is a "balanced" regulator which means that it is balanced between purely prescriptive (rule based) regulation and purely non-prescriptive regulation (concept-based using expert opinions). Under this regime, the applicant is expected to propose, based on considerations contained in Regulatory Documents and applicable Canadian Codes and Standards, how they will meet the requirements of the Regulations under the *Nuclear Safety and Control Act*. This is intended to allow applicants a measure of flexibility in the methods they will use to support their unique licensing case. The applicant's proposal is then reviewed by CNSC Staff against modern industry practices and documents under the CNSC regulatory framework. However, it must be recognized that additional review effort may be needed by CNSC Staff when alternative approaches to meet regulatory requirements are proposed.

2. Overview of Licences

In the CNSC's regulatory regime, nuclear power plants are defined as Class I nuclear facilities. The *Class I Nuclear Facilities Regulations* require separate licenses for each of the five phases in the lifecycle of a nuclear power plant:

Licence to Prepare Site (which is accompanied by an Environmental Assessment which is required by a separate piece of federal legislation called the *Canadian Environmental Assessment Act*.);

Licence to Construct;

Licence to Operate;

Licence to Decommission; and

Licence to Abandon.

The regulatory requirements that pertain to nuclear power plants are found in:

The *General Nuclear Safety and Control Regulations;*

The *Radiation Protection Regulations;*

The *Class I Nuclear Facilities Regulations;*

The *Nuclear Substances and Radiation Devices Regulations;*

The *Packaging and Transport of Nuclear Substances Regulations;*

The *Nuclear Non-Proliferation Import and Export Control Regulations;* and

The *Nuclear Security Regulations.*

These regulations provide licence applicants with general performance criteria, and list the information which all applicants must submit to the CNSC as part of the licence application. Applications for licences must be accompanied by licence fees, as set out in the *CNSC Cost Recovery Fees Regulations,* which is available at **www.nuclearsafety.gc.ca**. Other legislation, enacted by Parliament, with which the applicants are required to comply includes, but is not limited to:

The *Nuclear Liability Act;*

The *Nuclear Fuel Waste Management Act;*

The *Canadian Environmental Assessment Act;*

The *Canadian Environmental Protection Act 1999;*

The *Fisheries Act;*

The *Species at Risk Act*;

The *Migratory Bird Convention Act*; and

The *Canada Water Act*.

The *Nuclear Safety and Control* Act (NSCA) does not have provisions for combined licences for site preparation, construction, or operation. Separate licences must, therefore, be granted for each phase, and would be issued in sequence. However, applications to prepare a site, to construct and to operate a new nuclear power plant could be assessed in parallel. Only the *Licence to Prepare Site*, the *Licence to Construct* and the *Licence to Operate* are discussed in this paper.

At a high level:

- The *Licence to Prepare Site* permits the licensee to clear land, build site service infrastructure, and re-contour land. The licensee may also be permitted to excavate the plant footprint if the specific plant design to be built has been identified and appropriate high level design information about the facility has been reviewed and accepted by the CNSC. Construction of any plant Structures, Systems or Components, such as the plant footings and base mat, will not be permitted under a *Licence to Prepare Site*.

- The *Licence to Construct* permits the licensee to perform all plant construction and commissioning activities up to but not including first fuel load into the core.

- The *Licence to Operate* permits the licensee to load fuel, complete remaining construction and commissioning with fuel in the core, and proceed to long-term commercial operation.

The total duration from the application for the *Licence to Prepare Site* to the issuance of the *Licence to Operate* has been established as 9 years, subject to certain factors.

3. Parallel *License to Prepare Site* and Environmental Assessment (EA) Processes[1]

3.1. Environmental Assessment

Before a *Licence to Prepare Site* can be issued by the Commission of the CNSC, the *Canadian Environmental Assessment Act* requires an EA to be completed to identify whether the project in question is likely to cause significant environmental effects, and to determine whether those effects can be adequately mitigated. EAs examine the five phases in the lifecycle of a nuclear power plant: siting, construction, operation, decommissioning and abandonment. For new nuclear power plants, the CNSC initiates an EA when a proponent applies to the CNSC for a *Licence to Prepare Site* and submits a complete *Project Description* [1]. A Project Description is a document, prepared by the proponent of a proposed major resource project that provides details about the key components and activities associated with the project, as well as the anticipated interaction between the project and its environment.

The Environmental Assessment (under the *Canadian Environmental Assessment Act*) and the *Licence to Prepare Site* (under the *Nuclear Safety and Control Act*) have overlapping but distinct information requirements. For instance, an EA requires more information about potential accidents and incidents than the CNSC regulations require in a *Licence to Prepare Site*. Conversely, the CNSC regulations require information that is not typically included in an EA, such as a requirement to submit the proposed quality assurance program for the design of the facility. Both environmental assessment and *Licence to Prepare Site* processes occur concurrently. This ensures that the information submitted by the proponent can be considered by public and government agencies through a single process, and any appropriate decisions under EA and licensing can be made by a single body.

Currently, the reviews of new reactor projects of this type are conducted in the public forum by a Joint Review Panel of typically three to four persons appointed for their expertise by the Federal Minister of the Environment in consultation with the President of the CNSC. The Joint Review Panel members are also appointed as temporary members of the CNSC Commission with full powers under the *Nuclear Safety and Control Act* to issue the *Licence to Prepare Site* at the end of the process. The review panel terms of agreement are contained in a *Joint Panel Agreement* written specifically for each project and co-approved by the Federal Minister of the Environment and the President of the CNSC.

3.2. Licence to Prepare Site

Prior to issuing a *Licence to Prepare Site*, the Commission must be satisfied that the site is suitable for the full lifecycle of the project and that it is feasible to perform the site preparation activities in a manner that will satisfy all health, safety, security and environmental protection requirements. It is the responsibility of the applicant to demonstrate that it will make adequate provisions, when applying for a licence.

The following aspects are considered in the evaluation of the suitability of a site over the life of a nuclear power plant:

- The potential effects of external events (such as seismic events, tornadoes and flood) and human activity on the site;

[1] These processes are discussed in more detail in the CNRA International Workshop white paper entitled *Siting Practices and Site Licensing Process for New Reactors in Canada*, Mr. M. deVos, CNSC, Canada, April 2010.

- The characteristics of the site and its environment, which could influence the transfer (to persons and the environment) of radioactive and hazardous material that may be released; and,
- The population density, population distribution and other characteristics of the region, insofar as they may affect the implementation of emergency measures and the evaluation of the risks to individuals, the surrounding population and the environment.

Under the regulations, an applicant must submit, for any licence, a Project Description of the facility and plans showing the location, perimeter, areas, structures and systems of the facility. An application for a *Licence to Prepare Site* does not require detailed information or determination of a reactor design; however, high level design information is required for the EA that precedes the licensing decision for a *Licence to Prepare Site*. An application for a *Licence to Construct* must contain more detailed information about the reactor design and a supporting safety case.

The technical information arising from the consideration of external events, site specific characteristics and supporting assessments, is used as input into the design of the nuclear power plant, and must be included in the application. The CNSC Staff's conclusions and recommendations from technical reviews are documented in reports submitted to the Joint Review Panel.

The CNSC is developing GD-368, "Licence to Prepare Site Application for Class 1A Reactors with Thermal Output Greater than 5 MW – Guidelines" [2] to provide guidance regarding applications for a *Licence to Prepare Site*. Please refer to the accompanying CNSC paper "Siting Practices and Site Licensing Process for New Reactors in Canada" for further information.

4. Long-lead Items

It is recognized that the procurement of major components may need to begin well before a construction licence is granted. These are referred to as long-lead items.

Prior to the start of manufacture of any long-lead items, the applicant must first obtain code classification approval from the CNSC for the items under consideration. The application for code classification approval should include the procurement schedule, and must demonstrate that the items will meet all regulatory requirements. Subsequently, the applicant must disposition any changes between the version of the code used to procure the long-lead items, and the version of the code established for the construction of the plant via the construction licence (code effective date). A disposition report must be submitted to the CNSC in support of final code classification approval and prior to installation of the item.

In addition, the CNSC verifies that the applicant is performing adequate oversight of the design, procurement and manufacture of long-lead items. This may include inspection of the applicant's surveillance processes and procedures, and verification of the acceptability of any independent third party inspection services utilized by the applicant. CNSC staff may also conduct inspections at manufacturer facilities.

5. Licence to Construct

When applying for a *Licence to Construct* a nuclear power plant, it is the responsibility of the applicant to demonstrate to the CNSC that the proposed design of the nuclear power plant conforms to regulatory requirements, and will provide for the safe operation on the designated site over the proposed life of the facility. The information required in support of the application to construct a nuclear power plant is referred to as the "safety case" and includes, for example:

- A description of the proposed design for the nuclear power plant, taking into consideration the physical and environmental characteristics of the site;
- Environmental baseline data, on the site and surrounding area;
- A Preliminary Safety Analysis Report, showing the adequacy of the design;
- Measures to mitigate the effects on the environment and health and safety of persons that may arise from the construction, operation or decommissioning of the facility;
- Information on the potential releases of nuclear substances and hazardous materials, and proposed measures to control them; and,
- Programs and schedules for recruiting and training staff for the construction, commissioning and operation phases of the project.
- Programs and activities that will be undertaken by the applicant to perform oversight of design, procurement, construction, commissioning and operation activities to provide assurance that the plant will conform to regulatory requirements and the design and safety analysis as presented in the application.

A more complete listing of the information to be submitted may be found in Section 5 of the Class I Nuclear Facilities Regulations (available at http://www.nuclearsafety.gc.ca). CNSC Regulatory Guidance document GD-369 "Construction Licence Applications for Nuclear Power Plants: Guidelines" [3] provides additional guidance regarding construction licence applications. The development of this document has utilized IAEA Safety Standards Series GS-G-4.1 "Format & Content of Safety Analysis Reports" [4]. Other key CNSC documents include:

RD-337 "Design of New Nuclear Power Plants" [5] (based on IAEA NS-R-1 "Safety of Nuclear Plants: Design" [6]);

RD-310 "Safety Analysis for Nuclear Power Plants" [7]; and

RD-346 "Site Evaluation for New Nuclear Power Plants" [8] (based on IAEA NS-R-3 "Site Evaluation for Nuclear Installations" [9]).

Overall, the CNSC's approach for the regulatory review of the application is designed to:

- Obtain reasonable assurance that the applicant is qualified to carry on the activities that the licence will authorize, and that the applicant will, in carrying out those activities, make adequate provision for the protection of the environment, and the health and safety of persons,
- Provide transparency and clarity, to maximize efficiency and effectiveness; and
- Obtain reasonable assurance that the facility design meets all regulatory requirements and can be constructed, commissioned and operated safely as designed, and that no new safety issues will be identified prior to reactor operation.

After the construction licence application has been received, the CNSC performs a comprehensive assessment of the design documentation, the Preliminary Safety Analysis Report, the construction program, and all other information required by the regulations. The assessment focuses on determining whether the proposed safety case meets regulatory requirements. Specifically, the evaluation involves rigorous engineering, scientific analysis and engineering judgment, taking into consideration the CNSC's experience and knowledge of the best practices in nuclear plant design and operation, as gained from existing power plants in Canada and around the world.

In addition to reviewing the information included in the application, the CNSC also verifies that any outstanding issues from the site preparation stage have been resolved. The CNSC staff's conclusions and recommendations from these reviews are documented in reports submitted to the Commission; the Commission then makes the final decision on the issuance of the construction licence.

Additional areas of focus for the regulatory review include:

- The readiness of the applicant to provide adequate management oversight of the project, in particular with regard to manufacturing and construction activities, with a schedule as to how the organisation will develop that oversight as the project develops.

- Design and safety analysis, assessing whether the proposed design and safety analysis, along with other required information, meet regulatory requirements. Design and safety analysis is to be accompanied by supporting experimental results, tests and analysis. This is particularly important for novel design features and where the applicant has proposed alternative approaches.

- The independent peer review of the safety assessment conducted by individuals or groups separate from those carrying out the design, including a clause-by-clause assessment against RD-337, "Design of New Nuclear Power Plants".

- The commissioning program.

- General plans, including schedules, for the development of the operating organisation, training, staff certifications and operational procedures. The applicant is expected to demonstrate that due consideration has been given to the preparation of an operating organisation that is ready to commission and operate the facility; and

- Policies, strategies, and provisions employed for radiation protection, emergency preparedness, environmental protection, management of radioactive and hazardous waste, decommissioning and safeguards. Detailed information is not needed at this stage; however sufficient information must be provided to show that adequate provisions have been made in the design.

As noted earlier, the Commission will not issue a licence unless it is satisfied that the applicant will make adequate provisions to protect health, safety, security and the environment, and to respect the international obligations to which Canada has agreed. As such, it is the responsibility of the applicant to show that there are no major safety issues outstanding at the time the Commission considers the application for a construction licence. In order for the applicant to be able to demonstrate this with confidence, it is necessary for the design of the facility and the safety analysis to be well advanced and supported by appropriate and adequate research including experimental tests and analysis.

During the construction phase, the CNSC carries out compliance activities to verify that the licensee is complying with the NSCA, associated regulations and its licence. Such compliance activities focus on confirming that plant construction is consistent with the design, and that the licensee is demonstrating adequate project oversight and is confirming that quality assurance requirements are being met. Regulatory oversight activities include, but are not limited to:

- Inspections, surveillance, reviews, witnessing of commissioning tests, evaluations of commissioning test results;
- Inspections at manufacturer facilities;
- Assessing the effectiveness of applicant's oversight of construction and commissioning activities; and
- Granting approvals pertaining to commissioning hold points.

Towards the latter part of construction, regulatory attention turns towards the inactive commissioning program (without fuel loaded) and associated activities. The purpose is to verify, to the extent practicable, that all the systems, structures and components have been installed correctly and are performing per the design intent including their response to abnormal plant conditions (as credited in the safety analysis).

6. Facility Commissioning

Objectives of regulatory oversight of the facility commissioning program are to determine:

- That the commissioning program is comprehensively defined and implemented to confirm that the structures, systems and components (SSCs) important to safety and the integrated plant will perform in accordance with the design intent, safety analysis and applicable licensing requirements;
- That the operating procedures covering all operating and abnormal states have been validated to the maximum extent practicable;
- That the commissioning and operating staff have been trained and qualified to commission the plant, and operate it safely in accordance with the approved procedures; and
- That the management system has been adequately defined, implemented and assessed to provide a safe, effective and quality working environment to perform and support the conduct of the commissioning program.

For each phase of commissioning, plant management is expected to establish a set of commissioning control points (CCPs) to achieve a transparent, accountable and effective process to ensure that the defined pre-requisites for the release of each CCP have been formally demonstrated.

Licensing CCPs are regulatory hold points requiring prior CNSC authorization to proceed further. Non-licensing CCPs are usually treated as CNSC witness points. All applicable non-licensing CCPs must be satisfactorily completed as part of obtaining the release from licensing CCPs.

7. Licence to Operate

When applying for a *Licence to Operate* a nuclear power plant, it is the responsibility of the applicant to demonstrate to the CNSC that it has established the management system, plans and programs that are appropriate to ensure safe and secure operation. Information required in support of the application for a licence to operate includes, for example:

- The Final Safety Analysis Report; and
- Finalized policies, programs and supporting procedures to ensure safe operation of the facility covering areas such as:
  - Operation and maintenance of the nuclear facility;
  - Handling of nuclear substances and hazardous materials;
  - Control of the release of nuclear substances and hazardous materials into the environment;
  - Preventing and mitigating the effects on the environment and health and safety, resulting from the operation and subsequent decommissioning of the plant;
  - Readiness of emergency preparedness measures, including assistance to deal with an abnormal off-site release; and
  - Facility security.

A more complete listing of the specific information required to obtain a licence to operate a nuclear power plant is found in Section 6 of the *Class I Nuclear Facilities Regulations* (available at **http://www.nuclearsafety.gc.ca).**

In addition to assessing the information included in the application to operate the nuclear power plant, the CNSC also verifies that any outstanding issues from the construction licensing stage have been resolved. The CNSC staff's conclusions and recommendations from these reviews are documented in reports submitted to the Commission, which then makes the final decision on the issuance of the operating licence.

The *Licence to Operate* will enable the operator to load fuel and begin active commissioning. The initial operating licence is typically issued with conditions (hold points) pertaining to fuel load, approach to first criticality, low power tests, and tests during ascension to full power. All relevant commissioning tests must be satisfactorily completed before a hold point can be released.

During the subsequent long-term operation of the plant, the CNSC carries out compliance activities in order to verify that the licensee is complying with the NSCA, associated regulations and its licence terms. If the compliance activities identify any non-compliance or adverse trend, there is a range of corrective measures that the CNSC can take, ranging from a request for licensee action to prosecutions.

8. Timelines for Licenses

As discussed above, the regulatory process for new power plant licensing, from receipt of the initial application to commercial operation, can be divided into three phases:

- EA and Licence to Prepare Site;
- Licence to Construct; and

- Licence to Operate.

As a regulatory agency, the CNSC must satisfy itself that the Crown's duty to consult and, if appropriate, accommodate, has been met towards Aboriginal communities whose rights may be impacted. Activities related to Aboriginal consultation take place throughout the life-cycle of the project, including pre-submission licensing phase, the Environmental Assessment phase, and all licensing phases including the period of plant operation.

As shown in Table 1, the CNSC estimates that the total time from the receipt of application to the issuance of a licence to operate is approximately nine years — taking into consideration that a number of activities may proceed in parallel.

Table 1 outlines the estimated durations of the EA and Licensing phases, and includes approximate time for applicant's activities, such as preparation of the site and construction of the facility. The applicant's activities occur in parallel with regulatory activities. These time estimates are based on the following:

- The CNSC will receive complete and comprehensive applications. This means that the design of the facility and the safety analysis are well advanced early on in the project (Construction Licence stage or earlier) and supported by appropriate and adequate research including experimental tests and analysis.

- The time needed for resolution of comments on submissions, or any safety issues identified, is minimal; and

- The time required for the 2-day licence hearing process is included.

Table 1: Approximate Duration of the Environmental Assessment and Licensing Steps

Activity	**Duration (nominal)**
Aboriginal Consultation	ongoing
Environmental Assessment and Licence to Prepare Site *(includes development of Joint Review Panel Agreement and EIS Guidelines)*	24 months
Applicant prepares site	18 months
Licence to Construct – at least 6 months overlap with the previous activities	30 months
Licence to Operate	24 months
Applicant's activities, e.g. plant construction	48-54 months
Total duration from the application for the Licence to Prepare Site to Licence to Operate, taking into account overlapping	

environmental assessment/licensing and applicant's activities,
which may run in parallel **9 years**

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1. *Guide to Preparing a Project Description for a Major Resource Project*, Government of Canada Major Projects Management Office December 2008 (http://www.mpmo-bggp.gc.ca/desc/pdf/guide-eng.pdf).
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Licensing Experience of New Reactor (APR1400) in Korea

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Abstract

This paper describes an overview of nuclear regulatory framework and Licensing Experience of New Reactor (APR1400: Advanced Power Reactor 1400) in Korea such as licensing process, pre-operational inspection and safety review results for APR1400, etc. In addition, this paper has been prepared to share the information on the regulatory issues in safety review results of the APR1400.

1. Introduction

The Korean regulator currently regulates 28 NPPs in total, of which 20 NPPs are in operation and the other 8 NPPs are under construction. The Korean Government and nuclear industry launched a 10-year National Project in 1992 to develop technologies involved in the design of an advanced power reactor. The utility applied the Pre-application Safety Review (PSR) for the APR1400 in January 2000, and the Standard Design Approval (SDA) for the APR1400 in July 2001. The Korean regulatory system adopted the two-step licensing approach, namely Construction Permit (CP) and Operating License (OL), but added PSR and SDA developing the APR1400 in 2001. The APR1400 was designed based upon the 1000 MWe Korean Standard Nuclear Power Plant (KSNP) design. The safety review focused on the design differences between KSNP and APR1400, especially on the up-sizing effects from 1000 MWe to 1400 MWe. As a result of the safety review of the APR1400 safety aspects, it can be concluded that the APR1400 is designed to meet all the safety requirements and can be constructed safely. The utility applied CP for Shin-Kori unit 3&4 in October 2003, and the CP was granted in April 2008 after a review of about 4.5 years. Safety review of Shin-Uljin unit 1&2 are undergoing.

2. Licensing Process

Pursuant to the Atomic Energy Act of Korea, the licensing procedure for nuclear installations basically consists of two stages; the CP and the OL. Overall licensing process in Korea is shown in Figure 1. The Ministry of Education and Science Technology (MEST), and the Korea Institute of Nuclear Safety (KINS) developed the PSR and SDA System for the safety review of APR1400. The safety review of APR1400 applied the developed system and existent system.

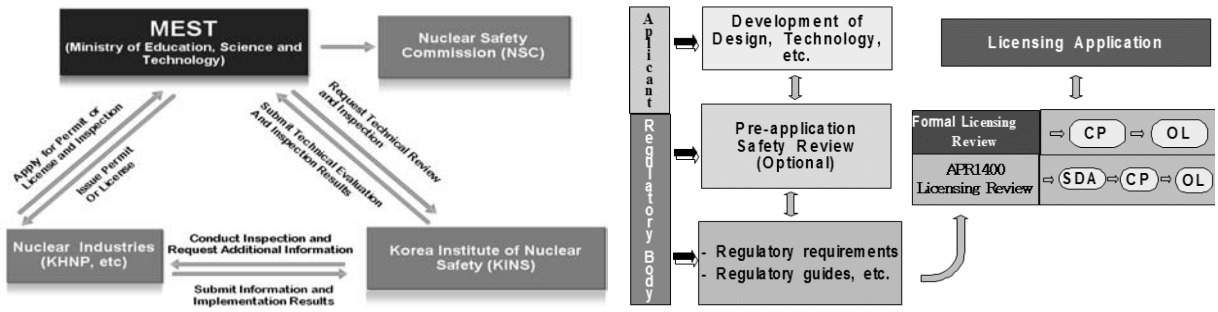

Fig.1. Overall licensing process of nuclear in Korea

2.1 Pre-application Safety Review (PSR)

As early as possible in the development stage of the APR1400, the utility wishes to secure the licensibility of proposed designs with advanced features such as passive safety systems, digitalized instrumentation and control systems, etc., so development can proceed in a stable environment. In this regard, the Government, MEST and KINS introduced the PSR to nuclear industries, to encourage advanced interaction of applicants with the regulatory body, not only for early identification of regulatory requirements but to provide more timely and effective regulation. In addition, the PSR system enables the reflected comments of related parties, including the public, from the view point of the regulatory body, concerning the desired characteristics of the APR1400. Such interaction and guidance early in the design stage contributes toward minimizing complexity and enhancing stability in the licensing.

2.2 Standard Design Approval (SDA)

The standardization of nuclear power plant designs is important for the utility because it significantly enhances safety, reliability, availability, and economy. Approved standard designs benefit public health and safety by concentrating resources on specific design approaches, by stimulating standardized construction practices and quality assurance, and by fostering more effective maintenance and operation. Accordingly, it is expected that standardization of nuclear plants could further improve safety in future plants and promote more efficient reviews. A standard design approval review was performed prior to a construction of a nuclear power plant in accordance with the Atomic Energy Act of Korea and applied first to the licensing of APR1400. A legislation of the SDA was completed in January 2001, and was applied to the licensing of the APR1400 standard design nuclear power plant for the first time according to the application from the utility, Korea Hydro and Nuclear Power (KHNP), in July 2001.

2.3 Construction Permit (CP)

In order to obtain a CP for nuclear installation, the applicant shall submit to the MEST an application for the CP accompanied by the preliminary Safety Analysis Report (SAR), the quality assurance program (QAP) for design and construction, and the radiological environmental report. Based on the results of the safety review by KINS of the application for the CP, the Minister will issue the CP after deliberation by the Nuclear Safety Commission (NSC).

The safety review of the application for the CP is conducted to confirm that the site and the preliminary design of the nuclear installation are in conformity with the relevant regulatory requirements and technical guidelines. It includes safety reviews of the principle and concept of reactor facility design, the implementation of the regulatory criteria, the evaluation of the environmental effects resulting from the construction, and a proposal for minimizing those effects. Also, the radiological environmental report to be filed together with the application for the CP as well as for early site approval should contain the public's opinion from the area surrounding the nuclear installation through a public hearing, as necessary.

2.4 Operating License (OL)

To obtain an OL for a nuclear installation, the applicant shall submit to the MEST an application for the OL accompanied by the final SAR, the operational technical specifications, the QAP for operation, and the radiological emergency plan. Based on the results of the safety review by KINS of the application for the OL and the results of pre-operational inspections, the Minister will issue the OL after deliberation by the NSC. The safety review of the application for the OL is conducted to confirm that the final design of the nuclear installation is in conformity with the relevant regulatory requirements and technical guidelines and that the nuclear installation may continue to operate throughout its lifetime.

2.5 Pre-Operational Inspection (POI)

The purpose of pre-operational inspection is to confirm whether the structures, systems, components (SSCs) of plants are manufactured, installed, and tested in compliance with the SAR and QAP, and whether the performance of related facilities meet relevant technical requirements. The scope of pre-operational inspection covers not only the facilities of the safety related functions but also those important to safety. The pre-operational inspection is composed of 5 stages based on the field activities: structure inspection; installation inspection; cold functional test (CFT) inspection; cold hydrostatic test (CHT) and hot functional test (HFT) inspection; initial fuel loading and startup test inspection.

3. Safety Review

3.1 Standard Design Approval

The safety review focused on confirming the safety of the advanced design features, including severe accident measures. During the review period, there were five rounds of Request for Additional Information (RAI), and about 2,200 RAIs were raised. Licensing issues during the safety review for SDA were the following items:

- Thermal-hydraulic loads and pool temperature of In-containment Refueling Water Storage Tank

- Performance Evaluation of ECCS

- Consideration of Environmental Effect in Fatigue Evaluations of ASME Code Class 1 Components

- Soft Control Application for Digital I&C System

- Human Factors Engineering for the Advanced Control Room etc.

As a result of the safety review of the APR1400 safety aspects, it can be concluded that the APR1400 was designed to meet all the current safety requirements and could be constructed safely. Therefore, the SDA of APR1400 was issued in May 2002. At the issuance of the SDA, supplementary actions for major confirmatory items of the following were requested.

- Uncertainty analysis for the core cooling capability at late reflood phase

- Verification of design suitability for soft control and safety console

- Submission of design report concerning steam generator tube integrity

The APR1400 standard design served as a reference plant for Shin-Kori unit 3&4. The utility's reports on confirmatory items were submitted before the application of construction permit for Shin-Kori unit 3&4 in July 2003. The confirmatory items were resolved properly during the safety review of CP for Shin-Kori unit 3&4.

3.2 Shin-Kori unit 3&4

Design certification of APR1400 was issued in May 2002 after two years of safety review by KINS. The utility applied CP for Shin-Kori unit 3&4 in October 2003, and the CP was granted in April 2008 after a review of about 4.5 years.

The design safety of the preliminary design of the nuclear reactor and related facilities was checked, the adequacy of the site was assessed by reviewing the safety of the planned construction site, and the impact of the radioactivity that might be caused by the construction and operation of the nuclear reactor and related facilities on the surrounding environment was assessed by reviewing the preliminary safety analysis report (PSAR), the radiation environmental report, the construction quality assurance program, the explanatory statement on the use of nuclear reactor, and the explanatory statement on technical capabilities in respect to the design of the nuclear reactor. The documents were attached to the application for CP of Shin-Kori unit 3&4.

To check the overall safety of Shin-Kori unit 3&4, the implementation of the follow-up actions for the APR 1400's standard design approval, major design features of Shin-Kori unit 3&4, inclusion of domestic and overseas experience in nuclear power plant operation, and other matters were reviewed in depth. To check the safety for major design features of Shin-Kori unit 3&4, the reactor cavity flooding system (CFS), the safety class 1 system considering environmental impact and other matters were reviewed.

4. Conclusion

Pre-application safety review and standard design approval enhanced communication with stakeholders and regulatory stability for new reactor licensing. Standard design and safety approaches of APR1400 satisfied the current safety standard. Licensing issues identified in both pre-application review and SDA review were resolved during the safety review of construction permit for Shin-Kori unit 3&4. The detailed evaluation results of additional matters will be confirmed in the construction and detailed design phase.

As a result of the safety review of the application for construction permit of Shin-Kori unit 3&4, the location, structure and equipment of the nuclear reactor and related facilities were satisfied the current safety requirements, the public health and the environment from the impact of the radioactive materials generated from the construction of the facilities can be protected in service.

References

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Pre-Project Regulatory Reviews of ACR-1000™ AND EC6™

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1. Summary of ACR-1000 and EC6 Design Features

Atomic Energy of Canada Limited (AECL) has developed the Advanced CANDU Reactor™[2]-1000 (ACR-1000™) and the Enhanced CANDU 6 ™ (EC6™) reactors as evolutionary advancements of the CANDU 6™ reactor. There are 11 CANDU 6 units currently licensed and in operation since the early 1980s in Canada, Argentina, Korea, Romania, and China, with exceptional lifetime operating performance records (total of 500 CANDU™ at-power operating years).

The ACR-1000 is a two-unit nuclear power plant, each unit with a gross electrical output of 1165 MWe. The ACR-1000 design is largely based on the proven design concepts of reactor and process systems of current CANDU plants with a number of design innovations. The ACR-1000 is a heavy-water moderated and light-water cooled pressure tube reactor with 2.4% enriched fuel arranged in a 43-element CANDU fuel bundle, using proven on-power CANDU-specific refuelling technology.

The ACR-1000 reactor assembly consists of 520 channels in a reduced square lattice pitch, with larger-diameter calandria tubes (CTs) than current CANDU designs, contained within a CANDU-typical calandria vessel. The Reactor Coolant System (RCS) contains light-water coolant operating at higher temperatures and pressures than current CANDU designs for increased turbine cycle efficiency, and is arranged in a CANDU-typical two-loop figure-of-eight configuration with four steam generators, four RCS pumps, four reactor outlet headers, and four reactor inlet headers, with associated inlet and outlet feeders. The fuel handling system consists of two fuelling machines for on-line refuelling. The safety systems are designed for high reliability, redundancy, separation and defence-in-depth; they include the two shutdown systems, the emergency core cooling system, the emergency feedwater system, and the containment system. The ACR-1000 applies the four quadrant design approach for improved operating performance.

The EC6 is a two-unit nuclear power plant, each unit with a gross electrical output of 725 MWe. The EC6 has evolved from the CANDU 6 plants which have proven safety and performance records, and features a number of safety and operability enhancements. The EC6 is a heavy-water moderated and cooled pressure tube reactor with natural uranium fuel arranged in a 37-element CANDU fuel bundle, using proven on-power CANDU-specific refuelling technology.

The EC6 features a modular, horizontal fuel-channel core, with a separate low-temperature, low-pressure moderator providing an inherently passive heat sink for protection against severe accidents. The

[2] ACR-1000, EC6, and CANDU are registered trademarks of Atomic Energy of Canada Limited (AECL)

surrounding reactor vault is filled with light water, and also provides passive cooling in potential severe accident situations. As with all CANDUs, the EC6 has two fully independent and capable safety shutdown systems and a separate reactor control system, which can also shutdown the reactor in a number of anticipated operations occurrences. The major improvements incorporated in the EC6 design include: a more robust containment with passive features (e.g., thicker walls, steel liner; enhanced severe accident control using emergency heat removal systems); improved shutdown performance for increased large LOCA margins; upgraded fire protection systems; additional design features to improve environmental protection for workers and public; automated and unitized back-up standby power and water systems; improved reactor trip coverage; and a design life up to 60 years with a mid-life refurbishment of critical equipment such as fuel channels.

2. CNSC Pre-Project Regulatory Reviews

The Canadian Nuclear Safety Commission (CNSC) is Canada's nuclear regulatory agency operating under the *Nuclear Safety and Control Act* (NSCA). The CNSC regulates the use of nuclear energy and materials to protect the health, safety and security of Canadians and the environment, and to meet Canada's international commitments on the peaceful use of nuclear energy.

In order to reduce the risk to a reactor project from licensing-related changes, the CNSC (as do other regulators) offers a pre-project review of reactor designs. While not legally binding, the review provides valuable feedback to a vendor on the design well in advance of an application for a construction licence.
The objectives of a pre-project design review are to:

- assess whether a reactor design is, at an overall level, compliant with the CNSC regulatory requirements;
- assess whether the design meets the CNSC's expectations for new nuclear power plants in Canada; and
- identify potential fundamental barriers to licensing a reactor design in Canada.

To achieve the above stated objectives, the CNSC staff assesses the safety and security aspects of the design. This review provides an opportunity for the CNSC staff to assess the design prior to any licensing activities, and to identify potential issues for resolution related to the compliance of the design with regulatory requirements and expectations. Pre-project reviews help increase regulatory certainty, reduce risk of licensing delays, and ultimately contribute to public safety.

A pre-project review has been completed for both ACR-1000 (Phases 1 and 2) and EC6 (Phase 1).

3. ACR-1000 Pre-Project Regulatory Review

The pre-project review consisted of two phases, starting on April 1, 2008 and ending on August 30, 2009, with a third (optional) phase initiated by AECL as follow-up of Phase 2 (Section 3.3).

3.1 Phase 1

Phase 1 was an overall assessment of the information submitted in support of the ACR-1000 design against CNSC regulatory requirements and regulatory documents. Its purpose was to determine whether the design intent is compliant with the CNSC requirements and meets CNSC's expectations for the design of new nuclear power plants in Canada [1].

The review included the following 16 focus topic areas: safety principles (defence in depth); specific design expectations of structures; systems and components important to safety; reactor core nuclear design; fuel design; means of reactor shutdown; robustness of the design against malevolent acts; safety analysis; human factors engineering; radiation protection; protection from fire; protection against out-of-core criticality; quality assurance; safeguards; and security.

The CNSC stated the following conclusion at the end of Phase 1 review: *"At an overall level the design intent is compliant with the CNSC regulatory requirements and meets the expectations for new nuclear power plants in Canada"*.

3.2 Phase 2

This phase went into further review of design details with a focus on identifying whether there are any potential fundamental barriers to licensing the design in Canada. The review was conducted in 17 focus topic areas, the same as stated above in Phase 1, plus the supporting R&D.

The main report submitted to the CNSC was a Generic Safety Case Report (GSCR), which provided an integrated summary of the design and safety analysis, and which followed the format and content of a Preliminary Safety Analysis Report (PSAR). The GSCR included design details and descriptions of all major structures, systems and components, accompanied with bounding deterministic safety analysis, Level 1 PSA analysis, and description of the deterministic and probabilistic analysis methodologies and computer tools used.

The CNSC stated the following conclusion at the end of the Phase 2 review: "*CNSC staff's review of the 17 focus areas did not identify any fundamental barriers to licensing the ACR-1000 design in Canada, subject to the successful and timely completion of outstanding R&D, and the resolution of key findings in the focus areas. CNSC staff has provided detailed comments in each of the 17 focus areas and these comments are related to work that staff recommends should be completed before a construction licence decision is made by the Commission"*.

CNSC staff considers an adequate R&D program to be of critical importance in support of new features of the reactor design since the ACR-1000 design contains a number of new features that require R&D. The CNSC review found that the overall ACR-1000 R&D program was derived logically from the existing knowledge base and was comprehensive and adequate. All key safety-related R&D will be completed prior to the submission of an application for a licence to construct.

Safety management during the design process is necessary to ensure that safety is embedded into the design in a conservative, systematic and structured way. CNSC staff reviewed AECL's design process including the Quality Assurance Manual for the ACR-1000 project, and conducted an audit at the engineering offices of AECL. Overall, CNSC staff concluded that an adequate design process was in place.

3.3 Phase 3

The Phase 3 review started in September 2009 and is scheduled to be completed by end of November 2010. The main objective of the Phase 3 pre-project review is to provide a follow up of selected topics in Phase 2. Successful implementation of Phase 3 should simplify the Construction Licence review.

The Phase 3 review is being conducted in the following focus topics areas: classification of structures, systems, and components, sensitivity/uncertainty analysis using TSUNAMI method for ACR physics applications, reactor core nuclear design, fuel design, on-line preventive maintenance strategy, severe accidents, probabilistic safety assessment, human factors, QA actions following Phase 2, safety analysis methodology, and R&D in support of emergency core cooling system design.

4. EC6 Phase 1 Pre-Project Regulatory Review

The EC6 pre-project Phase 1 review started on April 1, 2009, and was completed at the end of March 2010. For the Phase 1 review, the CNSC staff selected 17 review topics to assess the EC6 design, similar to the ones described above for the ACR-1000 pre-project review.

The CNSC stated the following conclusion in the Phase 1 review*:"At an overall level the design intent is compliant with the CNSC regulatory requirements and meets the expectations for new nuclear power plant designs in Canada. This conclusion would be further confirmed during a Phase 2 review when required information for open specific technical items identified for each review topic will be fully addressed. The CNSC staff anticipates that these items could be brought to closure during a Phase 2 review".*

5. Conclusions

The regulatory pre-project reviews conducted by the CNSC on the ACR-1000 and EC6 designs help increase regulatory certainty, reduce the risk of licensing-related delays in implementation of these designs in Canada and internationally and ultimately contribute to public and environmental safety.

CNSC has completed Phase 1 and Phase 2 pre-project reviews of ACR-1000, and Phase 1 pre-project review for EC6. CNSC concluded that both designs meet the Canadian regulatory requirements and expectations for new builds in Canada.

References

[1] CNSC Regulatory Document, RD-337, "Design of New Nuclear Power Plants", Published on November 2008.

Developing National Regulations in the United Arab Emirates

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Abstract

The Federal Authority for Nuclear Regulation (FANR), in preparing, issuing and implementing regulations is seeking to be consistent with IAEA Safety Standards, to use risk informed and performance-based methodologies, to capitalise on licensing in the country of origin and to follow internationally recognised practices. FANR's intent is to produce high level regulations which are not prescriptive and which focus on the essential aspects of safety. Regulatory guides will also be provided to assist licensees with compliance.

This paper discusses an overview of the regulatory framework in the UAE, the planned scope of the proposed regulations, the approach being taken under an internal management system to develop these regulations and regulatory guides in the UAE and. The current status and future plans will also be provided.

1. Background

The White Paper on *Peaceful uses of Nuclear Energy* issued in 2008[1] set out the case for nuclear energy by showing that national annual peak demand for electricity is likely to rise to more than 40,000MW by 2020, reflecting a cumulative annual growth rate of roughly 9% from 2007 onward. The White Paper also concluded that nuclear power was the optimal means for meeting this demand increase in an environmentally appropriate manner.

In September 2009 the UAE President, H.H. Sheikh Khalifa bin Zayed Al Nahyan, enacted Federal Law by Decree No. 6 of 2009, Regarding the Peaceful Uses of Nuclear Energy*, which establishes* the Federal Authority for Nuclear Regulation (FANR) as the UAE's nuclear regulatory body. The Law provides that the Authority shall be managed by a Board of Management comprising not less than five members in addition to a Chairman and deputy chairman. H.E. Dr. Ahmed Mubarek Al Mazrouei was appointed Chairman and H. E. Abdulla Nasser Al Suwaidi, Deputy Chairman.

The Abu Dhabi government subsequently established the Emirates Nuclear Energy Corporation (ENEC) as the Nuclear Energy Program Implementation Organization (NEPIO), as recommended by the International Atomic Energy Agency (IAEA). ENEC is charged with the establishment of a peaceful nuclear programme that delivers the benefits of nuclear power to the people of the UAE.

The FANR, as currently established, two divisions; operations and administration. Within the operations division are four departments; radiation safety, security, safeguards and nuclear safety. Based on the type of regulation or regulatory guide to be issued, each department is responsible for drafting the technical bases in their respective areas while the overall process is managed within the nuclear safety division.

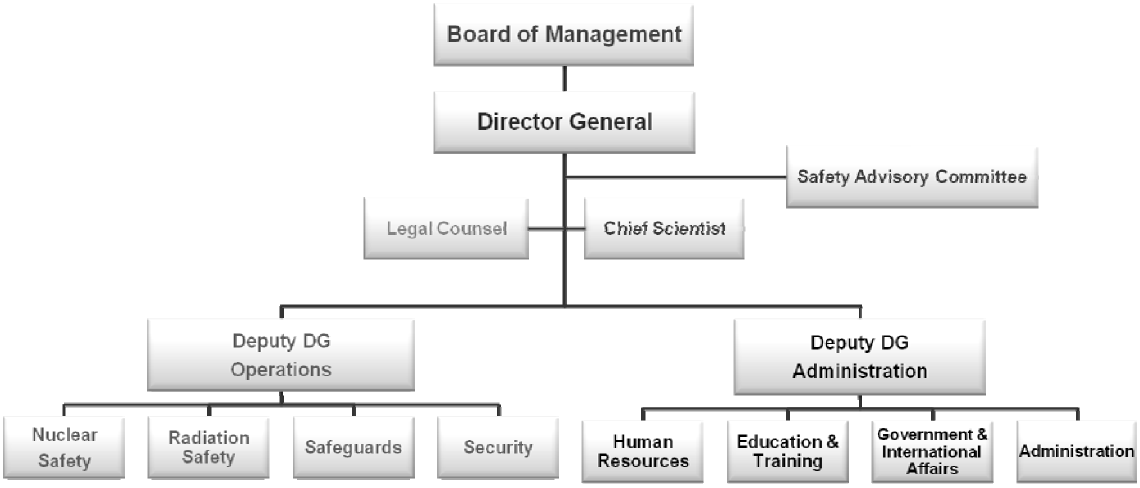

Figure 1 – FANR Organisation

The FANR staff developed a schedule and list of priority regulations for completion. The process used was part of an overall FANR Integrated Management System (IMS), based on IAEA Standards [2]. The regulations and regulatory guide framework is one of the core processes in the IMS. Currently the forecast is to complete the majority of regulations by the end of 2010.

2. Law by Decree No. 6 Specifics

Article 4 of the Decree established a public organisation under the name of "Federal Authority for Nuclear Regulation (FANR), as a fully independent body with the objective to regulate and develop the Nuclear Sector in the State towards peaceful purposes (only) and to ensure Safety, Nuclear Security and Radiation Protection."

Article 6 of the Decree states "The Authority shall be exclusively responsible for issuing all Licenses to practice any of the Regulated Activities in the State and any other licence stipulated in this Law by Decree, its implementing regulation or any other regulation issued by the Authority or amending, suspending, revoking such Licenses or refusing to grant it, provided that such refusal is reasoned. The Authority may impose condition on Licenses pursuant to this Law by Decree, its implementing regulation issued hereby."

Article 14 directs that the Board shall appoint a Director General to exercise the functions specified in the Law by Decree. Dr. William Travers, former Executive Director of Operations for the NRC, was appointed as the first Director General of FANR.

Article 38 states:

1. The Board shall issue the regulations specifying the requirements which all Operators must comply with and follow.

2. The Authority shall prepare explanatory guidelines on how to comply with the regulations

3. In developing regulations and guidelines, the Authority shall take into consideration comments from stakeholders, information made available by experts and internationally recognised standards and recommendations such as IAEA Safety Standards.

3. Regulated Activities

The Law by Decree No. 6 enumerates a number of regulated activities that must be licensed by the Authority. FANR's approach in developing national regulations focuses on the following main objectives:

- conform with IAEA Safety Standards,
- be risk informed, performance based,
- capitalise on licensing by the vendor country of origin, and
- follow other internationally recognised practices.

The intent is to produce **high level regulations, which are not prescriptive**, focusing on the essential aspects of safety.

One of the first steps taken by the Authority in early development was to review in detail the provisions of the Law by Decree and establish a list of the regulations that would be needed to carry out its mission. The regulated activities as established in the Law by Decree were benchmarked against those applied by other regulatory bodies and with international standards and guidance material such as IAEA and WENRA. Following this, specific regulations were identified. The initial result identified 20 potential regulations (those shown in italics were determined to have high priority).

- *Siting of Nuclear Facilities*
- *Design of Nuclear Facilities*
- *Radiation Dose Limits & Optimisation of Radiation Protection for Nuclear Facilities*
- *Application for a License to Construct a Nuclear Facility*
- *Nuclear Facility Construction*
- *Physical Protection including Access Controls*
- *Import / Export Controls*
- *Safeguards and Material Control and Accounting*
- *Radiation Protection and Radioactive Waste Management for Nuclear Facilities*
- *Emergency Preparedness at a Nuclear Facility*
- *Transportation of Radioactive Materials*
- *Application for a License to Operate a Nuclear Facility*

- *Design Modifications during Operation*
- *Operational Safety including Testing, Surveillance and Reporting*
- *Certification of Operations Personnel*
- *Administrative Liabilities and Penalties*
- *Criminal Penalties*
- *Application for Operating License Extension of a Nuclear Facility*
- *Decommissioning*
- *Decommissioning Trust Fund*

As the focus shifted to development of regulations and the discussions evolved, the staff began debating the different possibilities regarding the issues of risk informed performance based and quality assurance and the integrated management approach. Two viewpoints emerged on the issues of Management Systems and Probabilistic Risk Assessment[3]. Either of these issues could be handled within the different regulations as needed, or separate regulations could be written to cover the requirements. FANR decided on the latter option and has pioneered regulations specifically dealing with Management Systems and PRA.

Other discussion also took place concerning activities being undertaken in the Radiation Protection area and two more regulations were identified. Accordingly, FANR has added the four regulations below to the list of regulated activities:

- *Management Systems for Nuclear Facilities*
- *Probabilistic Risk Assessment*
- *Basic Safety Standards for Facilities and Activities involving Ionising Radiation*
- *Security of Radioactive Sources*

4. Process

Prior to the passage of the UAE nuclear law, the initial FANR Staff and a group of consultants starting setting up a management process to develop the regulations and regulatory guides that would be needed. The regulations and regulatory guides are developed following a Core Process within the FANR IMS. The process covers 13 steps starting with identifying the need for a regulation, through the drafting and internal/external reviews, resolution of comments and final approval by the FANR Board of Management. Each step of the process is clearly defined and documented and where necessary added procedures are

[3] FANR has chosen to use the term PRA whereas some countries use the term PSA.

provided to give further direction. The Section Head for Regulations and Guides, in the Nuclear Safety Department, works with the lead department to guide each document through the process.

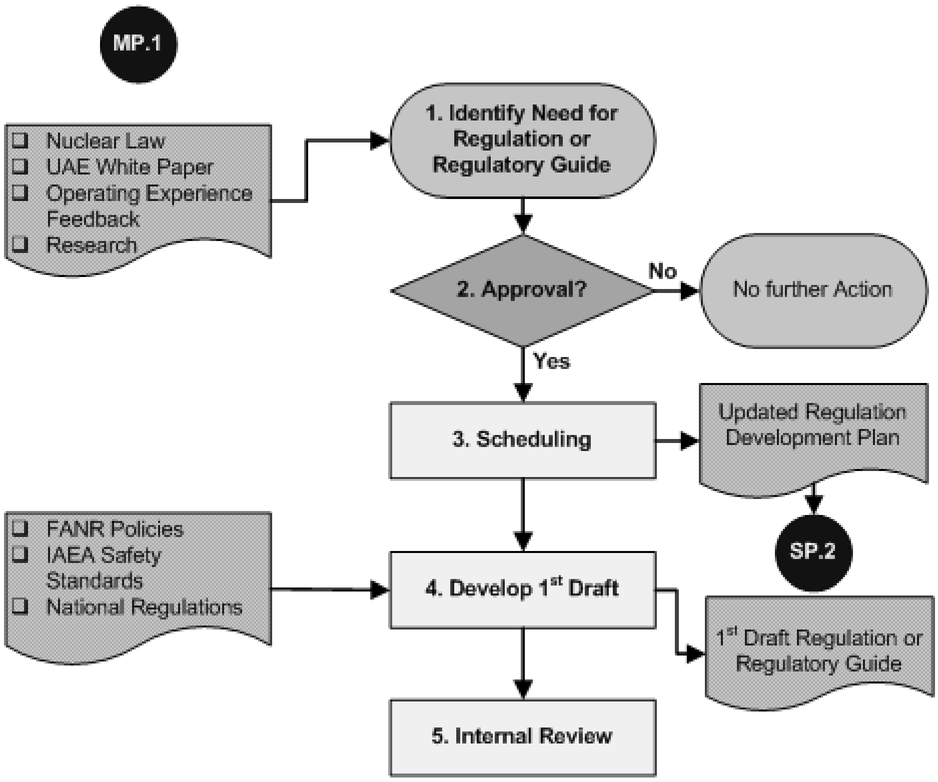

Figure 2 - CP.1 Process (Steps 1 through 5)

Key steps in the process are summarised below:

- The starting point for the development of all regulations is the IAEA Standards and Guides. If no IAEA standards or guides exist other international or national sources are used. Regulations are first developed in English and after receiving internal approval, legal review and editing they are submitted for translation into Arabic. The terminology is based on the IAEA Safety Glossary (English and Arabic versions) expert review and legal opinion.

- The starting point for regulatory guides is the guidance developed by the “country of origin” (i.e., Korea) but it also includes an evaluation of IAEA guidance and other best practices. Guidance is either adopted if it already exists and is suitable, or is developed using input from these sources as appropriate.

- Internal review process requires achieving consensus among all involved technical departments and the approval of the deputy director general and director general. Based on the timing and scheduling advance copies are unofficially transmitted to the licensee for their initial review.

- Unless restricted for security concerns, all regulations and regulatory guides are processed through a 2-step external stakeholder review. The first step provides governmental ministries and agencies a 30-day period to review these documents and provide their comments. Following resolution of these comments, and based on the recommendation of the Director

General of the FANR and agreement by the BoM, the document(s) are then posted on the FANR web site for public comment for another 30 days.

- The final regulations are submitted to the BoM for approval and then posted in the official government "Gazette" and on the FANR website (http://fanr.gov.ae). Regulatory Guides do not require approval of the BoM or posting in the Gazette and are approved directly by the Director General.

5. Current Status

At the end of June 2010, the FANR BoM formally approved Regulation (FANR-REG-04), Regulation for Radiation Dose Limits & Optimisation of Radiation Protection for Nuclear Facilities. During July and August Regulations 01 (Management Systems); 02 (Siting); 03 (Design); 05 (PRA); 06 (Application for a License to Construct a Nuclear Facility); 08 (Physical Protection); 11 (Radiation Protection and Radioactive Waste Management); 12 (Emergency Preparedness); 13 (Transportation of Radioactive Materials); and 24 (Basic Safety Standards for Facilities and Activities involving Ionising Radiation) completed the internal and external review process.

6. Conclusions

This paper provided a brief summary of work undertaken by the Federal Authority for Nuclear Regulation (FANR) of the United Arab Emirates over the past 2 years, prior to and after the approval of the Federal Law by Decree No. 6 to develop a comprehensive set of regulations and guides. The work has followed the concepts laid out in the initial "white paper" to conform to equivalent IAEA Safety Standards, be risk informed performance based, capitalise on certification by country-of-origin, and incorporate other internationally recognised practices.

The use of a step by step process within an integrated management system setup for FANR has enabled the staff to have an efficient and effective procedure and methodology to use in the development and approval of regulations and regulatory guides.

References

1. The UAE Policy on the *Evaluation and Potential Development of Peaceful Nuclear Energy* committed the UAE to complete operational transparency and international cooperation for any nuclear program (Spring 2008)

2. IAEA Safety Standards, GS-R-3, The Management System for Facilities and Activities (2006)

3. IAEA Safety Glossary, Terminology Used in Nuclear Safety and Radiation Protection (English 2007), Arabic (2007)

Exploitation of BEPU Approach for the Licensing Process

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Within the licensing process of the Atucha II PHWR (Pressurized Heavy Water Reactor) the BEPU (Best Estimate Plus Uncertainty) approach has been selected for issuing of the Chapter 15 on FSAR (Final Safety Analysis Report). The key steps of the entire process are basically two: a) the selection of PIE (Postulated Initiating Events) and, b) the analysis by best estimate models supported by uncertainty evaluation. The key elements of the approach are: 1) availability of qualified computational tools including suitable uncertainty method; 2) demonstration of quality; 3) acceptability and endorsement by the licensing authority. The effort of issuing Chapter 15 is terminated at the time of issuing of the present paper and the safety margins available for the operation of the concerned NPP (Nuclear Power Plant) have been quantified.

1. Introduction

Among the general attributes of a methodology to perform accident analysis of a nuclear power plant for licensing purposes, the very first one should be the compliance with the established regulatory requirements.

A second attribute deals with the adequacy and the completeness of the selected spectrum of events which should consider the combined contributions of deterministic and probabilistic methods.

The third attribute is connected with the availability of qualified tools and analytical procedures suitable for the analysis of accident conditions envisaged in the concerned Nuclear Power Plant. Thus, a modern and technically consistent approach has been built upon best estimate methods including an evaluation of the uncertainty in the calculated results (Best Estimate Plus Uncertainties or BEPU approach).

The complexity of a NPP and of the accident scenarios may put a challenge for a conservative analysis and may justify the choice for a BEPU approach in the licensing process. This implies two main needs: the need to adopt and to prove (to the regulatory authority) an adequate quality for the computational tools and the need for the uncertainty.

The purpose of the present paper is to outline key aspects of the BEPU process aimed at the licensing of the Atucha II NPP in Argentina. The Atucha II is a heavy-water cooled heavy-water moderated, vessel type, pressurized reactor. The moderator fluid has the same pressure as the coolant fluid, but temperature is lower. Fuel channels, which do not withstand pressure difference during nominal operation, separate the coolant from the moderator. The thermal power produced in the moderator is used to pre-heat the feed-water.

A direct link with the bases of nuclear reactor safety shall be ensured by the "BEPU-description document". In the present case this is formed by the following main elements or steps:

1) Evaluation of the possibility to use a BE estimate within the context of the current national (i.e. of the Country where the NPP is installed) Regulatory Authority (RA) requirements. A pre-application document was submitted to the national RA. This included the consideration of past interactions between the RA and the applicant as well as the analysis of the licensing practice in the Country where the NPP was designed.

2) Outline of international practices relevant for the proposed approach. The experiences acquired in the use of Best Estimate analyses for licensing purposes are reviewed: this is true for probabilistic and deterministic analyses and specifically for the determination of radiological consequences.

3) Structure of the BEPU: a) categorization of PIE, b) grouping of events, c) identification of analysis purposes, d) identification of applicable acceptance criteria, e) setting up of the 'general scope' Evaluation Model (EM) and of related requirements starting from the identification of scenario related phenomena, f) selection of qualified computational tools including assumed initial and boundary conditions, g) characterization of assumptions for the Design Basis Spectrum, h) performing the analyses, i) adopting a suitable uncertainty method.

4) Under the item 3g): the roadmap pursued for the analysis foresaw the use of nominal conditions for the NPP parameters and the failure of the most influential system. The implementation of such roadmap implied the execution of preparatory code run per each scenario where all NPP systems were simulated. This also required the simulation the control and the limitations systems other than the protection systems. Once the "nominal system performance in accident conditions (following each PIE)" was determined, it was possible to select the worst failures and calculate a new (i.e. the "binding one") accident scenario.

5) Under the general scope of item 3e): several computer codes and about two dozen nodalizations have been used, developed and, in a number of cases, interconnected among each other.

6) Qualification was necessary for the computational tools mentioned under item 5), within the framework depicted under item 3). The issue constituted by qualification of code-nodalization user was dealt with in the same context. Specific methods or procedures including acceptability thresholds have been developed and adopted.

7) Under the scope of item 3i): the uncertainty method based on the extrapolation of accuracy, developed at University of Pisa since the end of 80's, was used to create the CIAU (Code with capability of Internal Assessment of Uncertainty) and directly used for quantifying the errors in the calculations, as needed.

The purpose of the present paper is to present an outline of the BEPU approach. At the time of preparing of the present paper a “rev.0” version of the Chapter 15 of the Atucha II FSAR has been issued. However, results are under preliminary scrutiny before being transmitted to the Regulatory Authority. Owing to this, no final results from the BE analysis of transients shall be expected in the paper.

2. Aspects for the Application of the BEPU Approach

The BEPU approach has been adopted as the methodology for accident analyses covering the established spectrum of PIE. Procedures have been applied to derive the list of PIE and to identify applicable acceptance criteria. Finally, the application of computational tools including nodalizations required suitable boundary and initial conditions and produced results related to the Atucha II transient scenarios originated by the PIE.

The proposed BEPU approach follows current practices on deterministic accident analyses, but includes some key features to address particular needs of the application. The approach takes credit of the concept of Evaluation Models (EM), and comprising three separate possible modules depending on the application purposes:

- For the performance of safety system countermeasures (EM/CSA).
- For the evaluation of radiological consequences (EM/RCA).
- For the review of components structural design loadings (EM/CBA).

where the acronyms CSA, RCA and CBA stand for “Core Safety Analysis”, “Radiological Consequence Analysis” and “Component Behaviour Analysis”. It may be noted that structural resistance of Containment as well as mechanical loads on RPV (Reactor Pressure Vessel) internals are calculated in the frame of CBA.

The selection of contents for the present section has been derived based on the US NRC Regulatory Guide 1.70, ref. [1], the US NRC Standard Review Plan, ref. [2], design industry safety documents, e.g., ref. [3], the FSAR of recently licensed NPP and the so called (Atucha II specific) BEPU report, already endorsed by the involved Licensing Authority, ref. [4].

The evaluation of the safety of nuclear power plant Atucha II does include required analyses of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. For these purposes, two complementary methodologies for safety analysis are applicable. The scope of accident analyses presented in Chapter 15 of the FSAR, however, comprises only deterministic safety analyses. Probabilistic safety analyses are presented in a separate document.

The Chapter 15 sections document the results of the performed deterministic safety analysis covering a sufficiently broad spectrum of transients and accidents (i.e. PIE), aiming at demonstrating that the plant can be safely operated within the established regulatory limits related to the integrity of the components, to the preservation of the safety functions and the barriers against radioactivity releases and to the related radiological impact.

In order to confirm that the plant transient and accident analyses represent a sufficiently broad spectrum of initiating events, the transients and accidents are categorized according their expected frequency of occurrence and grouped in nine families according to the type of challenge to the fundamental safety functions. The results of these safety analyses also provide a contribution to the selection of limiting conditions for operation, limiting safety systems settings, and design specifications for components and systems to protect public health and safety of the installations.

2.1. The basis for BEPU

A simplified flowchart of the rationale that brought to the planning and the application of the BEPU approach is given in Fig. 1 (details can be found in ref. [4]). The steps followed by the proposed approach can also be derived from the analysis of the diagram.

In the first step, as a function of the selected scenario and of the purpose of the analysis, the complexity of the evaluation model may range from a simplified qualitative evaluation (EM/QA) to a complete combination of the three possible modules (EM/CSA + EM/RCA + EM/CBA).

In order to evaluate the plant safety performance, acceptance criteria are properly selected according to estabilished international practice. The two main aspects which have been considered for developing the evaluation model with the ability of adequately predict plant response to postulated initiating events are intrinsic plant features and event-related phenomena characteristics.

For the two modules EM/CSA and EM/CBA, the first set of requirements for the evaluation model is imposed by the design characteristics of the nuclear power plant, its systems and components. Requirements on the capability of simulating automatic systems are of particularly importance for anticipated operational occurrences, in which control and limitation systems play a key role on the dynamic response of the plant.

It shall be noted that the concerned modeling features are consistent with the requirements that imposes the design of the limitation system according to the same standard as the reactor protection system. However, this rule does not apply to control systems. Nevertheless, the best response of the plant cannot be calculated without the detailed modeling of the control system. This has been considered in the present framework.

The second set of requirements is derived from the expected evolution of the main plant process variables and the associated physical phenomena. For the proposed approach, this is performed through the process of identifying the Phenomenological Windows (Ph.W) and the Relevant Thermal-hydraulic Aspects (RTA). The relevant timeframe for the event is divided into well defined intervals when the behaviour of relevant safety parameters is representative of the physical phenomena.

For the adequate simulation of the identified phenomena, computational tools were selected from those which have previous qualification using an appropriate experimental data base. Satisfactory qualification targets provide basis for acceptability of the postulated application.

Within the framework of the present FSAR chapter, the expression "computational tools" comprises:

- The best estimate computer codes.
- The qualified detailed nodalizations for the adopted codes including the procedures for the development and the qualification.
- The established computational methods for uncertainty quantification including the procedure for the qualification.
- The computational platforms for coupling and interfacing inputs and outputs from the concerned codes and nodalizations.

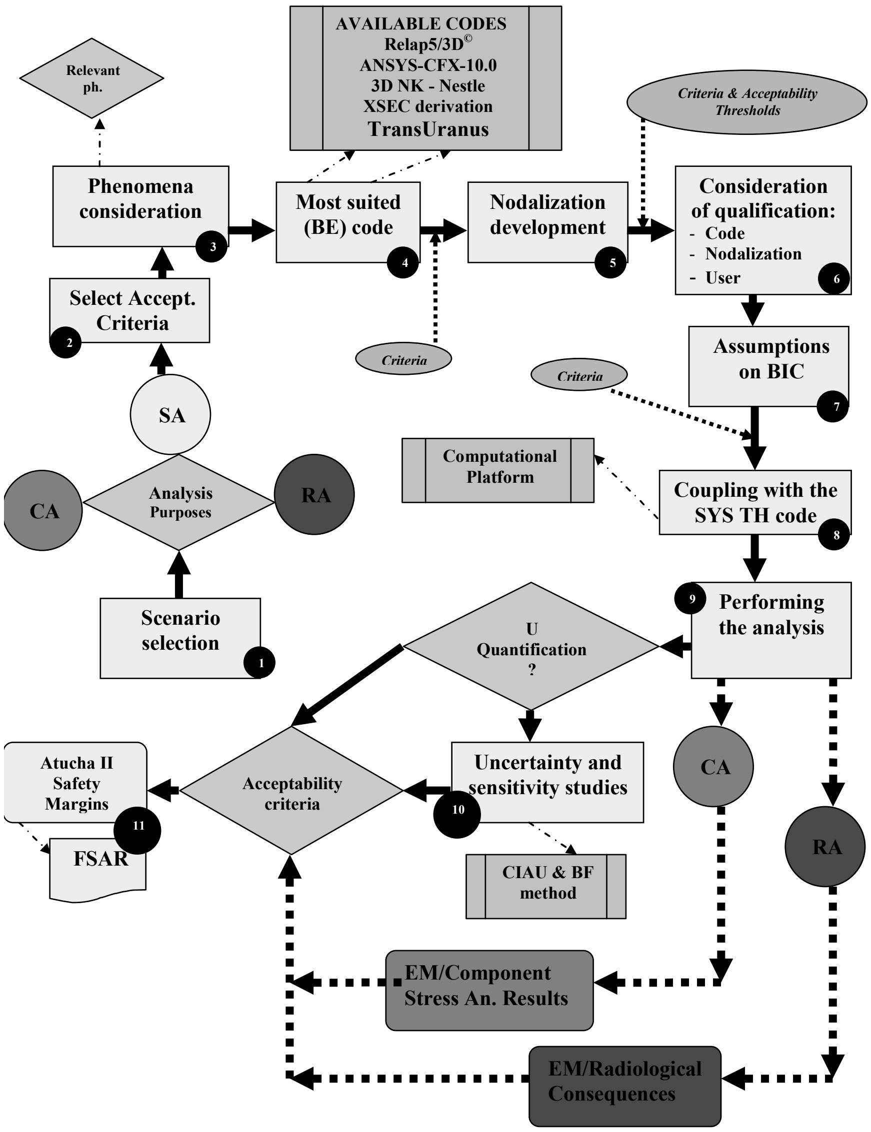

Fig. 1 – The BEPU Flow-Diagram.

3. Categorization of PIE

The design philosophy of Atucha II incorporates the principle that plant states that could result in high radiation doses or radioactive releases are of very low probability of occurrence, and plant states with significant probability of occurrence have only minor or no radiological consequences.

Accordingly, for design purposes, postulated initiating events are divided into the following event categories by their anticipated probability of occurrence, consistently with Probabilistic Safety Analysis (PSA) performed for the same NPP:

Anticipated Operational Occurrences (AOO)	Probability greater than 10^{-2} / year
Design Basis Accidents (DBA)	Probability less than 10^{-2} / year and greater than 10^{-5} / year
Selected Beyond Design Basis Accidents (SBDBA), including Anticipated Transients Without Scram (ATWS) and "extended spectrum" of LOCA (Loss of Coolant Accident)	Probability less than 10^{-5} / year

Accident conditions which stand out of these ranges of probabilities or that are not included in the SBDBA category, may also involve significant core degradation. These are out of the scope of this chapter and are treated separately within the frame of PSA studies.

The third event category (SBDBA) appears to be specific of the Atucha II FSAR and addresses large break LOCA and ATWS. The rationale for introducing this category derives from the design characteristics of the NPP and from previously agreed licensing steps (see also ref. [4]).

The categorization of large break LOCA as SBDBA is due to the exclusion of the maximum credible accident from the range of the design basis spectrum for Atucha II, and the adoption of the break size of ten percent on reactor coolant pipe (0.1 A) as the basis for fulfilling traditional regulatory requirements. So far, the double ended guillotine break is considered as a beyond design basis scenario.

Nevertheless, the demonstration of the design capability to overcome this event has still a relevant role in the safety performance evaluation. For this aim, however, currently used conservative approach for safety analysis may not be sufficient to guarantee that safety margins still exist. The use of best estimate methods is acceptable when a scenario is categorized as beyond design basis.

Regarding ATWS, similarly to some modern or evolutionary nuclear power plants, Atucha II design does present a diverse scram system (Fast Boron Injection System). In this sense, the original safety issue related to ATWS does not constitute a safety concern applicable to its design.

All selected scenarios are grouped in the nine families of events: each family covers events with similar phenomena, or events in each family are characterized by similarity of challenges in relation to the fundamental safety functions. The nine families are:

1. Increase in heat removal by the secondary system.
2. Decrease in heat removal by the secondary system.
3. Decrease in heat removal by the primary system.
4. Reactivity and power distribution anomalies.
5. Increase in reactor coolant inventory.
6. Decrease in reactor coolant inventory.
7. Radioactive release from a subsystem or component.
8. Disturbance in the refueling system and fuel storage system.
9. Anticipated transients without scram (ATWS).

An excerpt of the list including the description of 83 events is provided in Table 1 below. This also includes the type of analysis to be performed in relation to each transient. In this connection, three possible types of general evaluation purposes are foreseen for each scenario:

RCA those scenarios whose radiological impacts have to be calculated.

CSA those scenarios which are used for the design of safeguards or countermeasures (systems performance associated with the integrity limits for the barriers against radioactive releases).

CBA those scenarios which are used for reviewing the design of components or structures for stability or integrity (mechanical design loadings).

Table 1 – Excerpt from the List of PIE for Atucha II Chapter 15 of FSAR.

No	Transient	Section FSAR	Adopted Evaluation Model	Class of Accident
Increase in Heat Removal by the Secondary System		**15.1**		
2	FW System Malfunctions that result in an Increase in FW Flow (Stuck Open FW Control Valve)	15.1.2	CSA	AOO
Spectrum of Steam System Piping Failures inside and outside of Containment (MSLB)		15.1.5	-	
5	Leak of MS Line inside the Containment	15.1.5.1	CSA/RCA/CBA	DBA
9	Inadvertent Closing of the Moderator Cooler Bypass CV	15.1.7	CSA	AOO
36	Uncontrolled CR Withdrawal at the particular Power Level that yields the most Severe Results ◙	15.4.2	CSA	AOO
41	Spectrum of Rod Ejection Accidents ◙	15.4.7	CSA	DBA
Spectrum of SGTR		15.6.3	-	
46	Single SG Tube Rupture (*"Bordihn": SG Tube Failure*)	15.6.3.1	CSA	DBA
56	0.1A LOCA cold with Sump Swell Operation	15.6.5.1.2.4	QA	DBA
Large Break LOCA		15.6.5.1.3	-	
57	2A LOCA cold (*DEGB. Different Break Sizes and Positions are investigated*) ◙	15.6.5.1.3.1	CSA/RCA/CBA	SBDBA
72	Leakage on the Refueling Machine and Auxiliary Equipment	15.8.2	RCA	DBA
Anticipated Transients Without Scram (ATWS)		**15.9**		
74	Mechanical Failure of the Control Rods in case of Emergency Power Mode	15.9.1	CSA	SBDBA

In relation to anticipated operational occurrences (AOO), it has to be proved that they do not propagate into accidents. Additionally, the analysis shall demonstrate that the systems actuated by operational instrumentation and control systems and by limitation and reactor trip systems are sufficiently effective to:

- Maintain the integrity of the barriers against radioactivity release, as no fuel centerline melting, unrestricted continued operation of fuel assemblies, and ensured integrity of the reactor coolant pressure boundary (CSA related evaluation purposes).
- Maintain component loadings within the allowable limits for this category of events as it is addressed in the FSAR Chapters 4 to 6 (CBA related evaluation purposes).
- Prevent radioactive releases to the environment in excess of the allowable limits for this category of events (RCA related evaluation purposes).

For design basis accidents, even though they are not expected to occur, only limited consequences are accepted. For DBA it has to be demonstrated that the safety system countermeasures actuated by the reactor protection system are sufficiently effective to:

- Maintain adequate integrity of the barriers against radioactivity release, as limited fuel centerline melting, limited loss of integrity of fuel cladding, or integrity of the containment (CSA related evaluation purposes).
- Maintain component loadings within the allowable limits for accident conditions, and may be addressed in the FSAR Chapters 3 to 6 (CBA related evaluation purposes).
- Prevent radioactive releases to the environment in excess of the allowable limits for accident conditions (RCA related evaluation purposes).

For the SBDBA, the aim of the analyses is to demonstrate that measures for mitigation of consequences are sufficient and effective to:

- Ensure residual heat removal, maintaining sufficient integrity of the barriers against radioactivity release (CSA related evaluation purposes)
- Prevent radioactive releases to the environment in excess of the allowable limits for accident conditions (RCA related evaluation purposes).

In order to complete the set of targets for the analyses, event specific purposes are added, considering scenario-related safety system countermeasures or performance, as well as challenged component structural limits. To assess plant safety performance, figures of merit are derived for each purpose of the considered event.

4. Adopted Computational Tools

The computational tools include a) the best estimate computer codes; b) the nodalizations including the procedures for the development and the qualification; c) the uncertainty methodology including the procedure for the qualification; d) the computational platforms for coupling and interfacing inputs and outputs from the concerned codes and nodalizations.

An idea of the interaction among the considered computational tools can be derived from Fig. 2 and Table 2, both dealing with codes, category a) above. The following to be noted:

- A chain of codes is needed for exploiting the three-dimensional neutron kinetics capability of the Nestle code.
- MCNP code has the role of providing ‘reliable-reference’ results at the steady state condition.

- Melcor is used as a back-up code to support the application of the Relap5-3D © when modeling the containment.
- The "ultimate" code for calculating the PTS risk, deterministic analysis, is Ansys.
- Dynetz is "intimately" coupled with Relap5/3D ©: however, the entire control, limitation and protection systems of Atucha II are modeled and interaction with the thermal-hydraulic code is foreseen at each time step.

Table 2 – List of Codes Used for BEPU Accident Analyses.

No	Code Type	Code Name
1	System Thermal-Hydraulics	Relap5/3D © (TH model) including DNBR and containment
2	I&C Modeling	Dynetz
3	Computation Fluid Dynamics	CFX
4	Structural Mechanics	Ansys
5	Fuel (mechanics)	Transuranus
6	Neutron Physics (and supporting)	Nestle
7		Helios
8		MCNP
9		Scale Package: Newt-Origen (Triton), burn-up oriented
10		Scale Package: Keno, static 3D neutron physics
11		NJOY
12		Dragon
13	Radiological Consequences (and supporting)	MCNP-Origen for radioactivity source-term
14		Relap5/3D © (Radiological model)
15		Melcor-Maccs
16		Arcon96
17		Rodos

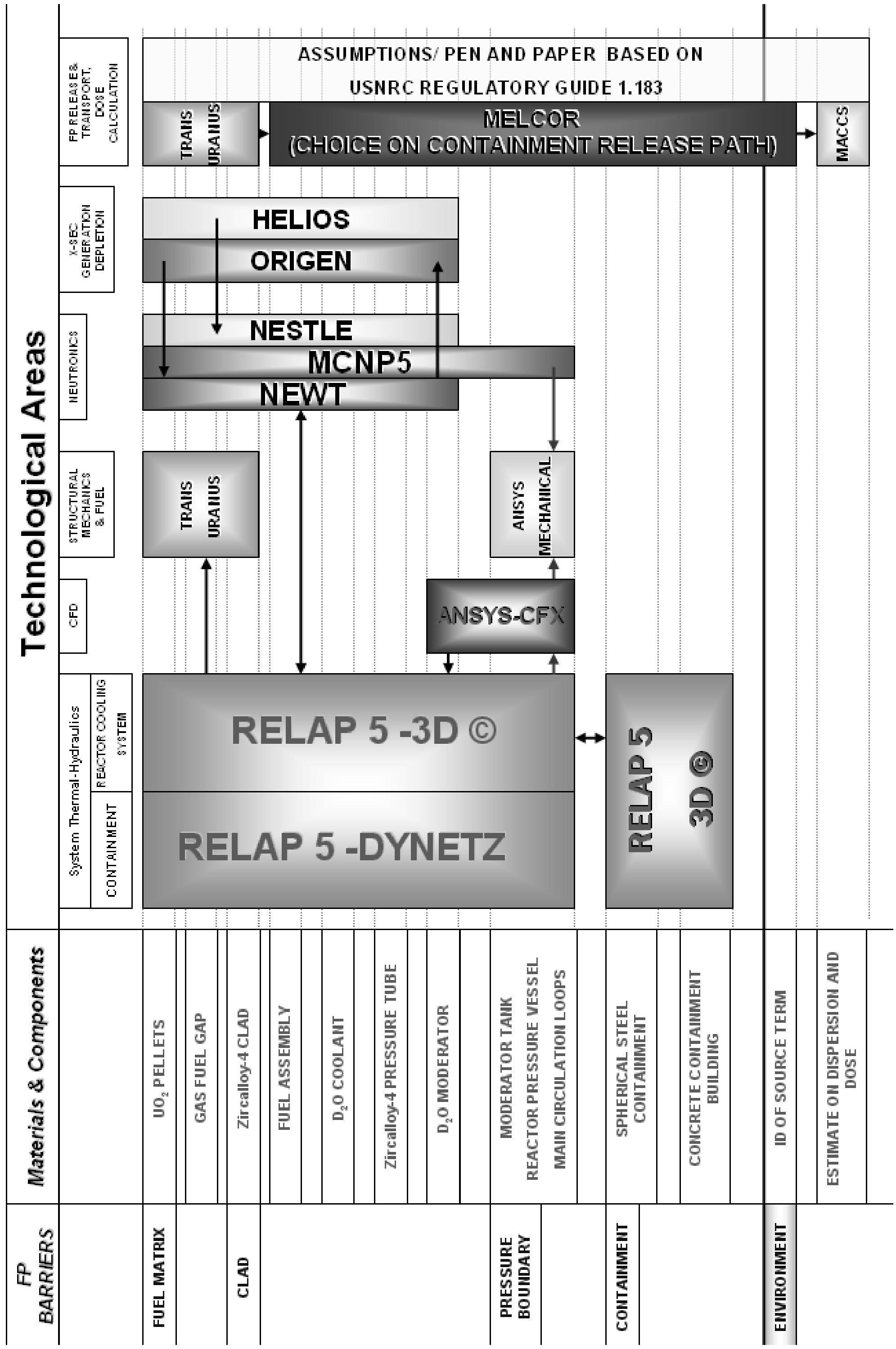

Fig. 2 – The Interaction among the Computer Codes used for BEPU Accident Analyses.

4.1. The Qualification

A key issue for the BEPU is represented by the qualification. This shall be demonstrated for each of the four categories of computational tools discussed above. It is out of the scope of the present paper to provide details adopted to show the achievement of a suitable level of qualification. However, an idea can be derived from the section below dealing with UMAE, i.e. Uncertainty Method based upon Accuracy Extrapolation (here used to demonstrate the qualification of the thermal-hydraulic nodalizations).

4.2. The Uncertainty Method

In principle, whenever a best estimate method is applied for licensing purposes, uncertainty quantification is needed. Therefore the UMAE-CIAU procedure, or even the CIAU having UMAE as "informatics engine", is used in the present context, ref. [4].

The UMAE is the prototype method for the consideration of "the propagation of code output errors" approach for uncertainty evaluation. The method focuses not on the evaluation of individual parameter uncertainties but on the propagation of errors from a suitable database calculating the final uncertainty by extrapolating the accuracy from relevant integral experiments to full scale NPP.

Considering integral test facilities which are simulators of water cooled reactors and qualified computer codes based on advanced models, the method relies on code capability, qualified by application to facilities of increasing scale. Direct data extrapolation from small scale experiments to reactor scale is difficult due to the imperfect scaling criteria adopted in the design of each scaled down facility. The direct code application to different scaled facilities (i.e. without the availability of experimental data) and to the corresponding NPP can be biased or affected by systematic errors. So the only possible solution to ensure the best use of the code in predicting NPP behavior is the extrapolation of accuracy (i.e. the difference between measured and calculated quantities). Experimental and calculated data in differently scaled (relevant) facilities are used to demonstrate that physical phenomena and code predictive capabilities of important phenomena do not change when increasing the dimensions of the facilities. The flow-sheet of UMAE is given in Fig. 3. The following can be added:

- The red line loop on the right of the diagram constitutes the way to qualify the code, the nodalization and the code-user in relation to the capability to model an assigned transient.
- In case the conditions (thresholds of acceptability) in the rhomboidal block 'g' are fulfilled, the NPP nodalization can be built-up having in mind the experience gained in setting-up ITF nodalizations.
- The NPP nodalization (left of the diagram) will undergo a series of qualification steps including the co-called 'Kv-scaled' calculation.
- Additional acceptability thresholds must be met under the block "k". In case of adequate fulfillment of criteria a qualified nodalization is available for NPP analyses (so called Analytical Simulation Model – ASM).
- The FFTBM (Fast Fourier Transform Based Method) to quantify the accuracy, is used at the level of the block "g" and, if requested, of the block "k".

- The results of the ASM may benefit of the extrapolation of the accuracy to characterize the uncertainty.

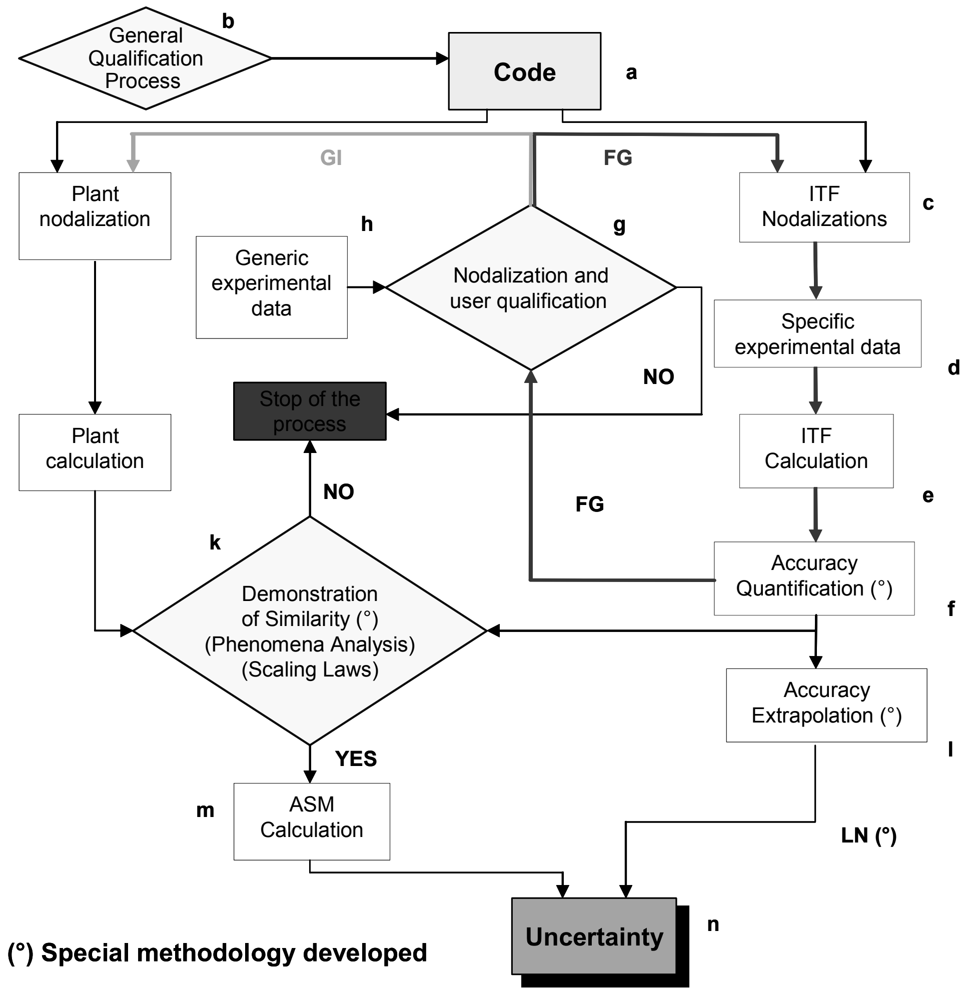

Fig. 3 – The Flow-Diagram of UMAE.

All of the uncertainty evaluation methods, including UMAE are affected by two main limitations:

- The resources needed for their application may be very demanding, ranging to up to several man-years;

- The achieved results may be method/user dependent.

The last item should be considered together with the code-user effect, widely studied in the past as mentioned in ref. [4], and may threaten the usefulness or the practical applicability of the results achieved by an uncertainty method. Therefore, the Internal Assessment of Uncertainty (IAU) was requested as the follow-up of an international conference jointly organized by OECD and U.S. NRC and held in Annapolis in 1996, e.g. see ref. [4]. The CIAU method, ref. [5], has been developed with the objective of eliminating/reducing the above limitations. The basic idea of the CIAU can be summarized in two parts, as per Fig. 4:

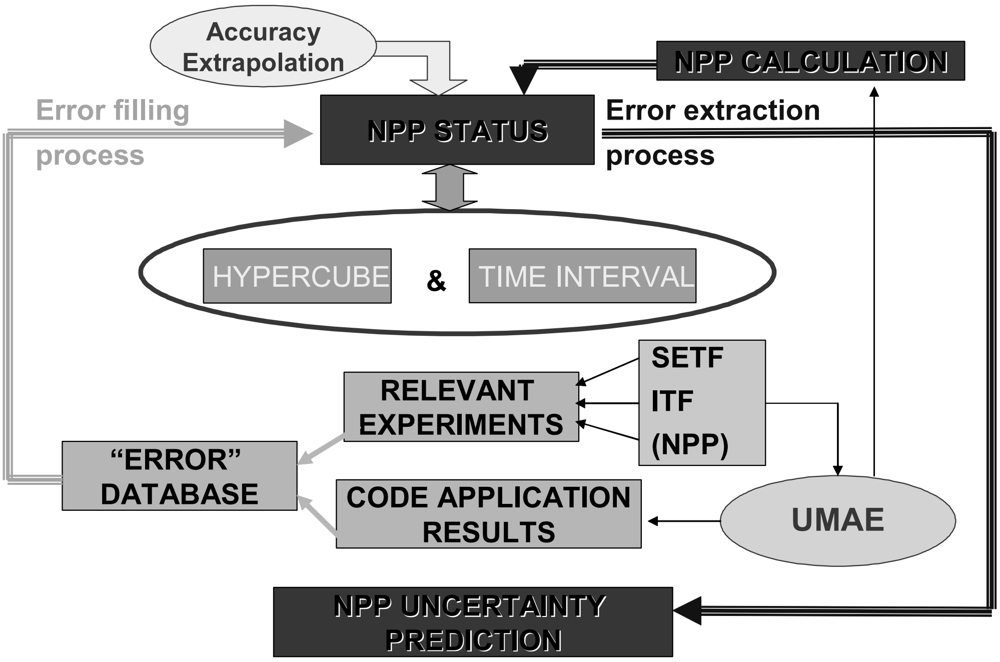

Fig. 4 – Outline of the Basic Idea of the CIAU Method.

- o Consideration of plant status: each status is characterized by the value of six relevant quantities (i.e. a hypercube) and by the value of the time since the transient start.

- o Association of an 'extrapolated error' or uncertainty with each plant status.

Six driving quantities are used to characterize anyn hyoercube. In the case of a PWR the six quantities are: 1) the upper plenum pressure, 2) the primary loop mass inventory, 3) the steam generator pressure, 4) the cladding surface temperature at 2/3 of core active length, 5) the core power, and 6) the steam generator down-comer collapsed liquid level.

A hypercube and a time interval characterize a unique plant status to the aim of uncertainty evaluation. All plant statuses are characterized by a matrix of hypercubes and by a vector of time intervals. Let us define Y as a generic thermal-hydraulic code output plotted versus time. Each point of the curve is affected by a quantity uncertainty (Uq) and by a time uncertainty (Ut). Owing to the uncertainty, each point may take any value within the rectangle identified by the quantity and the time uncertainty. The value of uncertainty, corresponding to each edge of the rectangle, can be defined in probabilistic terms. This satisfies the requirement of a 95% probability level, e.g. acceptable by US NRC.

5. Conclusions

An outline has been given of relevant features of the BEPU approach pursued for the Chapter 15 of the FSAR of Atucha II NPP.

The execution of the overall analysis and the evaluation of results in relation to slightly less than one-hundred PIE revealed the wide safety margins available for the concerned NPP that was designed in the 80's. Key issues for a BEPU-based Chapter 15 of any FSAR are:

a) Proper selection of PIE.

b) Simulation of I&C system response.

c) Availability of proper computational tools.

d) Qualification and quality assurance.

e) Last but not least: endorsement and acceptability by the Licensing Authority.

Acknowledgements

The work leading to the issue of BEPU Chapter 15 of Atucha II FSAR lasted more than two years and involved more than thirty scientists, including recognized international experts, working at NA-SA and at University of Pisa. The current authors coordinated the group and acknowledge the contribution of any individual.

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SESSION TWO

Olkiluoto 3 Construction Experience
Petteri Tiipana (STUK, Finland)

Experiences with Tomari-3 Construction
Jinich Miyaguchi (Mitsubishi Heavy Industries, LTD, Japan)

Lessons Learned from Past and Ongoing Construction Projects
Omid Tabatabai (NRC, USA)

Flamanville 3 EPR, Safety Assessment and On-site Inspections
Corinne Piedagnel, François Tarallo, Bernard Monnot (IRSN, France)

Experience with the Flamanville 3 Construction
Robert Pays (EDF, France)

Extensive Analysis of Worldwide Events Related to the Construction and Commissioning of Nuclear Power Plants: Lessons Learned and Recommendations
Marc Noel (UE/JRC)

Olkiluoto 3 Experience

Petteri Tiippana, STUK, Finland

This paper discusses the experience from the Olkiluoto 3 nuclear power plant project from regulator's point of view. There are certain factors that have affected greatly the project progress. First, Olkiluoto 3 nuclear power plant is the first European Pressurised Reactor (EPR) being constructed. Secondly, construction of the unit started after a fairly long break in nuclear power plant construction in Europe, which had resulted in loss of experienced and qualified engineering and manufacturing resources. These factors have to be kept in mind when evaluating the experience from Olkiluoto 3.

Experience discussed in this paper have to do with the licensing and regulatory oversight process, completion of the design prior to construction, experience and knowhow of the participating organisations, quality management in a nuclear construction project, advanced manufacturing and construction technologies, turnkey contract with regard to licensee's responsibility, safety culture aspects in a nuclear construction project, and the role and importance of regulator's oversight.

1. Regulatory framework in the country of construction

Each nation is solely responsible for the safety of its nuclear installations. Therefore, there are also national practices how nuclear power plants are licensed and how the safety and quality of these plants are verified during construction and operation.

Differences in licensing, regulations and regulatory practices may have an impact on the design of the plant. There may be differences in how the detailed design has to be documented and how and when it needs to be submitted for approval to the regulator. To avoid surprises due to differences, it is beneficial for the owner and plant vendor to familiarize themselves early enough on the national practices and regulations to ensure that regulatory expectations and processes can be taken into account in the project implementation. In addition, the owner and the plant vendor have to understand what are the national safety goals and safety requirements that the plant has to fulfill, and what they mean to the detailed design of the plant. These have to be clarified and explicitly defined by the owner in terms of design criteria in the bidding documentation to avoid difficulties in the future steps of the project. It is recommendable to discuss those design criteria also with the regulator in connection with the bidding process before signing the contract.

In the Finnish regulatory system, the licensee has to submit the design and working documentation[4] of safety significant systems, structures and components to the regulator for approval. Regulator's approval of the documentation for most safety significant structures and pressure equipment is required prior to start of construction or manufacturing. Also the pressure equipment manufacturers, and inspection and testing organisations have to be approved by the regulator prior to start of manufacturing, inspections and tests, respectively. If the information provided on the design or on the manufacturer is not acceptable to the regulator and needs to be improved, there may be consequences to the project progress. In addition, when reviewing the working documentation the regulator may define witness or hold point type of inspections. These inspections are conducted before and during construction or manufacturing to verify that the

[4] For example concreting plans, technical description of pressure vessel manufacturing, welding procedures, inspection plans, etc.

component is being manufactured as described in the documentation. These are generic examples on regulatory approval and inspection processes that are applied in Finland and which have to be taken into account in the Finnish projects.

2. Completion of the design prior construction

One of the most important factors that have affected the progress of Olkiluoto 3 project is the status of detailed design of the plant and its' systems, structures and components at the time when civil construction was to be started in the beginning. In general, it has turned out to be time and resource consuming to incorporate new conceptual features into the detailed design of the plant. In Olkiluoto 3, these new safety and design features had to do, among other things, with provisions against large passenger airliner crash and systems and structures needed to cope with severe accidents.

The amount of work needed to complete the detailed design is enormous and the design work may involve both in-house (vendor) and subcontracted engineering staff. Furthermore, design of a first of a kind plant may be much more iterative process than redesigning a plant that has already been built.[5] In this context, the management of the design process becomes very important. It is beneficial to all stakeholders to ensure the availability of qualified and experienced engineering resources and mature design management processes before start of the detailed design. This is needed to ensure a once through review and approval of the design and working documentation, i.e. to avoid rotation of documents between the involved parties.

Depending on the regulatory review and approval process, iteration of the design and design documentation may result in several rounds of regulatory review. This may overload the vendor, licensee and regulatory organisations. From safety and regulatory perspective it is important that the licensee and vendor are able to show when safety significant issues are going to be fixed and when regulatory approvals are expected. These points could be presented in a licensing schedule indicating all safety relevant points where regulator's review and approval is needed.

If the design is not complete enough prior to construction, it risks the timely start and continuous progress of construction. This may result in construction and manufacturing delays, difficulties in contracting and managing subcontractors, challenges in the design configuration management, redesign and rework on site due to a need to change already completed civil works.

3. Management of design in a construction project

As written in the previous paragraph, subcontractors may do significant parts of the detailed design of the plant's systems, structures and components.[6] In addition, the nuclear industry is widely globalised and also the vendors may be multinational. This means that the detailed design of the plant can be done by several organisations and in different locations.

Both the use of subcontractors and the global nature of vendor organisations highlight the importance of proper design management processes. This includes written a description of design configuration and change management processes, together with a transparent and traceable requirement management. In principle, it should be possible for an outsider to be able to follow how plant level design requirements are transferred and communicated to the system level and from there to the design of structures and

[5] Eventhough the plant has been built somewhere else, there is anyhow some redesign that needs to be done due to site specific aspects, due to different in subcontractors, and due to different owner and regulatory requirements in a country of construction.

[6] This may depend on the vendor and project type

components. Design management process should ensure adequate communication between civil, process, electrical and I&C engineers. One should also ensure that safety and risk experts are directly involved in the vendor's design review process. They should verify that for example the principles of redundancy, separation, and diversity are consistently applied in all disciplines including both frontline and support systems.

One important factor for avoiding misunderstandings and for managing the interfaces between the design levels and the different organisations that work on the plant, system and component level designs is provision of explicit design and implementation requirements.[7] Explicit requirements will also ensure that the design of structures and components meet the requirements set by the system and the plant level. It is not possible for the licensee and regulator to approve the documentation if it is not explicit. This causes additional updates of the documentation and extra work for all organisations.

4. Experience and knowhow of the licensee and the vendor, management of subcontractors

Licensee's and vendor's key persons (e.g. project directors, people responsible for safety and design of the plant, Quality Assurance and Quality Control) shall have experience in nuclear power construction or operation. This is the key to the success. Right experience ensures that nuclear specific issues are known and timely identified and right amount of attention, resources and time is allocated to the important areas.

Regulator should verify that the licensee has adequate human resources for the project from the beginning. Resources should include e.g. staff needed to oversee the activities of the vendor and staff needed to review and approve of the plant's design. Licensee's need of resources varies during the project and hence it should conduct staff planning covering the entire project. In the plans it should be able to show the number of staff and their qualifications needed in different phases of the project as well as where licensee can find the resources. Staff planning is important especially in a small country if more than one nuclear power project is starting or ongoing. The use of consultants in lieu of licensee's own staff should be planned and justified. When balancing the use of consultants and own staff it should be taken into account that the design and construction phases are most useful for the licensee to build up the know-how and experience which is needed to operate the plant safely.

Depending on plant vendor's in-house capabilities the amount of subcontracted work varies. If the vendor does not have manufacturing or construction capabilities in its own organisation, the amount of contracted work is significant. This type of situation highlights the importance of subcontractor management. The subcontractor selection criteria and approval process should be clearly defined and agreed between the vendor and the licensee. Licensee may want to consider setting restrictions to the length of the subcontractor chain (e.g. vendor's subcontractor may not order work from another subcontractor unless agreed prior contracting). It should be understood by the licensee and the vendor that subcontractors with limited or no nuclear experience require special attention with regards to training and guidance prior to start of activities and oversight during manufacturing and construction.[8] In addition, contracts between the vendor and its subcontractors shall be clear especially in nuclear specific issues (e.g. quality assurance and quality control requirements differing from conventional industry). The understanding and implementation of these requirements shall be audited prior to start of activities and verified when the activities are ongoing.

Olkiluoto 3 project faced a challenging situation in the beginning of the project being the first nuclear construction project in Europe since several years. Idle period in Europe had led to a situation where many

[7] Design and working documentation shall not contain implicit expressions such as "mainly", "in principle", "in general", "and/or" "whenever possible" etc.

[8] This may also apply to experienced manufacturers if they have not had nuclear specific work for some time.

of the experienced manufacturers had left the sector due to lack of business. This meant that the vendor had to educate many new subcontractors to be able to work for Olkiluoto 3 project. Education has taken a lot of effort from the vendor and in some cases first components or structures produced by a new manufacturer have not met the criteria. It has turned out to be a very difficult task to verify the real know how, experience and preparedness of the subcontractors on the shop floor prior to start of activities and this has not always succeeded.

5. Role of quality management

Requirements for Quality Assurance (QA) and Control (QC) are very specific to safety critical applications like nuclear power plants. In general, the conventional industry is nowadays quite familiar with the international quality standards like ISO. It has to be noted that ISO and other conventional standards alone are not enough for safety significant activities in a nuclear power project. What is enough shall be clarified, defined and agreed between the regulator, the licensee and the vendor in the very first days of the project. Requirements for the quality assurance and quality control shall be commonly understood throughout the project (e.g. regulator, licensee, vendor and its subcontractors). This means for example that quality requirements to be applied in different safety classes are clearly defined and agreed before subcontracting and procurement starts so that they can be clearly written in contracts and specifications.

One specific example is the definition and process for a non conformance. The criteria for a non conforming performance or a product shall be clear. Raising and reporting a non conformance shall follow a uniform process throughout the project. Criteria to classify a non conformance to minor, significant or critical shall be well defined, Roles and responsibilities to report and resolve a non conformance shall be clear. The process has to be effective (e.g. closure of a non conformance should not be too time and resource consuming). In Olkiluoto 3 project, the utilisation of non conformance process tends to stop when the direct cause for the non conformance has been found and corrective actions to repair the problem has been completed. Root causes are not always studied thoroughly which sometimes results in repetitive non conformances.

Management at different levels has to be educated to understand the role and significance of QA and QC in a nuclear construction project. Understanding will lead to management's commitment to high quality. Commitment comes visible when management uses the information and the system as a management tool (e.g. grouping, categorisation and analyses of non conformances leading to activities to minimize non conformances in the future).

6. New and advanced manufacturing technology

As well as new design features, also new and advanced manufacturing and construction technologies may require additional time and effort to be qualified for the purpose. Areas where new technologies will be applied should be identified in the beginning of the project to be able to pay attention and allocate adequate time for the qualification process. Experience in Olkiluoto 3 project has shown that in some cases the qualification pieces have not met the specifications at the first attempt.

7. Licensee's responsibility in turn key contract

From regulator's point of view, licensee is always responsible for safety independently of the contract type. Licensee has to control and oversee everything that has to do with safety of the plant. With a turn key contract the licensee has contracted a vendor for example to design, build and commission the nuclear power plant. In practice it also means that the vendor is responsible for the design, construction and commissioning of the plant. Vendor again may shift the responsibility to its subcontractors depending on the area subcontractors are responsible for.

Turnkey contract type highlights the importance of clear and explicit requirements for the design, manufacturing, construction, installation and commissioning of the plant. Olkiluoto 3 project has shown that it is not always simple for the licensee to interfere the project when the work is in progress and requirements for the work are not explicitly defined. It can be simpler to wait and interfere afterwards and prove that the component or structure does not meet the criteria (even though it may have been obvious when the work was in progress).

Even turn key projects are manageable, but in addition to explicit requirements a necessary prerequisite is systematic, transparent and traceable requirement management process applied together with strong, competent and safety oriented QA/QC personnel who are able to verify compliance with the requirements.

8. Safety culture in a construction project

Construction of a nuclear power plant does not differ from an operating nuclear power plant from safety culture point of view. Safety and quality must have higher priority than costs and schedule. This message has to be very clear and transmitted from the licensee and vendor management to all participating organisations and to all levels of the organisations. Management's acts and decisions in the project have to be consistent with the message.

In order to ensure that safety and quality has the highest priority in every day activities throughout the project, everyone has to understand the safety significance of the work one is responsible for. Understanding is essential to promote personal responsibility for safety and quality. This is a challenge in a construction project where thousands of people are involved and many of them have no previous experience or knowledge on nuclear power plants.

The level of safety culture is tested whenever problem situations are encountered. One of the outcomes of good safety culture is that safety and quality problems are openly raised, discussed and reported. The ways and routes to raise them have to be made known to the workers. Atmosphere to report safety and quality issues has to be open, free of punishment and encouraged by the foremen. When a safety issue has been raised by a worker, foreman or someone else has to give feedback to the person and inform whether the issue was significant or not and how it was resolved. Otherwise the person may feel that the organisation is negligent to safety concerns or even wants to hide them. The importance of encouraging workers to raise safety and quality issues as well as the importance of giving feedback to workers has to be addressed in the training of foremen. At the Olkiluoto 3 site a number of different nationalities are working together. Therefore also differences in cultures and languages had to be addressed when reporting routes for workers were established.

It is management's task to create a good safety culture in the construction project. However, the role of foremen working with workers is very significant. Foremen have to be able to manage workers from different nationalities and cultures, encourage workers to report safety and quality issues, give feedback to workers, and especially to promote personal responsibility by making them understand the safety significance of their work. All this sets special requirements on foremen's selection, training and personnel management skills.

9. Regulatory issues in new construction

Olkiluoto 3 experience has shown the importance of stringent regulatory approach and inspections. These are needed to verify that the performance of the licensee, the vendor and the subcontractors meet the expectations and that the equipment and structures meet the specifications set by the design. There are some cases where Quality Control inspectors of manufacturer, vendor, and licensee may have not been strong enough to enforce stopping of work and making necessary timely corrections. These may have something to do with a turnkey contract together with cost and schedule pressure caused by a stop of work. In such situation, an intervention by a regulatory inspector has been needed. Prerequisite is that regulatory body is competent, independent, has strong powers and enforcement tools.

Olkiluoto 3 project has raised a lot of media interest in Finland as well as internationally. The amount of interest has been remarkable and resource consuming. Therefore it is important that regulator is prepared to interact with media. One of the essential factors is that regulatory decisions are transparent and can be published either pro-actively or when asked by media, other organisations or members of the public.

10. Conclusions

Starting new build is very demanding because much of the earlier experience and resources have been lost from the nuclear industry. Therefore, adequate time has to be allocated to good preparation of the project before the actual construction start. This includes e.g. building competent organizations, complete design incorporating possible new design features, qualified new manufacturing and construction technologies, ensuring availability of qualified designers, constructors and manufacturers to implement the project, and resolving potential regulatory uncertainties.

Construction of a nuclear power plant is a complex project. It for example includes different technical disciplines within various stakeholders, a number of subcontractors, and workers from different countries. It is evident that challenges will be met in a nuclear power plant construction project. Therefore close monitoring and oversight both by the licensee and the regulatory body is necessary to ensure achievement of specified quality, i.e. meeting the technical standards and criteria that the vendor has specified and that have been approved as part of licensing and design documents.

The know-how and experience of all stakeholders has increased in Olkiluoto 3 project. Progress has been made during the project, and after the "teething problems" the civil construction has proceeded well. However, it seems to be characteristic that the beginnings of new phases in the project (concreting activities, welding activities, installation activities, commissioning activities etc.) require special attention and preparations from all stakeholders.

While there have been many non-conformances and re-manufacturing needs, the corrective actions have been taken in line with the Quality Assurance and Quality Control practices specified for the project. The final quality in Olkiluoto 3 structures and components has not been compromised although in some cases it has required special efforts to achieve and demonstrate the expected quality. These have included extensive and time consuming tests, inspections and extensive new analysis to prove that the required standards have been met. In some cases components or structures had to be re-manufactured or constructed. The observed difficulties at the construction stage have not raised concerns on the safety of the power plant when it will be ready to operate.

Lessons Learned from Past and Ongoing Construction Projects

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The nuclear industry in the U.S. faced many construction quality and design issues in the 1970s and 1980s. In 1984, the NRC issued NUREG-1055, "Improving Quality and the Assurance of Quality in the Design and Construction of Nuclear Power Plants," to document the lessons learned from nuclear power plant (NPP) construction in the U.S. In recent years, several countries have begun either planning for or actually constructing new NPPs. For instance, in the U.S., the nuclear industry has submitted several combined license and design certification applications to the NRC for licensing reviews and approval to build 30+ new NPP units.

Latest construction experience from countries that are currently building new NPPs indicate that these countries are dealing with challenges that are similar to those issues that caused major quality assurance problems, delays, or even termination of several projects in U.S. in the 70's and 80's. The U.S. NRC is proactively taking measures to improve its regulatory programs as well as construction oversight activities before new NPPs construction begin in the U.S.

In late 2007, the U.S. NRC's Office of New Reactors established a construction experience program (ConE) to obtain and evaluate construction and operating experience events and to identify the lessons learned from these events. In March 2009, the NRC published an Office Instruction to provide a process for incorporating the lessons learned and insights from the design, construction, and operation of the international and domestic NPPs into the licensing reviews, inspections, and construction of new reactors in the U.S. Additionally, the ConE program staff developed a Web-enabled database to store, manage, and make construction experience information available to all NRC technical reviewers as well as inspectors. Because this database contains information from other countries' regulators that are considered non-public, the NRC has restricted the use of this database to the NRC staff only. However; through issuing multiple generic communications, the NRC staff has been sharing these insights and lessons learned publically with all domestic stakeholders and international partners.

This presentation will describe the NRC process for obtaining, screening, evaluating and incorporating the ConE insights into the NRC oversight and regulatory programs, and will also provide several examples of generic lessons learned from the evaluation of significant construction and operating experience events.

Flamanville 3 EPR, Safety Assessment and On-site Inspections

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As a Technical Support Organisation of the French Safety Authority (ASN), the IRSN carries out the safety assessment of EPR project design and participates in the ASN inspections performed at the construction site and in factories.

The design assessment consists in defining the safety functions which should be ensured by civil structures, evaluating the EPR Technical Code for Civil works (ETC-C) in which EdF has defined design criteria and construction rules, and carrying out a detailed assessment of a selection of safety-related structures. Those detailed assessments do not consist of a technical control but of an analysis whose objectives are to ensure that design and demonstrations are robust, in accordance with safety and regulatory rules.

Most assessments led IRSN to ask EdF to provide additional justification sometimes involving significant modifications. In the light of those complementary justifications and modifications, IRSN concluded that assessments carried out on design studies were globally satisfactory.

The participation of IRSN to the on-site inspections led by ASN is a part of the global control of the compliance of the reactor with its safety objectives. For that purpose IRSN has defined a methodology and an inspection program intended to ASN: based on safety functions associated with civil works (confinement and resistance to aggressions), the corresponding behaviour requirements are identified and linked to a list of main civil works elements.

During the inspections, deviations to the project's technical specifications or to the rules of the art were pointed out by IRSN. Those deviations cover various items, such as concrete fabrication, concrete pouring methodology, lack of reinforcement in some structures, unadapted welding procedures of the containment leaktight steel liner and unsatisfactory treatment of concreting joints. The analysis of those problems has revealed flaws in the organisation of the contractors teams together with an unsatisfactory control of the contractor's activities by EDF. However, some three years after the beginning of construction, the organization and strictness of the main civil contractor and of EDF construction team have improved.

1. Introduction

More than ten years after the latest construction in France, the Flamanville 3 site has opened, and the first French EPR type unit is due to diverge in 2012.

The construction license Decree was signed by the French Government in April 2007 and prior to the Operation Licence Application to be issued by EDF in 2011, an early assessment of regulatory documents and a control of the reactor construction are under way.

As a Technical Support Organisation of the French Safety Authority (ASN), the IRSN carries out the safety assessment of EPR project design and participates in the ASN inspections performed at the construction site and in factories.

Civil structures are the first elements of the unit to be built. The assessment of the civil engineering design of Flamanville 3 began in 2005, two years before the associated on-site inspections. According to IRSN, the latter are part of the overall assessment of the ability of civil works to meet their safety functions and requirements. In other words, the design assumptions and features (mechanical principles of structures, practical arrangements, material properties, provisions regarding durability, etc.) are confirmed only if they are correctly implemented and maintained in the plant.

The IRSN experience related to the safety assessment of the design and to on-site inspections is briefly described below.

2. Civil design safety assessment

The first step in design assessment consists in defining the safety functions which should be ensured by civil structures. Two kinds of safety functions are involved:

- nuclear safety functions such as radioactive material containment and radiation protection;
- functions directly related to a nuclear safety function like protection against internal and external hazards and supports of safety-related equipment.

External hazards can be earthquakes, aircraft crashes, external explosions, extreme temperatures, external flooding, etc.

Internal hazards are failures of components, internal flooding, fire, internal explosions, load drops, etc.

Considering these safety functions, safety requirements are defined as follows:

- leak-tightness and retention in order to meet release criteria;
- resistance, stability and supporting capacity;
- choice of materials and determination of their biologic thickness.

To meet the safety requirements, different behaviour requirements are defined for each structure or part of structure. For example, the EPR reactor building is made up of a double-wall concrete containment and a metallic liner, which is designed to guarantee leak-tightness in case of radioactive material release. The inner containment must withstand internal pressure due to high energy pipe breaks; the outer containment must withstand external hazards, particularly an aircraft crash, and contain the possible leakage from the inner wall, the steel liner included.

Once the behaviour requirements identified, the design of each structure is carried out according to Basic Safety Rules (RFS) which are ASN guides and EPR Technical Code for Civil works (ETC-C) in which EdF has defined design criteria and construction rules and has given a definition of leak and resistance tests related to the reactor containment.

First, IRSN assessment consists in evaluating the ETC-C which has been drafted by EdF, principally in accordance with French regulatory rules with some adaptations to the particular EPR project.
The rest of the assessment described below is based on the ETC-C code.

The IRSN evaluates the global model and calculation of the nuclear island carried out by EdF. For example, assumptions, input data and load cases introduced in the model are checked according to behaviour requirements. The global results which are used for the basemat and structures design are analysed in terms of consistency with assumptions and existence of margins.

Then, a detailed assessment of a selection of safety-related structures is carried out:

- nuclear island basemat;
- reactor building: steel liner, inner prestressed containment, outer containment (airplane shell), internal structures, pool;
- fuel building: internal structures, pool;
- safeguard auxiliary building;
- pumping station.

At this step, the IRSN analyses detailed calculation notes and drawings so as to ensure that results are consistent with the design methodology, the criteria and construction rules defined in ETC-C and also with "Art Rules" and the Book of Technical Specifications (RST).

Detailed safety assessments do not consist of a technical control but of an analysis whose objectives are to ensure that design and demonstrations are robust, in accordance with safety and regulatory rules.

Most assessments led IRSN to ask EdF to provide additional justification sometimes involving significant modifications. In the light of those complementary justifications and modifications, IRSN concluded that assessments carried out on design studies were globally satisfactory.

Besides, during its assessments, IRSN identified items whose construction should be inspected, because of their importance for safety, or because of special execution difficulties encountered in the past during the construction of NPPs.

3. On-site inspections: objectives and general overview

3.1. Objectives and general overview

As part of the global control of the compliance of the reactor with its safety objectives, a control of the construction is carried out by ASN and IRSN, through a limited number of inspections. Since the future Operator –EDF- is, according to the French law, sole responsible for the safety of the plant (as far as civil works are concerned), the main idea is to verify that EDF endorses its responsibility and masters the quality of the construction, and its compliance with safety objectives and technical regulations.

Thus, the general objectives of the inspections are the following:

- checking that the safety requirements established in the approved design are still met during the building activities, in particular that the designer's drawings and specifications are correctly applied by the civil contractors;

- verifying that good practice during construction is respected, resulting in a fair level of quality of the buildings;

- assessing the management and survey of its site by EDF, who is the final caretaker of the construction's quality, and the future operator of the plant.

IRSN has defined a methodology and an inspection program intended to ASN. Based on safety functions associated with civil works (confinement and resistance to aggressions), the corresponding behaviour requirements are identified and linked to a list of main civil works elements. Those requirements cover stability, no perforation, leak-tightness, fluid retention, protection against climatic conditions, together with the durability for the lifetime of the plant. Finally, a list of sensitive or exemplary elements is chosen as targets for inspections, that are programmed according to the construction schedule. That selection incorporates the experience gained from the construction and operation (incidents and maintenance) of the existing plants and from the Olkiluoto 3 site. The priority order affected to each element depends on its importance for the safety of the reactor.

The first inspections were carried out in 2007, dealing with the preliminary works: foundation rock excavations by blasting, underground galleries and risks caused by the construction activities to the safety of the two nearby units under operation. The concrete works developed during 2008 and the first structural concrete of the nuclear island was poured in December. Foundation works finished in early 2009 for the nuclear island and the pumping station, giving way to the construction of the first levels of the buildings and to the erection of the bottom part of the reactor containment.

IRSN systematically takes part in the inspections and identified non-conformities as well as bad practices. When necessary, IRSN sends warning letters to ASN, eventually followed by an action of EDF and a higher construction quality level.

A few technical problems highlighted during the inspections are described below.

3.2. Water excess in structural concrete

After two inspections held in October 2007, the IRSN stated in its November 2007 technical assessment that the water/cement ratio (0.50) of the structural concrete delivered by the civil contractor could be too high to meet the objectives of durability of the project, in marine atmosphere. The possible phenomena are higher porosity of concrete and additional cracking due to excessive shrinkage, causing a poor protection of steel reinforcement. Later on, the formulation of concrete was changed, to reach a better ratio (0.45).

3.3 Cracks in the concrete of the reactor building basemat

In early December 2007 the first lift of the reactor building basemat was poured, as first structural concrete of the nuclear island. This lift is a 1.8 m thick, 55 m diameter disk, leading to a concrete volume equal to 4225 m^3. Two days after the 40 hours long concreting, cracks on the whole surface of the lift were observed. Their initial openings, 1 to 3 mm, decreased after cooling to more limited values: 0.4 to 1 mm. The FA3 civil works contract limits cracks openings smaller than 0.2 mm. Those cracks were eventually grouted by the contractor.

The basic cause of this non-conformity is the thermal effect (expansion and gradient) due to the heat of hydration of the cement during concrete setting. Besides, this lift was concreted without any reinforcement mesh in its upper part. Such a mesh distributes the possible contraction strains, thus leading to elementary cracks of reduced openings.

IRSN thinks that the main risk linked to such a defect is a reduced durability of the structure, because of possible corrosion of the bottom reinforcement, and that the repair by grouting cannot give thorough guarantee on the protection of those rebars. Then, IRSN suggests that the presence of water below the basemat should be detected by EDF, during the lifetime of the plant.

In its November 2007 technical assessment, IRSN had pointed out the risks associated with the execution of large concrete blocks, namely thermal cracking and Delayed Etringite Formation, and asked a special care on temperature limitation inside those blocks during their execution. Moreover, IRSN states that the repair of abnormally cracked structural elements cannot be considered as a current method of execution of nuclear buildings.

3.4. Lack of reinforcement in the nuclear island basemat

During the inspection on 5 March 2008, the IRSN representatives noticed a local lack of transversal rebars in concrete block number 2 of the fuel building basemat, while the concreting of this block was under progress. This problem had not been identified by the internal control of the contractor, or by the EDF control. The concreting work was rapidly stopped, and resumed only after the reinforcement was completed, according to the execution drawings. Among others, this finding clearly showed that the preparation of its tasks by the civil contractor and the control by EDF were not satisfactory. After that inspection, EDF undertook corrective actions in order to improve its control and the quality of the work of the contractor.

3.5. Welding process of the containment steel liner

The metallic liner of the reactor building ensures the leak-tightness of the containement in the event of a possible serious failure on components which contain primary coolant water. In the event of such an accident, this 6 mm thick metallic skin on the inner face of the containment concrete structure constitutes the ultimate static barrier for radioactive products. This safety function must be provided throughout the lifetime of the power plant. A first checking of this safety function is made with the pressure air test performed at the end of the construction of the reactor building. This test is however a global test during which a leak cannot be easily located, so that a repair in order to restore leak-tightness will remain very problematic.

Consequently, guaranteeing liner leak-tightness, making it possible to consider a successfull pressure test result of the reactor building, mainly relies on the provisions taken before test, on both the level of the quality of liner design and the liner manufacture quality.

The liner is mainly made with thin P265GH steel sheets, and design and manufacturing are subject to technical requirements defined in the contractual specification related to civil works (RST 2.01). This specification requires in particular preliminary qualifications for welding procedures and welders in accordance with European standards, a welding in several layers of welded metal, and radiographic test of butt welds with a random sampling check on 10% welded lengths.

After an interruption of almost 20 years in the construction of PWR containment steel liners in France, IRSN naturally paid detailed attention to the first welding activity carried out on site on an element endorsing a safety function: the liner manufacturing.

IRSN took part in several inspections with ASN from the beginning of the liner welding operations on the Flamanville site. These inspections resulted in detecting deviations from technical requirements of RST2.01 specification, in particular on the welding procedure. These variations lead IRSN to recommend complementary examination tests, and a 100% non destructive vacuum box test on welds was defined. The following inspections revealed perfectible conditions of welding needing implementation of climatic conditions protection, and some non-conformities in documentations. At the same time, the assessments of the first random sampling x-ray inspection campaigns showed abnormally high rates of repairs for an easily weldable steel. These reports, signs of a welding activity not completely controlled in spite of the required qualifications, led IRSN to recommend a 100% volumetric non destructive test of welds until return to a normal situation. In front of these difficulties, the manufacturer defined and applied an action plan aimed to significantly improve the quality of works by optimizing the welding procedures, by the improvement of their conditions of implementation, and by complementary training sessions and selection of welders. After a few weeks of application of this action plan, the results of control on welds already indicate a clear improvement, and a return to a normal situation.

The difficulties faced on the Flamanville site confirm that welding activity on equipment providing a safety function, after a long interruption or by a manufacturer not familiar with the specific environment of nuclear energy, requires an approach going beyond the simple compliance with the technical provisions defined in contractual specifications. Taking into account the safety function of the component concerned, awareness of the importance of this safety function by all those involved in the manufacture appears necessary, first to a good perception of the required quality level, and second to reach quality level consistent with this safety function.

3.6. Unsatisfactory treatment of concreting joints

Concerning concreting joints, unsatisfactory treatments were pointed out during several inspections, for instance in November 2008 (no treatment at level -6.35 m in the gusset of the reactor building containment wall), July 2009 (deviation from the technical specifications, in the contractor's execution procedure) and August 2009 (inappropriate use of deactivator).

The project's technical specifications state the contractor should roughen the concrete joints using a water and air jet, or should present alternative methods to the approval of EDF. The contractor has presented alternative methods: use of deactivators and jackhammer. EDF approved those modifications.

IRSN recalls that the state of the art advises against those alternative methods, that generally lead to lower quality joints. Such defects may jeopardize the robustness of structures, that could be lower than expected in design, and reduce their durability because of faster than expected steel reinforcement corrosion.

Technical discussions with EDF are under way. As first steps, EDF has stopped the use of one deactivator, whose use for the treatment of concreting joints is not planned by its manufacturer, has undertaken a dedicated test program, and has strengthened its survey of the contractor's activities in that field.

3.7. Difficulties in anchor plates placing

During two inspections held in March and June 2009, a new problem was pointed out by IRSN at the interface between civil works and mechanical components. Insufficient strictness was noticed on placing and location control of anchor plates: a topographical survey was carried out after placing in the reinforcement but not after concreting, although plates can move from their initial place during concreting. Besides, the civil contractor frequently shifts the plates from their theoretical positions because of conflicts between the plates anchoring and the dense steel reinforcement of concrete structures. So, location deviations were higher than tolerances existing in ETC-C. Then, EDF released those tolerances but, after the inspections, on request of IRSN, EDF undertook corrective actions to improve the anchor plates placing. Furthermore, EDF decided to carry out a topographical survey for all mechanical plates, just after concreting, in order to identify and to communicate to the equipment designers, as soon as possible, important deviations which could modify arrangement and installation drawings.

3.8. Unsatisfactory location of prestressing ducts

In November 2009, EDF informed the inspectors that several horizontal prestressing ducts of the inner containment showed significant deviations from their specified locations, in the first concrete layer. Moreover, IRSN pointed out that the final control of two ducts had been only partially carried out. Those déviations, higher than tolerances, can jeopardize the inner containment resistance and its capacity to ensure the safety function required.

EDF issued a specific demonstration indicating those deviations were acceptable even so resistance margins are cut off. For the next concrete layers, corrective actions will be undertaken to obtain deviations lower than stated tolerances.

4. Conclusion

The IRSN safety assessment of the civil works detailed design confirms the consistency of that design with the safety objectives associated with the EPR project, and highlights items whose construction activity should be inspected.

During the inspections, deviations to the project's technical specifications or to the rules of the art were pointed out by IRSN. The analysis of those problems has revealed flaws in the organisation of the contractors teams together with an unsatisfactory control of the contractor's activities by EDF, even for the safety-related activities. Those findings, after endorsement by ASN, led EDF to carry out corrective actions: for example repair of cracks in the reactor building basemat, improvement of concrete and steel liner fabrication, more safety culture in the contractor's staff, strengthening of the supervisors team. Of special interest is the setting up by EDF of a civil design liaison team on the site, whose duty is a better coordination between design teams and construction activities.

The Flamanville EPR site is one example of the restarting of the nuclear civil construction. Some three years after the beginning of construction, the organization and strictness of the main civil contractor and of EDF construction team have improved.

Extensive Analysis of Worldwide Events Related to The Construction and Commissioning of Nuclear Power Plants: Lessons Learned and Recommendations

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1. Introduction

Lessons learnt from past experience are extensively used to improve the safety of nuclear power plants (NPPs) worldwide. Although the process of analyzing *operational* experience is now widespread and well developed, the need for establishment of a similar process for *construction* experience was highlighted by several countries embarking on construction of new NPPs and in some international forums including the Working Group on the Regulation of New Reactors (WGRNR) of the OECD-NEA.

In 2008, EU Member State Safety Authorities participating to the EU Clearinghouse on Operational Experience Feedback decided to launch a topical study on events related to pre-operational stages of NPPs. The aim of this topical study is to reduce the recurrence of events related to the construction, the initial component manufacturing and the commissioning of NPPs, by identifying the main recurring and safety significant issues.

For this study, 1090 IRS event reports, 857 US Licensee Event Reports (LERs [1]) and approximately 100 WGRNR reports have been preselected based on key word searches and screened. The screening period starts from the beginning of the databases operation (in the 1980s as far as IRS and LER database are concerned) and ends in November 2009. After this initial screening, a total of 582 reports have been found applicable (247 IRS reports, 309 LERs and 26 WGRNR reports).

Events considered for this study were those which have been initiated before the start of commercial operation, and detected before or even long after commercial operation.

The events have been classified into 3 main categories (construction, manufacturing and commissioning), and into further sub-categories (building structures, metallic liners, electrical components, anchors, I&C, penetrations and building seals, emergency diesel generators, pipes, valves, welds, pumps, etc.) in order to facilitate the detailed analysis with the final objective to formulate both equipment specific recommendations and transversal recommendations.

2. Main Trends

A trend analysis has been performed on the basis of the IRS applicable reports. The main conclusions of this analysis are the following.

- The items leading to the highest number of events related to pre-operational stages are I&C, electrical components and welding.

- For events related to the commissioning stages, the study shows clearly that I&C is the prevailing contributor (43% of all commissioning related events).

- The average detection time of the issues is about 8 years after the start of commercial operation. For some components, the detection time can be up to 40 years (in some case, the anomaly was even found out during the decommissioning of the plant). More specifically, the failures of active components like diesels (2.6 years) or pumps (4.5 years) are detected more quickly than the failures of passive components like civil structure (8.9 years), pipes (11.7 years) or welds (12.7 years) because these latest are generally affected by slow degradation phenomena initiated by an initial construction or manufacturing defect, and because of major differences in the surveillance programmes.

- The proportion of events with common cause failure is rather high (1 event out of 3 in average) and can be very high for some materials: more than 50% for civil work and fire protection, 45% for electrical components.

- More than 75% of the events are found out fortuitously (following failure, spurious actuation, fire, flooding or "by chance" through unrelated tests or inspections) and the rate of fortuitous detection is particularly high for some items: civil work, electrical components, I&C, pipes and valves.

- Moreover, some families like building structures, anchors, supports or electrical components accumulates a long detection time, a high rate of common cause failure and a high rate of fortuitous detection (see also [2]).

These trends emphasize the necessity to reduce as much as possible during construction, manufacturing and commissioning of a new reactor the number of deficiencies which can be major latent failures remaining undetected during long periods and can have actual consequences on safety, when the reactors starts to operate.

This emphasizes as well the necessity to detect the deficiencies at the construction stage, as it may be difficult to identify them during operation.

3. Lessons learnt and recommendations

The in-depth analysis performed has firstly allowed identifying detailed and concrete lessons learned for 20 categories of items. Due to size restriction, the details of the recommendations and lessons learnt are not given in this paper but can be found in the topical study [6].

In addition to these component / activity specific lessons learned, general recommendations have been identified. These recommendations consist in transversal recommendations (applicable for most of the activities) and general recommendations for each of the 3 stages of the pre-operational activities (construction, component manufacture and commissioning).

All these recommendations grounded on construction experience (i.e. event reports) can be usefully combined with other existing documents [3][4][5].

3.1. Transversal recommendations

Most of the transversal recommendations are related to the management of the construction project or to the management of the quality. Recommendations are focused on the following topics:

- Safety culture - top priority from the beginning of the project, for all staff and activities.
- Communication between the different companies and entities involved in the project.
- Management of non-conformances – clear definition, roles and responsibilities.
- Specific requirements for nuclear components – clear identification in supply contracts.
- Management of tasks interfaces – clear responsibilities.
- Third party quality control – responsibility distribution, independence, competence.
- Design change management – assessment process and documentation.
- Management of temporary devices – proper documentation and inventory.
- Cleanliness – coverage by QA&QC, cleaning should be followed by commissioning tests.
- Handling, packaging, transportation and storage – coverage by QA programme.

- Foreign Material Exclusion – applicable from the beginning of the project.

- Handling and lifting devices – included in QC & commissioning programmes, before load lifting.

3.1.1. Construction

For the construction phase, recommendations have been raised on the following topics:

- Welding – importance of the QC programme, staff training and experience.

- Labelling – as soon as possible when components are installed.

- Torque of screwed assemblies – need for independent verification.

- Wiring – wiring check.

- Fire protection – fire risk in stages where fire protection not yet fully installed.

- Impact on nearby NPP units in operation – risk analysis, pollutions.

3.1.2. Component supply

Two recommendations have been identified for component supply, related to:

- Purchasing of commercial grade components for safety related equipment by the equipment manufacturer.

- Proven mastership of technologies and manufacturing techniques - use of proven state-of-the art technologies, role of the regulator.

3.2. Commissioning

For the commissioning programme, recommendations have been raised on the following topics:
Scope of the tests – representativity of the test conditions, test comprehensiveness, fragmented tests, non-actuation tests, commissioning of different units (exhaustivity of each individual test programme), simultaneous tests (influence of tests on each other).

- Tests documentation

- Tests acceptance criteria – performance evaluation.

- Check of system reconfiguration after commissioning tests
- Timing of test performance - components in standby.

4. Conclusions

Based on different sources of information, a sound analysis of events connected to preoperational stages of nuclear power plants, detected before or after start of commercial operation, has been performed.

247 IRS event reports, 26 WGRNR reports and 309 US LERs were analysed and distributed into sub-families. A trend analysis of the IRS event reports was performed and led to the following conclusions:

The analysis performed shows that anomalies originating from the preoperational stages of NPPs are detected in average long after the start of commercial operation (latent failures). Moreover, the proportion of events leading to common cause failures is rather high and exceeds 50% for some families. Finally, more than 75% of the events are found out fortuitously.

These trends emphasize clearly the necessity to reduce as much as possible during construction, manufacturing and commissioning of a new reactor the number of deficiencies which can lead to major undetected latent failures and can have actual consequences on safety, when the reactors starts to operate.

Furthermore, the qualitative study of these events has allowed to identify specific lessons learnt and to raise transversal recommendations. Many of these transversal recommendations are related to the management of the project (responsibilities, communication, safety culture, third party management, etc.) and in particular to the quality management (management of non conformances, quality control, coverage of QA and QC programmes).

One of the main findings from this study is that the same safety requirements should be applied to the NPPs construction stage as well as to the operation stage, concerning for instance the safety management.

As a conclusion, this study shows that it is possible to raise lessons learnt and recommendations about construction from operational experience, i.e. based on event reported during operation of NPPs.

Nevertheless, it has to be underlined that much more recommendations could have been raised if more detailed information about construction, as detailed as in some WGRNR reports, had been available. Indeed, the IRS database does not report information about non-conformances which were detected and corrected prior to the plant operation, as they have no consequences on the safety of the operating plant. Moreover, in most of the IRS reports describing events due to construction deficiencies, the lessons learnt for current of future construction projects were often not fully considered by the writers who focused more on the corrective actions for the operating plant.

As a result, it would be highly valuable to increase the sharing of construction experience using a systematic and timely process, taking into account the high number of NPP currently under construction in the world.

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SESSION THREE

US Nuclear Regulatory Commission, Office of International Programs (USNRC/OIP) International Regulatory Development Partnership (IRDP)
Scott Newberry (AdSTM, Inc, USA)

Human and Organisational Factors in the Licensing Process for New NPPs: the Swiss Approach
Cornelia Ryser (ENSI, Switzerland)

UK Regulatory Expectations for the Development of Licensee Organisational Capability
Craig Reiersen, Steve Gibson (NII, UK)

Human and Organisational Factors in the Licensing Process for New NPPs: The Swiss Approach

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Past and recent construction as well as operating experience has shown the importance of giving due attention to human and organisational factors (HOF) in the design and construction of new NPPs. The present paper primarily deals with the development of regulatory requirements by the Swiss Federal Nuclear Safety Inspectorate (ENSI) concerning HOF engineering aspects, i. e. the design of the work systems of the new NPP (technical systems, work tasks, operating organisation). The requirements concerning quality and project management or the management system in general are not in the main focus of this paper.

The approach and requirements presented are not yet fully implemented in the Swiss regulatory framework and are still being worked on. Therefore the issues described in the paper have to be considered as provisional and might still be subject to some change in the next months as the approach and requirements are being further developed and finalised.

1. Human and organisational factors in the application for general and construction licenses

One of the design principles for NPPs, as defined by the Swiss Nuclear Energy Ordinance (NEO)[9], states that work stations and processes for the operation and maintenance of the installation must be designed so that they take account of human capabilities and their limits. Correspondingly, the ENSI requires from applicants for a construction license (called “applicants” in the following) that the HOF be adequately considered from the beginning and in all phases of the project of building a new NPP. The declaration of a commitment to develop and implement a thorough approach to integration of the HOF into the project was therefore expected from the three current applicants already in their general license (site license) application. For construction license application this systematic approach has to be developed and presented in detail in the submissions to the authorities.

2. The principles for integration of human and organisational factors in new build projects

The ENSI’s approach is based on one hand on well established approaches and methods applied by other safety authorities concerning human factors engineering and their experience gained in past and current new build projects and modernization projects, including in reactor design licensing processes (e. g. US NRC, the French ASN, the British HSE etc.). On the other hand strong emphasis was put on developing an approach commensurate to the Swiss licensing process and the specificities of the Swiss projects.

The ENSI builds on the basic assumption that the applicant is and must remain responsible for its project, particularly with respect to safety and quality. Therefore it is the applicant’s responsibility to determine the way to organise its project and hence also the way to integrate the HOF into the project in order to assure the safety of the installation. The ENSI is facing the challenge not to be too prescriptive in order not to jeopardize the responsibility principle, but still giving enough guidance to allow for a suitable integration of the HOF in new build projects and positive communication between ENSI and the applicants.

[9] Cf. Art. 10 NEO

The ENSI formulated a series of basic principles[10] the applicants shall apply when developing their own approach for HOF integration:

1. *Integrated, socio-technical perspective*: the nuclear installation is a socio-technical system; human, technology and organisation shall be considered and designed in an integrated manner.

2. *Systematic and continuous approach*: the HOF shall be considered in a systematic and continuous way along the course of the whole project and during all phases of the life cycle of the installation.

3. *Systematic consideration of operating and project experience*: existing experience from operation of predecessor or similar nuclear installations and with similar technologies from other industries, experience from the realization of other projects, as well as the current state of the art as conveyed in international standards and other literature sources shall be systematically analysed and integrated into the project.

4. *User-centred approach*: The design of work systems (i. e. technical systems, work tasks, operating organisation) shall be targeted at their (future) users. Users must be involved into the design process and the applicant must have a clear understanding of user and organisational requirements.

5. *Iterative approach*: Design solutions shall be developed in an iterative way; preliminary design solutions shall be tested (through verification & validation) with user involvement throughout the design process. Feedback from the tests shall be incorporated into the refinement of the design solutions.

6. *Multi-disciplinary project teams*: Project teams shall encompass all relevant skills and experience, including specialists in the HOF area.

3. Topical areas to be covered within the application for a construction license

Within the application documentation for a construction license, the applicants are requested to submit information concerning the integration of the HOF in the following areas:

- *Design process of the NPP*: the object of this part of the documentation is the design of the means (systems, tools, devices) that individuals or groups of individuals (particularly operating, maintenance and safeguard personnel) need to fulfil their tasks, which have an impact on the nuclear safety of the plant. The focus shall not be limited to the human-machine interfaces in the (main) control room, but shall include local control rooms and work stations, the layout and accessibility of buildings, rooms and installations, operating and emergency procedures, as well as organisational issues of the plant design. All plant states shall be considered.

- *Construction phase*: relevant issues include, for instance, safety culture and safety management issues in general, leadership, learning organisation, event analysis and experience feedback, multi-cultural management on construction site, construction staff training, development and implementation of tools to improve human performance etc.

- *Development of the future operating organisation*: During the construction phase, the operating organisation gains more and more topicality the closer the commissioning phase comes.

[10] Cf. particularly ISO 13407

However, the development of the operating organisation and the transition from the construction and project organisation to the operating organisation must be effected in a systematic way and must be planned and started timely in the project.

4. Documents to be submitted with the construction license application

For the three topical areas described, applicants are requested to submit a series of documents. These documents are partly process oriented, i. e. they describe the procedures, methods and organisation to integrate the HOF into the project phases, and partly product oriented, i. e. they describe the concrete measures and the results of the implementation of the processes.

Design process

- *HOF programme for the planning and design phase*: This document is process oriented and describes in concrete and specific terms how the HOF will be integrated in the planning and design of the plant. It explains the objectives and the scope of HOF integration, the technical programme (activities, steps, milestones, documentation etc.) and the methodologies to be applied along the project phases, the way specific HOF qualifications are integrated into the project team, etc.

- *General (technical and organisational) orientation for the design of HSI and procedures – Preliminary analysis of HOF related impact*: This product oriented document contains an analysis of the impact of the technical and organisational options pursued by the applicant on safety and the personnel. This analysis helps the applicant (as well as the ENSI) to grade its efforts related to HOF integration. For instance, if the pursued option is in use in other nuclear installations, and well documented design and operating experience is available, providing evidence for positive safety impact, less effort will have to be put into activities to demonstrate that the option has no negative impact. Does, on the other hand, the applicant aim at an option with strong innovative nature, for which no design and operating experience is documented yet, deeper analyses will be needed to demonstrate the positive contribution to safety of that option. These first two documents and accordingly the related activities need to be compiled early in the project. They represent for the applicants an important input for the bidding documentation and criteria for the selection of a plant designer.

- *Preliminary inventory of the means to be used by the personnel for monitoring, control and maintenance of the plant*: The solutions adopted within the specific plant design and their HOF related impact on safety have to be analysed and described in as much detail as is already possible at the moment of the submission of the construction license application.

Construction phase

- Within the *Management System for the Construction Phase* the applicant has to demonstrate that and how the HOF will be considered during the construction phase. In particular, the applicant must describe its concrete measures concerning, for instance, safety culture on the construction site, staffing and training programmes during the construction phase, analysis of events and experience feedback, etc.

Development of the future operating organisation

- *Concept for the development of the future operating organisation*: A concept on how the applicant will proceed to develop its future operating organisation is expected within the Safety Analysis Report for the construction license application (or as a separate document). A comprehensive documentation will be required in the operating license application.

The development and implementation of the HOF related programmes and activities and the submission of the documents and communication with the ENSI are the responsibility of the applicant. The applicant cannot shift this responsibility to the designer organisation(-s), although close cooperation with the latter is of course necessary. In addition, crucial activities concerning the integration of HOF into the design process, e. g. the definition of the applicant's HOF approach, its requirements (e. g. derived from the fact that the new plant is going to be built on the site of an existing unit, or from the regulatory requirements) and the general HOF related specifications (e. g. the applicant's expectations concerning the role, tasks and organisation of the operating staff, the degree of automation of the plant, the use of digital technologies in the control room, etc.) (cf. the above described document "*General ... orientation for the design of HSI and procedures…*"), have to be performed before the selection of a reactor type and a designer organisation.

5. Outlook

The ENSI is currently developing its approach for the assessment of the application documentation. Among other things, it is considering to require the submission and to stagger the assessment of parts of the application documentation over a period of time previously to the official submission of the application for the construction license. This approach seems to be highly desirable for certain subject areas characterised by irreversibility of the applicant's activities in the time period before the submission of the application. For instance, some activities can permanently compromise the project and have negative impact on safety if performed in a suboptimal way, or they cannot be caught up if not performed at the right time. This is particularly true, among others, for the HOF area where the activities to properly integrate the HOF into plant design have to be planned and performed from the beginning of the project in a systematic and competent way.

References

[1] Swiss Nuclear Energy Ordinance (NEO), English translation available: http://www.admin.ch/ch/e/rs/7/732.11.en.pdf

[2] International Organization for Standardization (1999). Human-centred design processes for interactive systems. ISO 13407:1999. (Recently replaced by ISO 9241-210, Human-centred design for interactive systems)

UK regulatory expectations for the development of licensee organisational capability

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The Nuclear Installations Inspectorate (NII) is responsible for the licensing and regulatory oversight of new nuclear power reactors in the UK. NII recognises that effective licensee leadership and management for safety are instrumental in the safety of new and existing nuclear installations. NII is consequently placing considerable emphasis on seeking assurance that prospective licensees develop an adequate organisational capability to manage and deliver nuclear safety in addition to constructing a design that has passed through a rigorous assessment process.

In order to make NII's expectations clear, and support a consistent approach to interactions with prospective licensees, NII has produced a suite of related guidance to help its Inspectors assess and influence the development of licensee organisational capability. This includes:

1. The safety management prospectus

Before the Nuclear Installations Inspectorate will grant a Licence the applicant should submit a Safety Management Prospectus (SMP**)** demonstrating that it will have adequate:

– management structure

– safety management arrangements

– resources

Thus the SMP is a fundamental element of the licensing basis. The SMP is akin to an "*Organisational Safety Case*" Guidance has been developed and can be viewed on Nuclear Directorate's website (T/AST/072).

There are a number of elements to the SMP which must be made explicit. These are:

1. It should provide a clear description of the type of activities carried out on the licensed site(s). This will provide information on the hazards and risks to ensure that the organisational arrangements are proportionate.

2. It should describe how the organisational structure meets the nuclear Safety, Security and Environmental (SSE) management needs of the business. Here we expect to see the nuclear policies and the organisational structure, functions and responsibilities to deliver the policies.

3. It should set out the organisation's approach to the governance of nuclear SSE. We expect to see high-level descriptions of the systems and processes for directing and controlling activities and maintaining oversight. It should describe the role of the Board and the Executive. It should also describe the role of the safety governance committees. The challenge function should also be described. It should include the strategy for developing and maintaining effective leadership and culture.

4. It should provide the strategy for developing and maintaining a licensable organisation with the right structure, resources and competences to deliver effective nuclear SSE. This should include resource strategy and oversight. This should also include the policy & strategy on use of contractors and retention of intelligent customer and design authority capability. And also describe the approach to developing and maintaining the "Nuclear Organisational Baseline".

5. It should show how a "learning organisation" culture is fostered. It should include an explanation of how the organisation absorbs and responds to lessons both from within and outside the organisation. It should describe the performance monitoring arrangements. It should also describe the arrangements to secure and promote open and learning culture. It should also describe how it will foster a challenge culture.

6. It should provide the strategy for managing change and maintaining live and effective management arrangements. This should include a strategic approach to review of factors covered within the SMP. It should describe the arrangements for maintaining these factors and assuring continued adequacy. It should also have links to arrangements for compliance with Licence Condition 36 [Control of Organisational Change].

2. The "nuclear baseline"

Licensees must demonstrate that they understand, and maintain, the resources and competencies needed to manage nuclear safety effectively. This is done through the nuclear baseline. The NII does not specify what the nuclear baseline should look like however it does expect the following factors to be considered:

- Organisation structure.
- Identification of <u>**all**</u> roles that impact on nuclear safety.
- Identification of numbers of personnel needed <u>**within**</u> licensee.
- Identification of *intelligent customer* roles.
- Reference to management of contractors.
- Identification of vulnerabilities.
- "Route map" to underpinning processes.
- Justification – analysis, performance Indicators etc.
- Reference to Baseline in change proposals.

3. Intelligent Customer capability and use of contractors

Licensees are expected to show that they maintain the core capability to understand the nuclear safety case. They must also show that they are able to retain control of nuclear safety at all times and manage work carried out on its behalf by contractors. They must:

- Understand the hazards and how to control them.
- Be in control of activities on its site.
- Possess detailed knowledge of the plant safety case.
- Directly employ, or otherwise source, sufficient SQEP'd staff to deliver these activities.
- Have a process to ensure that it retains sufficient in-house capability.
- Demonstrate how it achieves the above.

Guidance is under revision – T/AST/049.

4. Design Authority

Licensees are expected to define the role and function of the Design Authority and show that this function is adequately resourced, and operates with the suitable authority, to maintain plant design knowledge and integrity

- Expectations set out in INSAG -19.
- Key points:
  - NII accepts "ultimate" design knowledge rests with vendor.
  - Licensee must acquire sufficient capability to understand the need for, and make, decisions that affect nuclear safety.
  - NII expects gradual transfer of this capability from vendor to licensee at appropriate stages through construction and commissioning.
  - NII expects plan for this transfer with the end point being establishment of a DA within the licensee's organisation.

5. Licence Condition Compliance Arrangements

The standard licence has 36 conditions attached to it and from the date that the licence comes into force the licencee must have adequate arrangements for all the conditions. There are three elements to adequacy:

- *make* arrangements,
- *implement* the arrangements,

– *assure effectiveness.*

Arrangements can be proportionate to hazards during build programme. This means that what is considered to be adequate varies through the life of the licence. But it is for the licensee to ensure adequacy. Where relevant, compliance arrangements should encompass the whole licensee not just the site and includes the Board and Executive Team e.g. Licence Condition 12 – Competence.

6. Development of Organisational Capability

The SMP, Baseline & management arrangements should provide sufficient confidence in organisational capability to enable HSE to grant a licence, initially, to show the company has the capability to function as a licensee consistent with current stage of installation. NII will assess the arrangements and inspect their implementation for adequacy prior to granting a licence.

Baseline and arrangements must evolve in line with, and ahead of, the build and commissioning programme. The Licensee should have plans to achieve full Design Authority capability prior to operation. NII may place regulatory hold points, or expect licensee to set its own, for “organisational build” as well as plant build.

SESSION FOUR

NEA/CNRA Report on the Survey on Regulation of Site Selection and Preparation
Philip Webster (CNSC)

Insights from Siting New Nuclear Power Plants in the Central and Eastern United States
Clifford Munson, Andrew Kugler (NRC, USA)

Regulatory Frameworks and Issues on Site Selection and Site Evaluation for Korean NPPs
Hyunwoo Lee, Chang-Bock Im, Myunghyun Noh, Taek-Mo Shim, and Sang-Yun Kim(KINS, South Korea)

Siting Practices and Site Licensing Process for New Reactors in Canada
Marcel de Vos (CNSC, Canada)

Potential Nuclear Power Plant Siting Issues in the United Arab Emirates
Waddad Al Hanai (FANR, United Arab Emirates)

NPP Siting in Western Part of Java Island Indonesia: Regional Analysis Stage
Achamad Sarwiyana Sastratenaya (BATAN, Indonesia)

Temelin 3, 4 Siting
I. Kubanova (CEZ, Czech Republic)

Near Regional and Site Investigations of the Temelin NPP
Ivan Prachar (Energopruzkum Praha Ltd, Czech Republic), Jiri Vacek, Pavel Heralecky (CEZ, Temelin NPP, Czech Republic

Regulatory Issues and Challenges in Preparing for the Regualtion of New Reactor Siting: Malaysia's Experience
Azlina Mohammad Jais (AELB, Malaysia), Halimah Hassan (DOE, Malaysia)

Insights from Siting New Nuclear Power Plants in the Central and Eastern United States[11]

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Mr. Andrew J. Kugler, United States Nuclear Regulatory Commission

The staff of the U.S. Nuclear Regulatory Commission (NRC) has completed its review for four early site permits and for four standard reactor designs. It is currently reviewing applications for fourteen combined license applications and three additional reactor designs. The staff is applying lessons it has learned from the reviews to date to the review work going forward.

The licensing process being used by current applicants[12]differs significantly from that used by the current operating fleet. The previous process required two steps. First an applicant had to obtain a construction permit to build the plant. Then, near the end of construction, the applicant had to obtain an operating license. Under the process in Part 52, an applicant can apply for a combined license (COL) that allows construction and (once certain conditions are met) operation of a new plant – a one-step process. An applicant for a COL may reference an early site permit (ESP[13]), a standard design certification[14], both, or neither.

In addition to developing Part 52, the NRC also revised 10 CFR Part 100 by adding Subpart B, which includes sections 100.21, "Non-seismic siting criteria," and 100.23, "Geologic and seismic siting criteria." The NRC staff also revised the Standard Review Plan (NUREG-0800) and developed Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)." The NRC staff incorporated into the revision of NUREG-0800 and development of RG 1.206 some early lessons learned from its review of the first three ESPs.

Staff work begins before the application is received, as the staff interacts with the applicant to identify issues that will require special treatment or specific staff resources. After the application is submitted, if the NRC finds the application acceptable, the safety and environmental reviews begin, proceeding in parallel. The safety review culminates in the issuance of a safety evaluation report (SER) after it has been reviewed by the Advisory Committee on Reactor Safeguards. The environmental review[15] results in an environmental impact statement (EIS). Both of these documents are then reviewed in an adjudicatory hearing by the Atomic Safety and Licensing Board. The Board makes an initial decision, after which the Commission will make it's decision.

[11] Note: This paper was prepared, in part, by an employee of the United States Nuclear Regulatory Commission on his or her own time apart from his or her regular duties. NRC has neither approved nor disapproved its technical content.

[12] See Title 10 of the Code of Federal Regulations (10 CFR) Part 52, "Licenses, certifications, and approvals for nuclear power plants".

[13] An ESP is an approval of a site for one or more nuclear facilities under 10 CFR 52, Subpart A.

[14] A standard design certification is a Commission approval of a final standard design for a nuclear power facility under 10 CFR 52, Subpart B.

[15] See 10 CFR Part 51 for the NRC's environmental regulations.

1. Site Selection Lessons Learned from Site Safety Reviews

The updates to the regulations and regulatory guidance documents, described above, laid the groundwork for the staff's review of the first early site permit applications in 2003. In the area of geology and seismology, the applicants followed the guidance in 10 CFR Part 100.23 and specifically described in RG 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion." In order to characterize the seismic hazard, RG 1.165 recommended the use of probabilistic seismic hazard analysis (PSHA) for determining earthquake inputs for seismic design of nuclear power plants. Previously, Appendix A to 10 CFR Part 100 specified deterministic seismic hazard analyses (DSHA) for determining the Safe Shutdown Earthquake (SSE) ground motion levels. Under DSHA, an earthquake scenario, often a worst-case or maximum credible event is assumed in terms of magnitude and location. In contrast, PSHA incorporates the effects of all the earthquakes capable of affecting the site and includes the uncertainties in the earthquake size, location, rate of recurrence, and ground motion amplitude in the analysis.

RG 1.165 specifies a "reference-probability" hazard-based approach for specifying the SSE ground motion levels. However, of the three early site permit applications received by the NRC staff in 2003, two of the applicants chose not to use the reference-probability approach outlined in RG 1.165. Both of these applicants considered that the reference probability value was based on outdated seismic hazard evaluations and resulted in overly conservative site SSE ground motion levels for their sites. Rather than using the RG 1.165 approach, these two applicants adopted the seismic performance-based approach for determining the design level ground motion, which was under development as part of ASCE Standard 43-05. The adoption of this new approach resulted in considerable additional review time for the NRC staff and a re-evaluation of the use of the reference-probability approach for determining the SSE. The major difficulty with the use of the reference-probability approach is that as the hazard evolves with new data, models and research, it is difficult to evaluate the effect of these changes on the reference probability. The NRC staff's approval of the performance-based approach in place of the reference-probability approach led to the replacement of RG 1.165 with RG 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion."

The NRC staff is currently reviewing several Combined License (COL) applications that reference RG 1.208 and implement the performance-based approach. RG 1.208 also specifies that the seismic source characterizations developed by applicants for sites in the Central and Eastern United States (CEUS) may reference the Electric Power Research Institute (EPRI) seismic source models, developed in the 1980s, as a starting point for their site PSHAs. Recognizing that many new studies have resulted in re-evaluation of the seismic hazard for the CEUS, each of the ESP and COL applicants have examined these new studies before either using the original or updating the EPRI seismic models. Currently the staff spends a considerable amount of effort validating the continued use of the older EPRI source models, as well as revisions to the models proposed by the applicants.

In the area of hydrology, during its review of recent ESP and COL applications, the staff identified recurring issues involving inconsistencies and gaps between guidance provided in SRP Section 2.4.13 and SRP Section 11.2 with Branch Technical Position 11-6 relating to on-site hydrogeologic testing and measurements, conceptual model development of radionuclide transport in groundwater, and analysis of the radiological consequences of releases. To address the inconsistencies the staff developed two Interim Staff Guidance (ISG) documents. The first ISG (ISG-013) emphasizes the definition of the location and conditions of the assumed release, the role of mitigating design features, the definition of exposure scenarios, and potential technical specifications limiting radioactive tank contents. The second ISG (ISG-014) provides additional guidance on analyzing the aqueous transport of radionuclides through the subsurface with groundwater through the use of a structured hierarchical approach. This approach begins by determining the basic conditions for the analysis such as hydrogeologic characteristics, release location, groundwater pathways, travel time, and release volume. If mitigating design features are present and acceptable, the analysis is concluded, otherwise, contaminant transport analysis is performed using progressively more complex modeling techniques, as needed.

Lessons learned in the area of meteorology have also resulted in an ISG to clarify the NRC position on identifying winter precipitation events as site characteristics for ESP and COL applicants and site parameters for DC applicants for determining normal and extreme winter precipitation loads on the roofs of Seismic Category I structures. 10 CFR Part 52 requires that the meteorological characteristics of the proposed site should be identified with consideration for the most severe of the natural phenomena that have been historically reported for the site and surrounding area with sufficient margin for limited accuracy and period of time. The staff's ISG clarifies that not only should the historical maximum snowpack be identified, but also the historical maximum snowfall event should be determined as it might exceed the historical maximum snowpack. The ISG also provides guidance on how data reported as snowfall or snow depth should be converted to an equivalent weight or load on the roof of a structure. Finally, the ISG specifies the appropriate loading combinations to determine the design live load on the roof.

The examples cited above with regard to seismology, hydrology, and meteorology illustrate that the staff has proactively addressed the lessons learned from its reviews of ESP, COL and DC applications. As additional issues arise during the review of siting and design applications, the staff will continue to update its guidance documents in order to resolve inconsistencies and/or gaps.

2. Site Selection Lessons Learned from Environmental Reviews

The staff has identified a number of regulatory issues during its reviews. The information provided in the applications has presented problems in various respects. For example, a number of applicants have chosen to rely on old data or data from nearby sources. However, the basis for the applicability of this data is often unclear. The staff has also found that, in some cases, information and data provided by the applicant to other agencies differs in critical aspects from information and data provided to the NRC. In addition, in many cases applicants have provided data and a conclusion, but failed to make a case that the data supports the conclusion. We sometimes refer to this as "telling the story" – applicants need to provide a clear path from the data to the conclusions.

Another challenging aspect of the current reviews, for both the applicants and the reviewers, is the evolving regulatory environment. As an example, how far do we go in addressing global climate change[16]? Do we discuss reduced greenhouse gas emissions from nuclear plants? Which models do we use to predict changes in precipitation and temperatures? In a related topic, how do we manage "regulatory creep" (the tendency for reviewers to look for, and expect, more from each successive application)?

The staff has also experienced challenges regarding interagency coordination. For any review, there are a large number of Federal, State and local agencies that will have some role in licensing, permitting, or otherwise authorizing all or part of the project. And in many cases these agencies also consider site selection as a part of their review. At the outset it is useful to determine which agencies the applicant has already contacted and the extent of their interactions. The NRC staff has often been surprised by the relatively low level of communication and coordination between the applicant and affected agencies before the start of the NRC review. Once we've identified which agencies are involved, we need to determine each agency's role and authority. It's critical for each agency to understand what the other agencies are,

[16] See Commission Memorandum and Order CLI-09-20 dated November 3, 2009, in the matters of Duke Energy Carolinas, LLC, and Tennessee Valley Authority, regarding the consideration of greenhouse gas emissions and "carbon footprint" in licensing proceedings for new nuclear power plants.

and are not, going to be doing. From that point of understanding, we can coordinate review work to avoid duplication and potential conflicts.

After the first three early site permits, the NRC staff greatly expanded its efforts involving pre-application interactions. These efforts are aimed at both the applicants and the other agencies and have led to improved applications and better coordination with other agencies.

The NRC staff is also adapting to the evolution in the participation of public stakeholders. During the last big round of licensing for new plants, public access to data and information was difficult. And coordination among citizens had to be performed by phone, by mail, or in person. Today, the Internet, blogs, and other tools allow citizens instant access to information and provide the ability to rapidly share information and ideas with large numbers of people. Today's public interest groups are typically better informed and better organized, and they expect their Government agencies to keep up. Establishing a good web page for projects is only the beginning for us. We have to be more effective in reaching out to the groups and in working with them. We also need to plan for a much higher level of public involvement. But while we do this, we must not forget that there are still people who'll be affected by our decision who aren't linked in through the Internet. So we can't abandon the old-fashioned methods – paper documents in libraries for example – completely.

Finally, there are some technical matters that have arisen from the work to date. As I'm sure we've all seen, the Internet and Geographic Information Systems are great tools for our reviews. Just like the public interest groups, we can now access more information more quickly to get our jobs done. But it's a two-edged sword. Reviewers can face information overload or they can find themselves getting bound up in conflicting information from various sources. So it's important to manage the information flow and be selective regarding the sources used for the review.

An enormous challenge facing many of the proposed plants is access to and availability of water for cooling. Many parts of the U.S. are already challenged for water and adding big power plants with big heat loads exacerbates the problem. Access to a cooling water source is often the main driver when identifying alternative site locations. Applicants and regulatory agencies need to look for innovative solutions. Examples might include re-use of waste water or the use of dry, or a combination of wet and dry cooling systems. But keep in mind that every alternative brings with it new challenges.

The staff has also learned from recent experiences regarding the application of site selection process guidance, primarily the NRC staff's Environmental Standard Review Plan (ESRP)[17], and the Electric Power Research Institute's siting guide[18]. The staff has found that all of the applicants have deviated from the guidance in at least some respects. In some cases, the basis for the deviations is provided. But in most cases it isn't and the staff must request additional information to resolve the differences. The staff has also found in some cases that the applicant has not applied its siting criteria in the same way to all sites. This occurs most often when applicants attempt to use the results of older site selection studies. But we have also seen this problem occur when an applicants' selection process "evolves" over time.

A number of applicants have selected sites where they already have operating nuclear units. While on the surface this would appear to make sense, we have to look closely at the basis for the choice and at other options. For example, if existing units are already withdrawing a significant amount of water from a river, adding one or more new units at that location may lead to unacceptable impacts to the river. The staff must

[17] *Standard Review Plans for Environmental Reviews for Nuclear Power Plants,* U.S. NRC, NUREG-1555, March 2000, but including 2007 revisions. Site selection is addressed in Section 9.3.

[18] *Siting Guide: Site Selection and Evaluation Criteria for an Early Site Permit Application, Final Report*. Electric Power Research Institute (EPRI), 2002, Product ID: 1006878, EPRI, Palo Alto, California.

also balance the search for good alternative sites with the amount of effort involved. Environmental guidance leads us to look for sites that are "among the best" that can be found[19]. This means that we have to show that we used a sound process that would be expected to find good sites. But we don't have to prove that the chosen site is the best site possible.

Finally, the staff has found the comparison of sites to be challenging because in most cases you're not comparing impacts to similar resources. So, for example, you might be impacting a commercial fishery at one site, and impacting a historic resource at another site. Or you might be impacting a school system at one site and groundwater quality at another. Balancing those impacts against each other is difficult. The staff uses a scale of impacts (SMALL, MODERATE, and LARGE[20]) in its reviews and this scale can help. But in the end, professional judgment must be applied across the affected resources to come to a conclusion.

[19] *Standard Review Plans for Environmental Reviews for Nuclear Power Plants,* U.S. NRC, NUREG-1555, March 2000, but including 2007 revisions. See Section 9.3, Revision 1, page 9.3-6.

[20] Code of Federal Regulations, Title 10, *Energy*, Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," Subpart A, Appendix B, Table B-1, footnote 3.

Regulatory Frameworks and Issues on Site Selection and Site Evaluation for Korean NPPs

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1. Introduction

The regulator – the MEST (Ministry of Education, Science & Technology) technically supported by KINS (Korea Institute of Nuclear Safety) currently regulates the operation of 20 commercial nuclear power reactors owned by the exclusive government-funded nuclear utility - KHNP (Korea Hydro and Nuclear Power Co., LTD) that generates electricity at 4 NPP (nuclear power plant) sites, i.e., Kori, Wolsong, Ulchin and Yonggwang, in Korea. KHNP is constructing another 6 reactors at Shin-Kori (2 OPRs and 2 APRs) and Shin-Wolsong (2 OPRs) sites adjacent to existing NPP sites and currently seeking new sites for more reactors (Figure 1). Two more APRs are currently under safety review for the CP (construction permit) at Shin-Ulchin site.

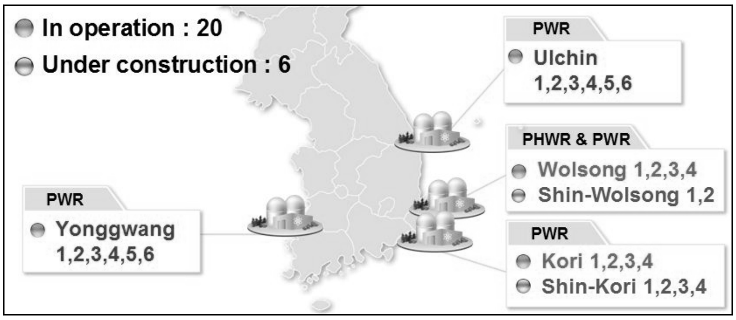


Figure 1. Nuclear power plants in Korea.

This paper introduces overall regulatory practices on site selection and site evaluation for NPP installation in Korea and some technical issues developed during recent license review processes, including identification of population centers, application of submerged intake and discharge structures at a multireactor site, and geological/seismic issues.

2. Regulatory frameworks and review process related with site evaluation

Under Korean nuclear regulatory frameworks, there are three major steps to confirm the site suitability and the site design parameters for the NPP installation - Site selection, Site evaluation, and Pre-operational inspection. KHNP selects a nuclear power plant site based on specific technical regulatory requirements, i.e., MEST Notices 2008-07 (Location, Population, Human-Induced Hazards, Geology, Seismology & Geotechnical Engineering), 2008-08 (Meteorology), and 2008-09 (Hydrology) (Table 1).

Table 1. Laws and technical standards applied to NPP site.[21]

Law	Articles 12 (Construction Permit) & 22 (Operation Permit) of Atomic Energy Act (AEA)
Ordinance of the Minister of Education, Science and Technology (MEST)	<u>Enforced Regulations concerning the Technical Standards of Standards of Reactor Facilities, etc.</u> - Articles 4 (Geology & Earthquake), 5 (Siting Limitation), 6 (Meteorological Condition), - Articles 7 (Hydrology and Oceanography), 8 (Human-induced Accidents), 9 (Emergency Plan)
Notice of the MEST	<u>MEST Notice 2009-37</u> [004] Technical Standards for Location of Nuclear Reactor Facilities [027] Regulation on Pre-operational Inspection of Nuclear Reactor Facilities [029] Technical Standards for Investigation and Evaluation of Meteorological Conditions of Nuclear Reactor Facilities [030] Technical Standards for Investigation of Hydrological and Oceanographic Characteristics of Nuclear Reactor Facility Sites [036] Objects of Consultations due to Installation of Industrial Facilities, etc. around the Nuclear Facilities
Review Guidance of KINS	KINS/GE-001, 1999, Safety Review Guide (SRG) for PWR reactors (Chapter 2) □ **now under review**

The proposed site is subject to an approval of a License for the Site Preparation from the MKE (Ministry of Knowledge and Economy), in accordance with Electric Source Development Promotion Act. The regulator is barely involved at this stage, just providing comments to the MKE on the Preliminary Radiation Environmental Report. KHNP shall then prepare and submit the result from the site evaluation of the proposed site in the PSAR (Preliminary Safety Analysis Report) and the RER (Radiation Environmental Report) as licensing documents of its application for the CP. KINS reviews the application documents, at the MEST's request, and provides the MEST with the Summary of the Safety Evaluation Report for its decision upon the issuance of the CP (Figure 2).

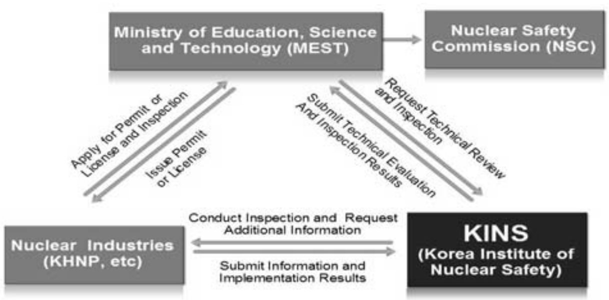


Figure 2. Interactive Mechanism in NPP Regulation.

After the issuance of the CP, KHNP is able to start excavation of the NPP foundations to the designed level(s) for the installation of the facilities. At this stage, KINS performs Pre-operational inspections

[21] http://www.kins.re.kr/english/nuclear/nuc_policy_01.asp.

according to the inspection plan in order to confirm the foundation stability and readiness to the facility installation.

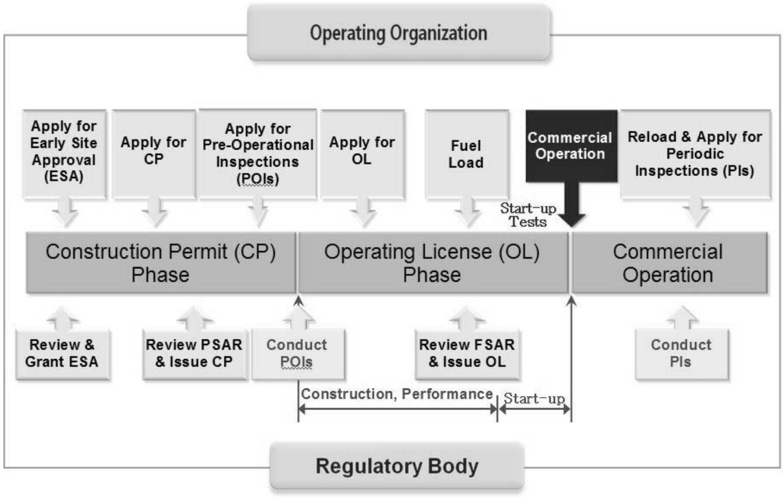
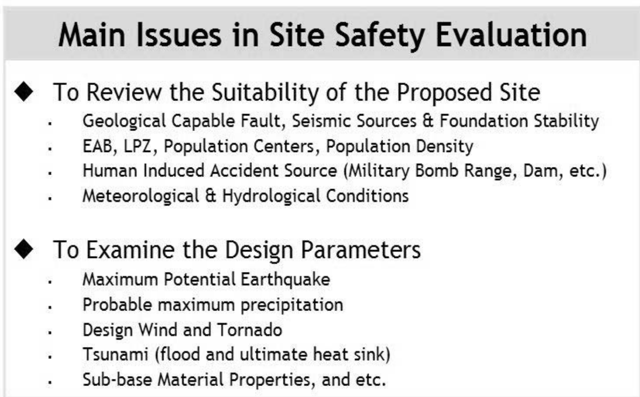


Figure 3. Licensing system and procedures.

3. NPP site review and technical issues

Site review for NPPs aims mainly at confirming the suitability and the design parameters of the proposed site. Main activities related to the site review are presented in the figure 4. Some of the regulatory experiences during recent safety review and pre-operational inspections, constraint to site evaluation, are described as follows.



Identification of population centers. An applicant for a reactor license is required by the MEST Notice No. 2009-37 (004), to designate a population center distance, defined as the distance from the nuclear reactor center to the nearest boundary of a densely populated center containing more than about 25,000 residents. The population center distance must be at least one-third times the distance to the outer boundary of the low population zone (LPZ) and the outer boundary of a population center shall be determined upon consideration of population distribution. To prevent controversy caused by the uncertainties in defining a clear boundary of population center, a computer code analyzing population distributions and defining outer boundary of population centers around a nuclear power plant has been developed (Lee et al, 2008)[22].

[22] Lee, H., Im, C.-B. and Hyun, S.-K., 2008, "Development of Regulatory Guide: Population Center", KINS/RR-641, 79p.

Dealing with capable fault. Tectonic structures located within a 320 km from the proposed reactor center are subject to document/site survey in terms of potential sources of design earthquake and surface faulting at the proposed NPP site. Capable fault with certain size within certain distance from the reactor is subject to a detailed fault investigation, including paleoseismological analysis, absolute age analysis and other Quaternary investigations, to identify peak ground acceleration and probability of surface faulting at the proposed site.

Submerged intake/discharge tunnels are used for the Shin-Kori units. That is because four operating NPPs in the Kori site and another four to come to the adjacent Shin-Kori site would cause rise of sea water temperature for the discharge water that would result in environmental impacts to the seawater (Maximum temperature of discharge water is limited to 40 ℃) and threat to stable cooling water supply for the reactors. Intake structures, including tunnels, are categorized and designed as safety-related structure because they supply emergency service water, and hence most of the constructional processes related to these structures, including the excavation and treatment of the underground openings for the structure installations are subject to pre-operational inspection (Figure 5).

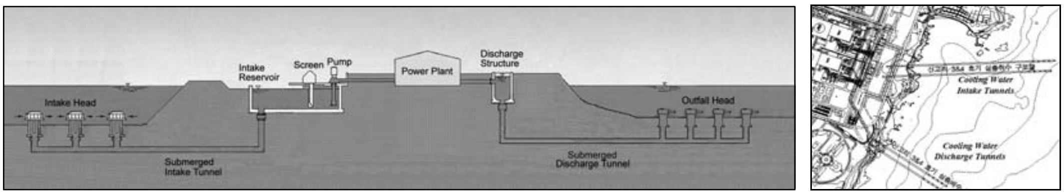


Figure 5. Conceptual profile (left) and plan view (right) showing cooling water intake tunnels applied to the Shin-Kori 3 & 4 units.

Site monitoring system. Because of the uncertainty of natural phenomenon or the lack of data, it is difficult to define clear site parameters in general, especially places fully cultivated/commercialized and tectonically less active like Korea. Monitoring of site parameters is critical and necessary for the specific NPP site located in such places in order to understand the site characteristics and predict long-term behaviors of the site. KINS has been establishing the 'Integrated Site Monitoring System for Nuclear Facility Site' for the purpose of monitoring and analyzing site parameters coming from 4 NPP sites and 1 Low and Intermediate Radioactive Waste Repository.

4. Concluding remarks

Korea has experience of reviewing sites for 20 NPPs, 2 research reactors and 1 radioactive waste repository. There have been questions and safety issues all through the regulatory experiences for the past 30 years and answers to those questions and challenges all the time. It is true that all these experiences should be helpful to the nuclear industry, especially for the new comers.

Siting Practices and Site Licensing Process for New Reactors in Canada

Mr. Marcel de Vos, Canadian Nuclear Safety Commission, Canada

Executive summary

"Siting" in Canada is composed of Site Evaluation and Site Selection. As outlined in CNSC Regulatory Document RD-346 *Site Evaluation for New Nuclear Power Plants*[23] (based on IAEA NS-R-3), prior to the triggering of the Environmental Assessment (EA) and licensing processes, the proponent is expected to use a robust process to characterize proposed sites over the full lifecycle of the facility, and then develop a fully documented defense of the site selection case. This case forms the backbone for submissions in support of the EA and the application for a *Licence to Prepare Site* which will be reviewed by the CNSC and other applicable federal authorities.

The Environmental Assessment process and *Licence to Prepare Site* in Canada do not require a proponent to select a specific design; however, CNSC does not accept a "black box" approach to siting. CNSC balances the level of design information required with the extent of safety assurance desired for any designs being contemplated for the proposed site. Nevertheless, the design information submitted must be sufficient to justify the site as suitable for all future licensing stages. The depth of plant design information contributes significantly to the credibility of the applicant's case for both the EA and application for *Licence to Prepare Site.*

The review process utilizes an assessment plan with defined review stages and timelines. The outcome of these reviews is a series of recommendations to a federal government appointed Joint Review Panel (which also serves as a panel of the "Commission") which, following public hearings, renders a decision regarding the EA, and subsequently, the application for a *Licence to Prepare Site*.

1. Introductory comments about nuclear regulation in Canada

The development, production and use of nuclear energy and the production, possession of use of nuclear substances is regulated solely at the federal level by the Canadian Nuclear Safety Commission (CNSC) under the mandate of the *Nuclear Safety and Control Act* (NSCA) and supporting regulations under the Act. The NSCA provides the Commission with a broad mandate to regulate all activities related to the use of nuclear energy and materials to protect the health, safety and security of persons and the environment; and to respect Canada's international commitments on the peaceful use of nuclear energy. This includes consideration of not only radiological hazards and effects but conventional hazards and effects as well.

The Licensee is the cornerstone of safety and is held accountable by their licence. Under the NSCA, no licence may be issued, renewed, amended or replaced unless, in the opinion of the Commission, the applicant:

[23] RD-346 Site Evaluation for New Nuclear Power Plants, September 2008 by the Canadian Nuclear Safety Commission through the Minister of Public Works and Government Services Canada, ISBN 978-1-100-10578-9

(a) is qualified to carry on the activity that the licence will authorize the licensee to carry on; and

(b) will, in carrying on that activity, make adequate provision for the protection of the environment, the health and safety of persons and the maintenance of national security and measures required to implement international obligations to which Canada has agreed.

The CNSC is a "balanced" regulator which means that it is balanced between purely prescriptive (rule based) regulation and purely non-prescriptive regulation (concept-based using expert opinions). Under this regime, the applicant is expected to propose, based on considerations contained in Regulatory Documents and applicable Canadian Codes and Standards how they will meet the requirements of the Regulations under the *Nuclear Safety and Control Act*. This is intended to allow applicants a measure of flexibility in the methods they will use to support their unique licensing case. The applicant's proposal is then reviewed by CNSC Staff against modern industry practices and against documents under the CNSC regulatory framework.

2. *Site Selection* and *Site Evaluation* in Canada

The selection of a site for the long term development of a nuclear reactor project is not a regulated activity in Canada and the choice of site is largely a matter between the project proponent and the municipalities and provinces / territories involved. The only exception to this practice is when the federal government, under the Ministry of Natural Resources assumes the role of proponent if it directly sponsors a federal (government-run) reactor project. In either event, the CNSC is not involved in the site selection process.

As part of the site selection process, the proponent is responsible for performing site evaluation activities for one or more candidate sites to determine whether the site will be suitable for the full lifecycle of the nuclear reactor project. The term "lifecycle" covers initial preparation, construction, operation, decommissioning and abandonment of the site. Site evaluation is not, in itself, a regulated activity in Canada and the proponent requires only the necessary local and provincial permits needed to conduct their investigative work on the site such as borehole drilling for geological characterization. Once the proponent has narrowed down their choice to a single site, they then proceed to build a detailed licensing and environmental assessment case to demonstrate to the CNSC that the site is suitable for the proposed project over its full lifecycle.

High level criteria for the evaluation of new nuclear reactor sites is described in CNSC Regulatory Document RD-346 *Site Evaluation for New Nuclear Power Plants*[24]. RD-346 represents the CNSC staff's adoption, or where applicable, adaptation of the principles set forth by the International Atomic Energy Agency (IAEA) in NS-R-3, *Site Evaluation for Nuclear Installations*[25]. The scope of RD-346 goes beyond NS-R-3 in several aspects such as the protection of the environment, security of the site, and protection of prescribed information and equipment, which are not addressed in NS-R-3.

Figure 1 below illustrates the inputs a new reactor proponent is expected to consider when developing all of the information that will be needed to defend the site suitability during the ensuing parallel Environmental Assessment and *Licence to Prepare Site* processes. The proponent must consider the requirements contained in two different pieces of federal legislation (the *Canadian Environmental Assessment Act* (CEAA) and the *Nuclear Safety and Control Act* (NSCA)), the criteria contained in RD-

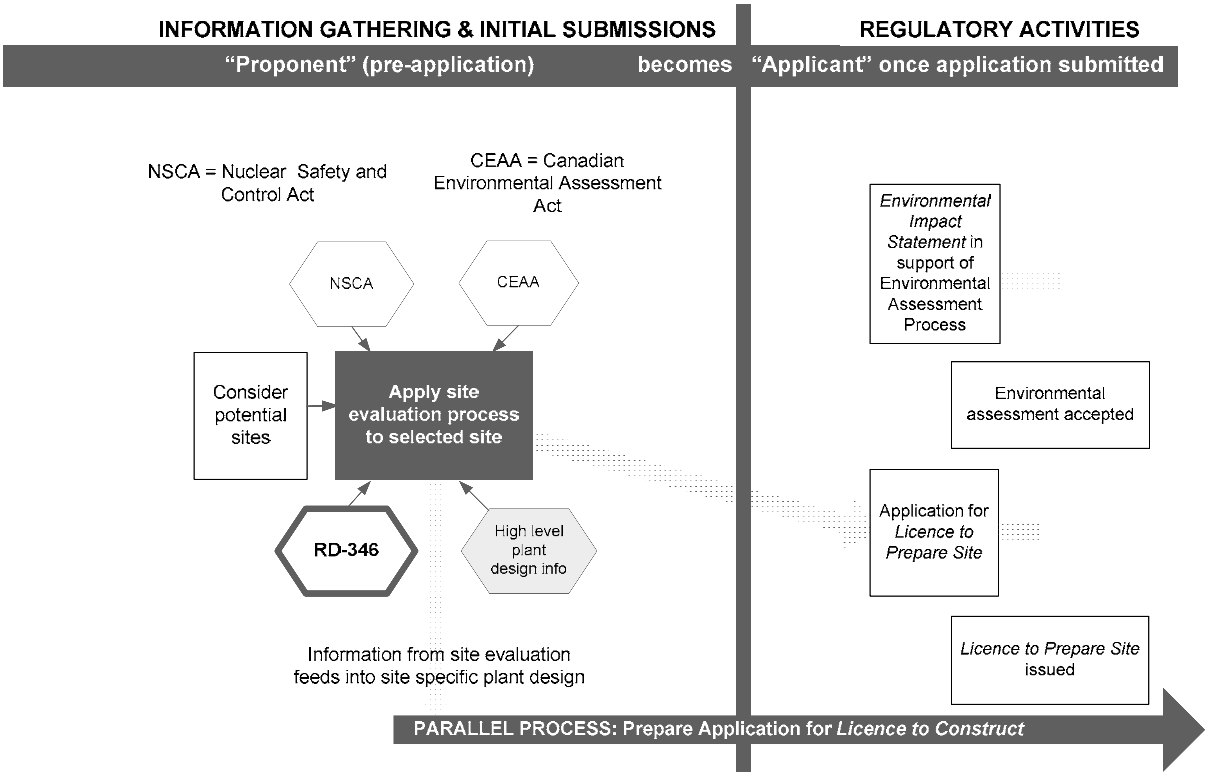
[24] Published September 2008 by the Canadian Nuclear Safety Commission through the Minister of Public Works and Government Services Canada, ISBN 978-1-100-10578-9.

[25] IAEA Safety Standards Series, Safety Requirements No. NS-R-3, *Site Evaluation for Nuclear Installations*, Vienna,

International Atomic Energy Agency, 2003, ISBN 92-0-112403-1.

346, and the design considerations for plant designs they are considering for the proposed site. The latter point is discussed in the next section of this paper.

Figure 1: Site Evaluation Interface with Licensing and Environmental Assessment



3. The level of design information expected to demonstrate site suitability

Regardless of the approach used by a proponent with regards to applying facility design information to their site selection case, a fundamental expectation of the CNSC is that the applicant will be expected to demonstrate a “smart buyer” philosophy. This means that they will be expected to demonstrate a clear understanding of the technologies they are proposing to use and the bases from which safety arguments are developed.

Decisions by a Federal Review Panel and the Commission on an Environmental Assessment under the *Canadian Environmental Assessment Act* and an application for a *Licence to Prepare Site* under the *Nuclear Safety and Control Act* for a new nuclear reactor project may be made with high level facility design information from a range of reactor designs without specifying the technology to be constructed. The design information provided by the proponent must be credible and sufficient to adequately bound the evaluations of environmental impacts and site suitability from a range of reactor designs that might later be deployed at the site and be the subject of an application for a *Licence to Construct*.

The bounding design parameters would have to contain sufficient information to describe the plant-site interface and take into consideration the characteristics of the proposed site. A combination of site characteristics and bounding design parameters will be the focus for comparison with the design characteristics of the actual plant selected for the application for a *Licence to Construct*.

The underpinning of the bounding approach is that the environmental impacts of the reactor design eventually selected for construction should be less than the bounding impacts assessed in the EIS. Similarly, if the site is deemed suitable to host nuclear units using bounding parameters, then the site should also be suitable for any reactor design that falls within the approved bounding envelope.

The design that is eventually selected for construction need not be specifically referenced in the proponent's Environmental Impact Statement (EIS) and *Licence to Prepare Site* application; but would have to fit within the bounding envelope in the approved Environmental Assessment. This evaluation would be performed once a reactor technology is selected and will be required to be demonstrated as part of an application for a *Licence to Construct*.
CNSC will accept qualitative information in support of the site selection case with the understanding that there will be an increased level of regulatory scrutiny during the construction and operation licensing processes to validate the claims made. At the *Licence to Construct* application phase, the applicant will be expected to submit detailed design information that will verify that the evaluations presented previously remain valid.

The less information provided at the onset, the greater the burden on the construction licence review process that fundamental barriers to licensing may appear. So it is in the applicant’s best interest to make their submissions as complete as possible at the onset.

The required level of design information is:

- a technical outline of the facility layout;
- qualitative descriptions of all major systems, structures and components (SSCs) that could significantly influence the course or consequences of principal types of accidents and malfunctions;
- qualitative descriptions of the functionality of the SSCs importance to safety;

- qualitative descriptions of principal types of accidents and malfunctions to identify limiting credible sequences that include external hazards (natural and human-induced), design basis accidents and beyond design basis accidents (severe accidents).

The limiting source terms must consider accident sequences that could occur with a frequency greater than 10^{-6} per reactor year of operation. For those less than 10^{-6}, but sufficiently close to this frequency, the rationale for not including them from further analysis should be provided.

A description of specific (out of reactor) criticality events must be provided showing that these events do not violate criteria established by international standards[26] and national guidance[27] as a trigger for a temporary public evacuation.

If the applicant chooses to pursue a *Licence to Prepare Site* without choosing a final technology for the site, the activities permitted under the issued *Licence to Prepare Site* would be limited to site preparation activities which are independent of any specific reactor technology (for example clearing and grading the site, building site support infrastructure such as roads, site power, water and sewer services).

4. *Conducting* regulatory reviews of submissions for the environmental assessment and application for a *Licence to Prepare Site.*

Under the *Canadian Environmental Assessment Act*, reactor projects of thermal output greater than 25 MW require a Comprehensive Study process to be followed, which is the most detailed form of environmental assessment review.

Large-scale and environmentally-sensitive projects, such as nuclear power plants, usually undergo an environmental assessment called a comprehensive study, which mandates public participation (nuclear power plants are included in the CEAA's *Comprehensive List Study Regulations*, which identifies the projects for which comprehensive studies are mandatory.) However the EA for a new nuclear power plant project would not be conducted as a comprehensive study if the project is referred to a panel or a mediator by the federal Minister of the Environment (following a recommendation by the Commission). A project's EA is referred for review by a panel (also referred to as a "Joint Review Panel") in the following cases:

- when it may cause significant adverse environmental effects, even after taking into account mitigation measures;

- when it is uncertain whether a project will cause significant environmental effects, given the implementation of mitigation measures; or

- where public concerns warrant referral.

Federal agencies involved in projects of this magnitude and complexity are obligated to consider a federal Cabinet Directive on Streamlining Regulation[28] which was issued April 1 2007. This directive drives

[26] Food and Agriculture Organization of the United Nations, International Atomic Energy Agency, International Labour Organization, OECD Nuclear Energy Agency, Pan American Health Organization, United Nations Office for the Co-Ordination of Humanitarian Affairs, World Health Organization, "Preparedness and Response to Nuclear or Radiological Emergency, Safety Requirements", Safety Standards Series No. GS-R-2, IAEA, Vienna, Austria, 2002

[27] Health Canada, "Canadian Guidelines for Intervention during a Nuclear Emergency", Document H46-2/03-32E, Ottawa, Ontario, November 2003 ISBN: 0-662-35147-9.

federal agencies to cooperate and seek review efficiencies to the maximum extent. What this means in the case of a new nuclear reactor project is that the EA and licensing processes are conducted in parallel to achieve an efficient and timely review by maximizing the sharing of common information.

The Joint Review Panel members are also appointed as temporary members of the CNSC Commission with full powers under the *Nuclear Safety and Control Act* to issue the *Licence to Prepare Site* at the end of the process. The Joint Review Panel terms of agreement are contained in a *Panel Agreement* written specifically for each project and co-approved by the Federal Minister of the Environment and the President of the CNSC.

In 2007, a federal agency known as the Major Projects Management Office (MPMO) was formed for the purpose of overseeing and tracking federal reviews as well as Aboriginal engagement and consultation for major resource projects. One of the MPMO's roles is to engage Federal stakeholder agencies, including the CNSC, who will be participating in the review process for a new nuclear reactor project to commit those agencies to achieve their deliverables within a common project timeline. These agency commitments are captured in a document known as a *Project Agreement*, which is unique for each project and ratified by the heads of the participating agencies.

The ultimate deliverable of the regulatory reviews by all federal agencies, which is led by the CNSC, is a series of recommendations to the Joint Review Panel such that the panel can:

- hold public hearings regarding the environmental assessment and *Licence to Prepare Site*;
- prepare and submit a environmental assessment report to the Federal Government for an EA decision from the Governor in Council; and,
- issue the *Licence to Prepare Site* to the applicant if a Governor in Council issues a positive EA decision.

Project Agreements approved to-date have provided for a maximum 6-month period for the initial conformity review of the Environmental Impact Statement (for the EA process), technical analysis and the start of the public notice period for the Joint Review Panel Hearings. Note that the 6 month review period does not include the time taken by the proponent for responses to any information requests from the Joint Review Panel.

The Environmental Assessment (under the *Canadian Environment Assessment Act*) and the *Licence to Prepare Site* (under the *Nuclear Safety and Control Act*) have overlapping but distinct information requirements, yet each process also contains elements not considered by the other process because of the legal mandate of the act that the process serves. As a result, a highly disciplined project management framework is required to coordinate complex review interactions, address project risks and ultimately meet the review timelines for both processes. For each project, the primary project management tool to coordinate the reviews is called an *assessment plan*.

Each project's assessment plan, with the ultimate deliverables and timelines in mind, breaks down the entire review for both processes to "topical review" elements that can be delegated to Review Leads for that area of expertise. All review elements can be categorized into six groups, namely:

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[28] Cabinet Directive on Streamlining Regulation, Government of Canada, ISBN 978-0-662-49149-1

1. General Applicant Information.

2. Applicant Programs, Processes & Procedures.

3. Health & Safety (Workers & Public).

4. Baseline Site Data.

5. Effects of Environment on Project.

6. Effects of Project on Environment.

These are further broken down into smaller reviews; for example, one review sub-topic within "Baseline Site Data" is "Regional and Site Baseline Extreme and Rare Meteorological Phenomena".

The review scope and depth for each sub-topic is contained in internal CNSC Management System procedures call *Staff Review Procedures*. For the example provided above, Staff Review Procedure SRP-EIS-01.3 *Regional and Site Baseline Extreme and Rare Meteorological Phenomena* contains the overall guidance an assigned Review Lead will use when reviewing and commenting on submissions from the applicant. The procedures are not overly detailed and are designed to allow for "guided expert-based" reviews.

There are 27 topical Staff Review Procedures available for the conduct of the review of a *Licence to Prepare Site* application and 43 topical Staff Review Procedures available for the conduct of the review of an environmental assessment.

Integration of all reviews into recommendations to the Joint Review Panel is conducted by two project officers, one for the Environmental Assessment and one for the licence application with constant interaction between the two. Where insufficient information has been provided by the applicant, proposed information requests are submitted to the Joint Review Panel for consideration and transmission to the applicant. If the request is significant enough, the six month clock may be stopped by the Joint Review Panel for the period required by the applicant to gather the information for the response. This ensures adequate time will be available for review of the additional information provided.

The results of all reviews are integrated into two documents for consideration by the Joint Review Panel:

1. Panel Member Document for the Environmental Assessment:

   - Is the information submitted by the proponent adequate to proceed to a public hearing?
   - Are there adverse environmental impacts and if so are they adequately mitigated?

2. Commission Member Document for the Licence to Prepare Site:

   - Is the applicant qualified to carry on the activity that the licence will authorize?
   - Will the applicant, in carrying on that activity, make adequate provision for the protection of the environment, the health and safety of persons and the maintenance of national security and measures required to implement international obligations to which Canada has agreed?

The Joint Review Panel then holds public hearings to hear views about the Environmental Assessment and the application for a *Licence to Prepare Site* from the public, Aboriginal groups, the applicant and other stakeholders. After the public hearings and a deliberation period, the Joint Review Panel submits their report to the federal Minister of the Environment who in turn presents it to the Governor in Council. Approximately two months later, the Governor in Council responds whether the project may proceed or not and the Joint Review Panel renders a licensing decision.

References

[1] RD-346 Site Evaluation for New Nuclear Power Plants, September 2008 by the Canadian Nuclear Safety Commission through the Minister of Public Works and Government Services Canada, ISBN 978-1-100-10578-9.

[2] IAEA Safety Standards Series, Safety Requirements No. NS-R-3, Site Evaluation for Nuclear Installations, Vienna, International Atomic Energy Agency, 2003, ISBN 92-0-112403-1.

[3] Food and Agriculture Organization of the United Nations, International Atomic Energy Agency, International Labour Organization, OECD Nuclear Energy Agency, Pan American Health Organization, United Nations Office for the Co-Ordination of Humanitarian Affairs, World Health Organization, "Preparedness and Response to Nuclear or Radiological Emergency, Safety Requirements", Safety Standards Series No. GS-R-2, IAEA, Vienna, Austria, 2002.

[4] Health Canada, "Canadian Guidelines for Intervention during a Nuclear Emergency", Document H46-2/03-32E, Ottawa, Ontario, November 2003, ISBN: 0-662-35147-9.

[5] Cabinet Directive on Streamlining Regulation, Government of Canada, ISBN 978-0-662-49149-1

Potential Nuclear Power Plant Siting Issues in the United Arab Emirates

Waddah T. Al Hanai, Federal Authority for Nuclear regulation (FANR), UAE

Abstract

Based on the need to develop additional sources of electricity to meet future demand and to ensure the rapid growth of its economy, the United Arab Emirates has embarked on a nuclear programme. The Federal Law by Decree No. 6 of 2009, Concerning the Peaceful Uses of Nuclear Energy was signed by the President, last fall. This law created the Federal Authority for Nuclear Regulation (FANR), which is developing the framework of regulations which will guide the UAE programme.

This paper reviews the development of the FANR regulation on Siting and the related environmental issues in general and those unique to the area. This will include steps being planned by the Authority to review the licence application and the current concepts being looked at for the inspection programme. Among the unique aspects the author will look at are the results from a recent in-depth study performed on dust and sand storms.

Introduction

A decade-long high economic growth in the United Arab Emirates, triggered and sustained by massive urbanization and industrial projects, have resulted in a new thinking to meet the expanding demand for electricity in the country (Figure 1). Carbon-free energy sources, particularly nuclear, have become UAE's decision of choice. Almost two years ago, the UAE issued its "White Paper" that summarizes UAE's policy decision for peaceful nuclear energy (Ref. 1), and in September 2009 released its "Nuclear Law" (Ref. 2), creating the *Federal Authority for Nuclear Regulation* (FANR) and thus heralding the country's safety infrastructure preparatory work for construction of a nuclear power plant after the policy decision has been taken. This is a major undertaking requiring careful planning, preparation and investment in a sustainable infrastructure that will provide legal, regulatory, technological, human and industrial support to ensure that the planned nuclear power plants will be constructed and managed in a safe and secure manner.

This paper will review the development of the FANR regulation on siting of UAE's first-ever nuclear power plant and the related environmental issues in general and those specific to the area. This will include steps being planned by FANR to review the licence application and the current concepts being looked at for the inspection programme. Among the unique aspects the paper will look at are the results from a recent in-depth study performed on dust and sand storms.

Preparation for License Applications:

UAE's Regulatory Authority (FANR) will regulate the development of a nuclear facility or activity, as applicable, from initial selection of the site, through design, construction, commissioning, and operation, to decommissioning or closure, in full transparency and as per internationally recognized standards and recommendations, such as IAEA Safety Standards.

The development of Regulations and Guides by FANR involves:

- Issuing regulations to establish requirements with which all Operators must comply. Such regulations provide a framework for more detailed conditions and requirements to be taken into account in the design, construction and operation stages as well as requirements for the content of individual licenses applications.

- Issuing Guides, of a non-mandatory nature, on how to comply with the regulations. FANR issues guidance on the the way to comply with design construction requirements and the format and content of documents to be submitted by the Applicant in support of applications for license. The Applicant is required to submit or make available to FANR, in accordance with agreed time-scales, all pertinent information on data and methods to be used in assessing the license application or the adequacy of a design and on analyses and documentation to be submitted to the regulatory body by the operator.

In developing regulations and guides, FANR takes into consideration comments from stakeholders and the feedback of experience. Due account shall also be taken of internationally recognized standards and recommendations, such as IAEA Safety Standards.

Steps planned by FANR to review license applications

FANR's framework for licensing regulations includes:

- Regulation for Siting (Site Selection, Site Evaluation and Site Preparation) of Nuclear Facilities.

- Regulation for Application for a License to Construct a Nuclear Facility.

- Regulation for Application for a License to Operate a Nuclear Facility.

Objectives, management, planning and organizational matters related to the review and assessment process for license applications are entrusted to a Licensing Project Manager.

A primary basis for review and assessment of applications for license is the information submitted by the Applicant.

Principle of Licensing the Country of Origin Design

As an assumption of the program the UAE has decided on acquiring, an out-of-the-shelf NPP technology, FANR, in the construction license stage, will be requested to review what is known as the "Country-of-Origin" design. FANR's principles of licensing the Country-of-Origin design require that:

- The reference design proposed by the Applicant must have been approved by the Regulatory Body of the Country-of-Origin (RBCoO).

- The Preliminary Safety Analysis Report (PSAR) must provide detailed evidence that the design proposed for construction in the UAE complies with FANR requirements for safety, security, and safeguards.

- The PSAR must reference the design approval by the RBCoO and related safety documents.

- The PSAR must identify any differences between the reference design and the UAE design and must describe how their safety significance has been assessed.

A thorough review and assessment of the Applicant's technical submission will be performed by FANR in order to determine whether the proposed facility or activity complies with the relevant safety objectives, principles, and criteria. In doing this, FANR's objective is to satisfy itself that:

(a) The available information demonstrates the safety of the facility or proposed activity;

(b) The information contained in the Operator's submissions is accurate and sufficient to enable confirmation of compliance with regulatory requirements; and

(c) The technical solutions, and in particular any novel ones, have been proven or qualified either by competent authorities, experience, or testing, and are capable of achieving the required level of safety.

Following its regulatory review and assessment, FANR shall:

Grant a license which, if appropriate, imposes conditions or limitations on the Applicant's subsequent activities; or

Refuse a license.

The Regulatory Authority shall formally record the basis for these decisions. The Regulatory Authority (FANR) shall define and make available to the Applicant the principles and associated criteria on which FANR judgments and decisions are based.

Development of the FANR Regulation on Siting
In conformance with IAEA's NS-R-3, the objective of UAE's Regulation for siting of a nuclear facility is to establish the requirements for (Figure 2):

1. Site selection,
2. Site evaluation, and
3. Site preparation for the proposed Nuclear Facility.

The regulation for site evaluation requires full characterization of the site-specific conditions so that the Nuclear Facility is protected against external hazards and so that any health and environmental impacts that might arise from its operation are minimal. Additionally, the requirements of this regulation shall be considered in conjunction with the requirements of other related regulations issued by the Authority.

Steps planned by FANR to review the siting licence application

A siting license application (to implement the process of selecting a site) is required to conform with the nuclear law and FANR requirements and guidance (Figure 3). The review and assessment process of the siting license application is summarized in Figure 4. Once a Siting License Application is received, FANR will issue – if acceptable - an ***Acknowledgement of Receipt***. FANR staff will then review the *Contents* of the Application for completeness in terms of providing:

1. General information,

2. A description of the proposed activities. The description should be adequate to show that proposed activities will be carried out in conformance with the FANR regulation for siting,

3. A description of the Management Systems that will be applied to the activities. The description should be adequate to show that the management system applied to the proposed activities conforms with the IAEA Safety Standard GS-R-3.

If completeness is verified, the review and assessment process will go through:

1. Initial assessment.

2. Final assessment.

3. If requirements are met, issue License and attached conditions.

The review and assessment process is supported by FANR-established procedures for the review and assessment of the application. These are step-by-step procedures that provide internal guidance and instructions to be followed by the FANR staff in the review and assessment process and guidance on the safety objectives and minimum acceptance criteria to be met. Detailed guidance on specific topics for review and assessment are to be provided, if deemed necessary. That is, the FANR review and assessment efforts will be focused more on those aspects of site evaluation that constitute site-specific characteristics or involve untested (innovative) features, to determine whether or not the applicable safety objectives and regulatory requirements have been met for each aspect or topic. To that effect, FANR review and assessment will take into consideration the results of external peer reviews conducted by national or international organizations on behalf of the Operator. The results of such reviews could provide FANR with additional insights into the activities of the Operator.

Inspection is an integral part of the FANR review and assessment of a siting license application. To this effect, FANR conducted a site visit and performed a site-investigation program review in order to understand the ongoing site-selection activities aimed at choosing the preferred site and to anticipate any further issues. Additionally, an inspection program is under formulation to oversee the site-preparation activities.

Related environmental issues

Dense fog, dust and sand storms, high ambient air temperatures, high cooling water temperature, and sabkha (or ***evaporites***) ground soil are noticeable characteristics of the region of the proposed NPP of the UAE. Dense fog and severe “dust/sand” storms are perhaps the two most site-specific and most safety-relevant meteorological phenomena.

“Dust/sand” storm (Figure 5) refers to two distinct meteorological phenomena of airborne-particles that need to be accounted for in the design of a nuclear power plant (NPP) in the United Arab Emirates.

“Dust” storm refers to a low-frequency, external, or “synoptic” meteorological event, characterized by a huge mass of air that carries small dust particles at low-speed and high altitude over a meteorologically large scale.

In contrast, “sandstorm” refers to a local and more frequent, wind phenomenon that displaces huge masses of sand particles (with a wider spectrum of size) at high speed but over a short distance.

The NPP should cope with both “dust” and “sand” storms.

The FANR/IRSN methodological study (Ref. 3) is aimed at identifying and defining the main sensitive issues that the licensee of a NPP should take into account in the dust/sandstorm characterization and its safety impact assessment.

Three main steps are identified in the study:

- The collection and analysis of the characteristics of dust/sandstorm (description, intensity, duration, physical particles displaced…),

- The investigation of the potential negative effects of dust/sandstorm on the Structures, Systems and Components (SSCs) of the NPP, in particular those pertaining to the safety functions,

- The identification of the main sensitive issues that should be addressed in the licensee study related to the evaluation of the protection of the NPP against dust/sandstorms.

The main input material for investigation has been the information collected during a Forum organized by FANR in mid 2009 with some UAE organizations and industries to the effect of collating their feedback, experience and knowledge regarding dust/sandstorms. Only limited literature exists on the characterization of UAE sand/dust storms. Eck *et al* (Ref. 4) identified limited observational data and large differences between data sets of measured particle-size distribution, despite reasonable consistency within a particular data set.

The characteristic parameters of a “sand” or “dust” storm are: dust/sand particle size distribution, particle shape, particle morphology, mineralogy and physical chemistry, wind speed, height, particle concentration above surface, duration of the event. These parameters are still only qualitatively described in the literature; perhaps because of lack of measured data in the world and the existence of gaps and constraints in instrumentation and observation regarding particle-size distributions of airborne dust (Ref. 5).

In addition to the hazard to health, the main potentially negative effects of dust/sandstorms on a NPP could be corrosion, abrasion, plugging, and clogging of filters, electric/electronic circuits and rotating parts of the systems, structures and components (SSCs) of a NPP: openings, sealing, HVAC systems, filters, off-site power sources/diesel generators, ultimate heat sink, electric/electronic devices and instrumentation, pipes/ducts/tanks, structures, drains, and exhausts. Quantitative specifications of the measures to be taken for the design, protection, test and maintenance of this list of SSCs still have to be selected from existing standards or developed.

A rigorous treatment of dust/sand grains as potential carriers of radioactive gases and radioactive aerosols (after a radioactive discharge accident in a NPP) may have to be developed.

Such a treatment should include the correlation of the effects of dust/sandstorm in the presence of fog and humidity. In such a coincidental event, the dust/sand grains are most likely coated with a thin water film, which will either deposit leading to the risk of "sand caking" on external SSCs, or enhance the adsorption of discharged radioactive nuclides and their transportation further than when there is no humidity or fog in the air.

The same rigorous treatment should be extended to study and quantify the risk of generating electrostatics through sandstorms, and if proven to be high, the effects of such electrostatics on electrical/electronic SSCs should be determined.

In conclusion, a methodology has to be designed to evaluate the risk of dust/sandstorms on a nuclear power plant and to address this risk at the design stage and define the corresponding protection means and provisions. The proposed methodology should consist of:

- Parametric characterization of dust/sandstorms,
- Identification of the plant vulnerabilities to the event,
- Identification of the structures, systems and components (SSCs) to be protected. This identification requires preliminary definition of the combinatory interaction with other independent internal and external events (fog, humidity, salt content in the air and in surrounding sabkha, tidal phenomena, temperature, electrostatics, …) as well as induced events to be considered in the design;
    - Definition, design and implementation of effective and efficient safety and security measures,
    - Evaluation of the sufficiency of the protective measures and upgrading them accordingly,
    - Examination of radiological aspects.

Conclusion

The United Arab Emirates have embarked on a nuclear power programme in order to meet future demand for electricity needed to sustain the rapid growth of the economy.The UAE's Federal Law by Decree No. 6 of 2009, Concerning the Peaceful Uses of Nuclear Energy, created the Federal Authority for Nuclear Regulation (FANR), which is developing the framework of regulations that will guide the UAE programme.This paper reviewed the development of the FANR regulations on siting of UAE's first-ever nuclear power plant and the challenges imposed by related environmental issues specific to the area. Dense fog and severe "dust/sand" storms are believed to be the two most site-specific and potentially most safety-relevant meteorological phenomena in the region of the proposed NPP of the UAE. The main negative effects of dust/sandstorms could be corrosion, abrasion, plugging, and clogging of filters, electric/electronic circuits and rotating parts of the systems, structures and components (SSCs) of the NPP.

The large experience in other industrial sectors shall be capitalized on in order to prevent any adverse situation.

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Figure 1: Expected Evolution of Electrical Power Demand and Supply in the UAE

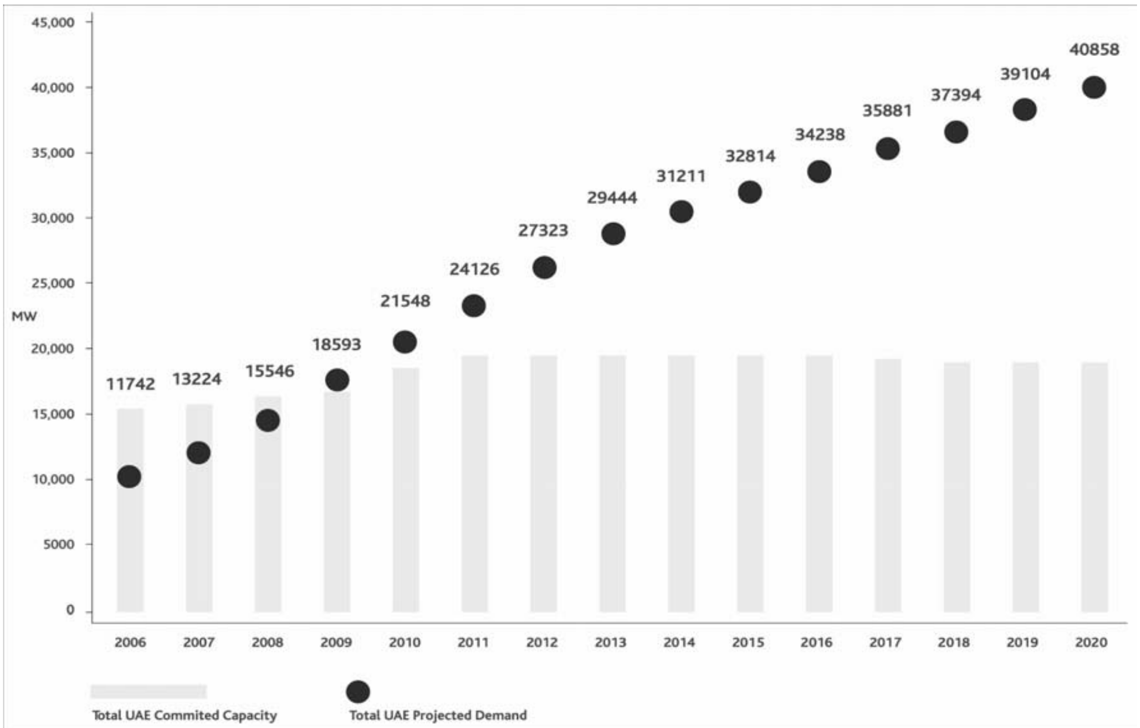


Figure 2: Licensing Timeline

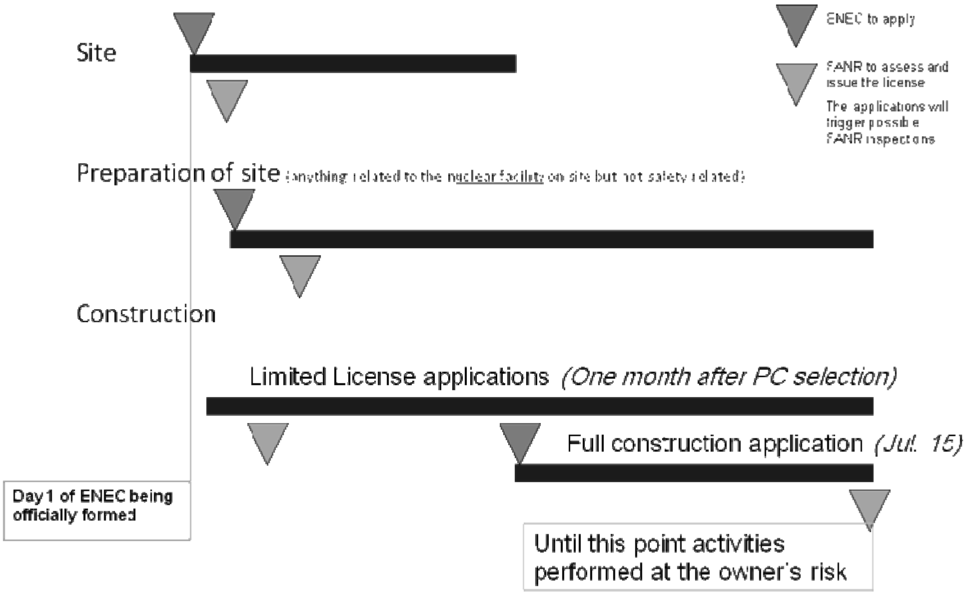


Figure 3: The Pyramid of Hierarchical Regulatory Levels. An Applicant will submit a License application in conformance with the nuclear law and FANR requirements and guidance

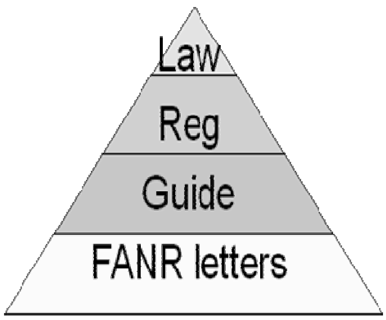
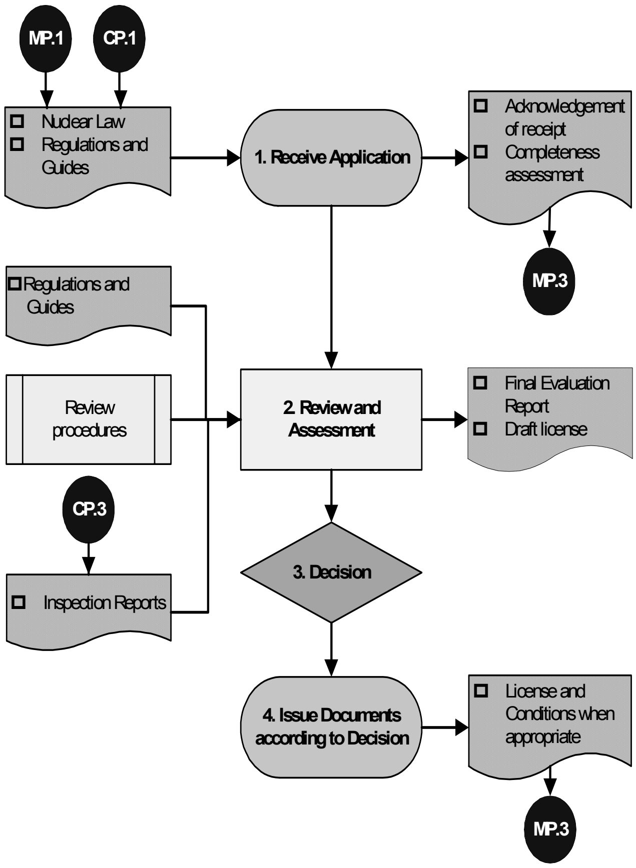


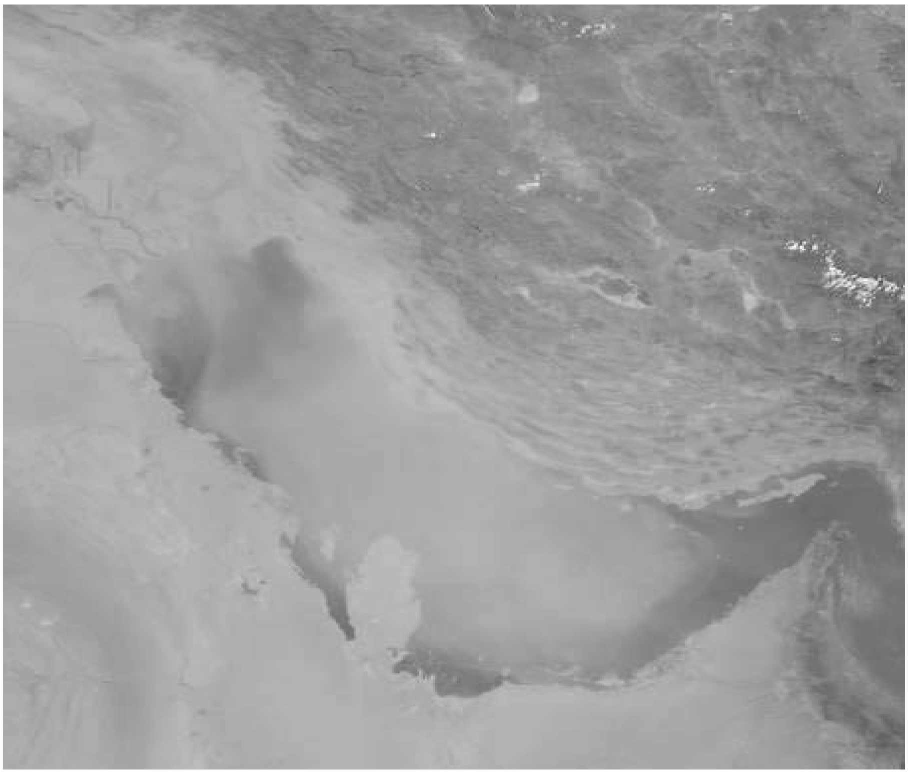
Figure 4: Licensing Flow Diagram (CP.1 = Core Process "Manage the Regulatory Framework for Ensuring Safeguards, Safety and Security"; CP.3 = Core Process "Assure Compliance"; MP.1 = Management Process "Direct and Manage the Organization"; MP.3 = Management Process "Manage Corporate Communication")



1. FANR receives Application, issues communication to stakeholders
2. FANR conducts Review and Assessment of the Application
3. Board decision to grant or refuse a License
4. FANR issues documents according to Board decision, issues communication to stakeholders

Figure 5: A: Dust storm moving towards UAE, B: Sandstorm in UAE

A



B

NPP Siting in Western Part of Java Island Indonesia: Regional Analysis Stage

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Yuliastuti, National Nuclear Energy Agency (BATAN), Indonesia

Extended Summary

Considering that Banten and West Java Provinces are dense regions of industry, therefore they require a large amount of electricity. Nuclear power plant is one option to be considered to anticipate the future electricity demand. To support the program, it is needed to look for some potential locations through NPP siting. The siting should meet the requirement of safety, safety aspects of the natural external events, human induced external events, public and environmental safety. Site selection is performed in several stages, where each stage has specific assessment criteria.

Siting is commenced with pre-survey activity to obtain several interest areas, the activity covers a wide area but the used data is very limited and only apply general criteria. The following activities after pre survey are site survey consisting of (1) regional analysis, (2) site screening, and (3) comparison and ranking stages. The objective of regional analysis is to obtain potential sites in the study area of 150 km radius from each interest area by using both general and specific criteria. The potential sites then screened to obtain selected candidate sites by using more detailed secondary data as well as survey activities such as geophysical investigation, a few of drilling, etc., within the radius of 50 km from each potential site. All the selected candidate sites are then compared and ranked to obtain preferred candidate site. Site evaluation is the next step to evaluate all site-specific parameter to obtain design basis parameters and as the basis for preparing site permit document. This paper presents the methodology and result of regional analysis stage. The objective of the activity is to obtain potential sites in the north coast of West Java and Banten Provinces by considering fourteen study aspects which could be categorize into safety related aspects, non-safety related aspect and public education. However, this paper only considers the safety related aspect specifically external event aspects.

Furthermore, external event aspect divided into natural and human induced aspects. Natural external event aspect consists of surface faulting, seismicity, subsurface material, volcanism, coastal flooding, river flooding, and other external potential hazard. Meanwhile, human induced external event aspect consists of stationery source item such as airport, harbor, military facility, chemical industries, etc, and mobile source item for example air corridor, fairway, gas/oil pipeline, road/highway network, railway, etc.
Regional analysis is commenced by assessing each aspect or known as topical analysis. Each topical analysis is then integrated by overlaying one aspect to another and result an Integrated Regional Analysis (IRA). Combining the hazard zone of natural and human induced external event would produce a free-hazard zone. Furthermore, it can be defined and delineated to 2 (two) potential sites namely Kramatwatu-Bojonegara and Pulau Panjang.

However, those potential sites to become a selected candidate site as the results of near regional analysis stage, need further investigation and some field works in order to achieve appropriate data, especially on verification of supposed capable faults and deep investigation of capable volcanoes in the radius of 50 km from the site.

Temelin 3,4 Siting

Iva Kubanova, CEZ, a. s. , Czech Republic

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1. Energetical situation in the Czech republic

In the future the Czech Republic will need new energetical resources in spite of current decrease of electricity consumption due to economical crisis. Nuclear power generation is considered as important part of energetical mix of the Czech Republic and this opinion is newly reflected in new government official statement issued in August 2010.

2. Nuclear projects of CEZ, a. s.

CEZ, a. s. prepares new nuclear power plants projects accordingly governmental expectations. Currently 3 projects are in preparation. Temelin 3, 4 project is in the most advanced status, tender is in progress. Potential construction of Dukovany unit 5 and new Jaslovske Bohunice units are analyzed in feasibility studies.

3. Temelin 3,4 Project Status

Temelin 3, 4 project activities were started 4 years ago. Preparatory analyses, market investigation, feasibility study including many particular studies were elaborated in years 2006 – 2008. Later on decision to work on bid invitation specification was done and followed. EIA process was started in July 2008 by Intension Announcement and continues. Public tender for EPC contract was announced in August 2009 and it is in progress accordingly schedule. Siting process is in the initial stage.

4. EIA Process

EIA process started in July 2008 through Intension Announcement in spite of anti – nuclear political climate in the Czech Republic. EIA process is interstate process, Austria and Germany participate. Investigation Process Protocol was issued by Ministry of Environment in February 2009 with 34 main conditions and 165 additional comments, requirements, statements. CEZ, a. s. adopted the positive approach with philosophy to deal with all conditions and requirements properly. Elaboration of EIA documentation took 18 months. In May 2010

CEZ, a. s. handed over the EIA documentation to the Ministry of Environment and consequently all legal steps followed including handover of EIA documentation to Austria and Germany. In next weeks and months all comments will be gathered by Ministry of Environment and relevant decisions and legal steps will follow.

5. Siting Process

Siting process is three - step process in the Czech Republic. Finished EIA process managed by Ministry of Environment is a prerequisite for siting approval issued by State Office for Nuclear Safety. Then the final territorial decision is issued by building (construction) authority which can be located either in the relevant close city or in Southern Bohemia Region Office. Temelin site is a proven site, so siting of Temelin 3,4 is focused mainly on verification of available site data and methods used, verification of site suitability for new generation reactors and application of current licensing requirements. As a basis Initial Safety Report must be prepared and covers topics prescribed by the Czech Atomic Law : general project information, site evaluation, technical concept description, preliminary evaluation of operation impact on population, environment (with reference to EIA process), future decommissioning method, physical protection analyses and siting quality assurance evaluation and principles for next stages. CEZ, a. s. currently prepares Initial Safety Report in format pre - agreed with State Office for Nuclear Safety.

Near Regional and Site Investigations of the Temelín NPP Site

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The Temelin NPP is worldwide through heated discussion with nuclear energetic opposition. In addition this discussion goes beyond a border of the Czech Republic. On the other side, results of several international supervisions shown that Temelin NPP is fully comparable with the safest nuclear power plants in the world regarding its technical design and safety functions.

1. New Temelin units siting

We can see that nuclear power is experiencing a renaissance all over the world, including the Czech Republic. Temelin NPP was originally designed to have four nuclear units. Now only two are operated, therefore a fulfillment of the original design is a logic step to optimize usage of the existing infrastructure and area. Energetic experts suppose that the completion of the Temelin NPP ensures reliable covering of the growing electricity consumption in the Czech Republic in the future and creates sufficient reserves.

Siting of the 3rd and 4th unit of the "New nuclear source" at the Temelin site has been running several years in form of feasibility studies. Siting works fully respect the existence of two units in operation and SNF (spent nuclear fuel) storage during the construction period. In spite of location very near to the existing nuclear installations Temelin site has many advantages:

- Known place from the point of view very good knowledge of geology, hydrogeology, geotechnics, safety hazards, social and environmental impacts etc.
- Existence of the free building plot owned by the Power Company and infrastructure, which can be used.
- Good public acceptance and political backing.

We are also facing some problems such as obdurate nuclear energetic opposition and demanding cross-border discussions.

2. Approach to Temelin new units siting

The following principles were considered to the siting process:

- Strict fulfilling of the IAEA recommendations and other regulations (national legislation).

- Combined action for new units siting and verification of existing units.

All knowledge, experiences and data from the existing Temelin NPP siting exploiting have been gathered with the application of the „up to date“ procedures and survey methods. Professionalism, profundity and high quality of the work has been kept. An important task is accommodating public relations cared by stakeholder.

3. Organization of siting works

Siting works are organized in cooperation with three main partners: stakeholder – expert team – regulatory body. Two departments of CEZ a.s. “Department of Engineering of 1th, 2nd Unit” and “Nuclear Power Plants Construction Analysis of 3rd and 4th Unit Department” were appointed by the stakeholder (CEZ a.s.) to organize siting works (to manage project, to determine project excavated goals, to derive preferences, to conclude contracts, to ensure the quality of works). Expert team consists of specialists from private companies, universities and research institutes. Experts ensure all research and survey work and they will defend a result of their work before laic and academic community. Revisory function will be provided by the regulatory body - State Office for Nuclear Safety (SÚJB).

4. Our important tasks

Although the Temelin site is situated in the area with low seismicity, item of seismicity is a basic argument against Temelin NPP mentioned by opposition parties. Therefore a seismic evaluation represents significant priority. Siting works are focused, in compliance with IAEA Safety Guide NS-G-3.3, on four issues: construction of seismotectonic model of the Temelin region, monitoring of micro earthquakes in near region, near regional investigations (especially of faults), and site area geotechnical surveys.

Main branches of our interest in the field of **regional assessment** are:

- Implementation of the new results of scientific research into FSAR (Final Safety Analysis Report).
- Revision and completion of the near regional and regional earthquake catalogue.
- Evaluation of seismic waves attenuation.
- Historical seismology studies.
- Assessment of strong earthquakes in Temelin NPP region.
- Input data collection for seismic hazard assessment.

Surveys of the tectonics in the vicinity of Temelin NPP were performed in the course of the years 1992 - 1994 under the supervision of the IAEA missions, always with positive evaluation. Since that time the research methods have been developed and significant international programs have been completed. Therefore we are aware of the necessity to perform new investigations by modern methods. Resolution of the Czech Austrian parliamentarian commission in December 2007 only pushed forward a decision to perform such new investigations.

Near regional investigations are performed by two parallel projects at this moment. The first project, supported by the State Office for the Nuclear Safety on the basis of agreement of the Czech Austrian parliamentarian commission, is focused to research the Hluboká fault. The Czech and Austrian experts participate in this project and jointly interpret survey results. The actual progress of the works is the performance of the field works – drilling surveys and trenching. The second project surveys the potential seismic a movement activity of Blanice gap fault.

The investigation plan was built-up to respect requirements and recommendations of the IAEA Standards, especially NS-R-3 and NS-G-3.3. Newly a paleoseismological method will be used in the near region of Temelin NPP.

Our activities are focused on the following issues in the field of **near regional investigations**:

- Evaluation of near regional faults and assessment of potential of its movement and seismic activity – especially Hluboká fault and Blanice gap fault.
- Research of Vltava river terrace system in the near region.
- Research of the faults using of paleoseismological method.
- Looking for evidences of recent seismic activity of near regional faults.
- Input data collecting for seismic hazard assessment.

Site area has been surveyed thoroughly during several periods of engineering - geological surveys for designed construction of 4 units VVER 1000. The last surveys were performed during 1980ies (1983-1989).

Geotechnical surveys of the nuclear power plant Temelin were divided to the phase of introductory engineering-geotechnical survey and 1st and 2nd phase of the preliminary resp. detail evaluation of construction-geological characteristics of the main construction site. The purpose of the 3rd phase - supplementary survey was to evaluate thoroughly bottom layer characteristics in the area of the main objects, such as reactor building, machinery building and cooling towers.

Pivotal surveying works at the nuclear power plant Temelin (1st and 2nd phase of surveys) included the following technical activities: deep drills for field test and measuring, shallow drills for engineering-geological evaluation, hydrogeological observation wells, test pits and probe trenches. Both phases included in total 66 core drills for field measurements (2 572.6 running meters), 315 test core drills (6 461.8 running meters), 8 observation wells (380.0 running meters), 24 supplementary test bores (162.0 running meters), 69 test pits (293.6 common meters) and 7 trenches (21 849 m^3). There were at the nuclear power plant Temelin constructions site (apart from a trench) 488 core drills excavated of the total footage 9 870 running meters.

Verification of physical-mechanical properties of solid rock resp. semirock basement focused on field tests. Logging, microseismic logging, presiometric measurements and load tests have been performed in drills and trenches. Taken samples of semi-disturbed soils or rocks have been tested in a laboratory. Survey works have been evaluated on standard bases with the application of available up-to-date computer programs and mathematic - statistic procedures.

Description, evaluation, maps etc. are not available in soft version. Nevertheless this documentation is considered as a large source of knowledge for easy application.

New site area investigations are focused especially to upgrade geotechnical data. Investigations started in year 2008 by collating and digitizing of the crucial geological knowledge about the construction site Temelin. The "Database of surveying objects" was this task result. The investigation plan after this data inventory was focused on the following:

- Investigation of unsafe places of the building place.
- Adding of the missing geotechnical data.
- Harmonizing of the data (mainly old data) with Guide recommendations e.g. IAEA Safety Guide NS-G-3.6.
- General geotechnical characterization of the building place.
- Full settlement of relevant geotechnical criteria of siting.

5. Seismic hazard assessment

The project of seismic hazard re-evaluation of the Temelin site was started in 2008. Now seismic experts are working on modern methodology of seismic hazard assessment conformable with Draft Safety Guide DS422. This methodology exploits results of international geological and seismological projects and it will be based especially on the probabilistic safety assessment approach. This document will be completed this year.

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Regulatory Issues and Challenges in Preparing for the Regulation of New Reactor Siting: Malaysia's Experience[29]

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1. Introduction

This paper aims at giving an overview about the issues and challenges facing regulatory authority, the Atomic Energy Licensing Board (AELB) in ensuring nuclear safety, security and safeguards (3S's) and other relevant authorities to meet an impending nuclear power programme post-2020, in particular at the beginning stage of preparation for the regulation of nuclear power reactor siting. A comparison with an international framework and guidelines of the International Atomic Energy Agency (IAEA) and other countries' practice was made to get an overview of the present adequacy of Malaysia's nuclear regulatory framework in preparation for Malaysia to consider and perhaps decide for a safe, secure and peaceful nuclear power project in Malaysia, in utilising nuclear power in a quest for energy diversity and security.

2. Malaysia's regulatory framework for NPP

Since February 1985, atomic energy activities in Malaysia are under the oversight of AELB with the authority given to it by the Atomic Energy Licensing Act, 1984 (Act 304). AELB's regulatory mandate was to ensure public and radiation workers health and safety including the safety of environment and property, security and the peaceful use of atomic energy activities, without imposing excessive requirements that might inhibit the growth of the industry. The AELB is the sole national regulator for all atomic energy activities in Malaysia including radioactive materials, industrial and medical applications, waste disposal facility and nuclear installations such as nuclear reactors. However, this includes utilizing input from other relevant authorities such as the Energy Commission (EC) on the Electricity Generation and Distributions (Electricity Supply Act, 1990) (Act 447), Department of Environment (DOE) on Environmental Impacts Assessment (EIA) (Environmental Quality Act, 1974 (Act 127), Department of Occupational Health (DOSH) on Occupational Health and Safety of the workers (Occupational Safety & Health Act 1994) (Act 514), the local authorities on Social Impact Assessment (SIA) (Town & county Planning Act 1976)(Act 172), public participation and the like, which all provides a collective approach and foundation for regulatory decisions.

As early initiatives taken towards the launching of a nuclear power project in Malaysia, AELB have identified several steps to be taken, in the effort to strengthen national legal and nuclear regulatory infrastructure. The Atomic Energy Licensing Act, 1984 (Act 304) is currently being reviewed in order to fulfil national needs and international requirement and practices such to include, *inter alia*, a strengthened

[29] CNRA International Workshop on "New Reactor Siting, Licensing and Construction Experience"; April 21th -23rd, 2010, Prague, Czech Republic

nuclear liability regime non-proliferation measures. As a result, there is also a possibility to made consequential amendments to other relevant act.

3. Regulatory issues and challenges

3.1 Licensing Preparation for a nuclear Power Plant in Malaysia

It is important to appreciate the value of independence and technically strong regulator, particularly in assuring the public that a nuclear power plant will be operated safely and securely, should ever Malaysia decides to embark on a programme to generate nuclear electricity. The level of public assurance will depend on the regulators for being a fair but firm regulator to ensure a high level safety standards and security are met, in addition to its peaceful uses. However, this shall be done in environment that strives to be as open and transparent as possible.

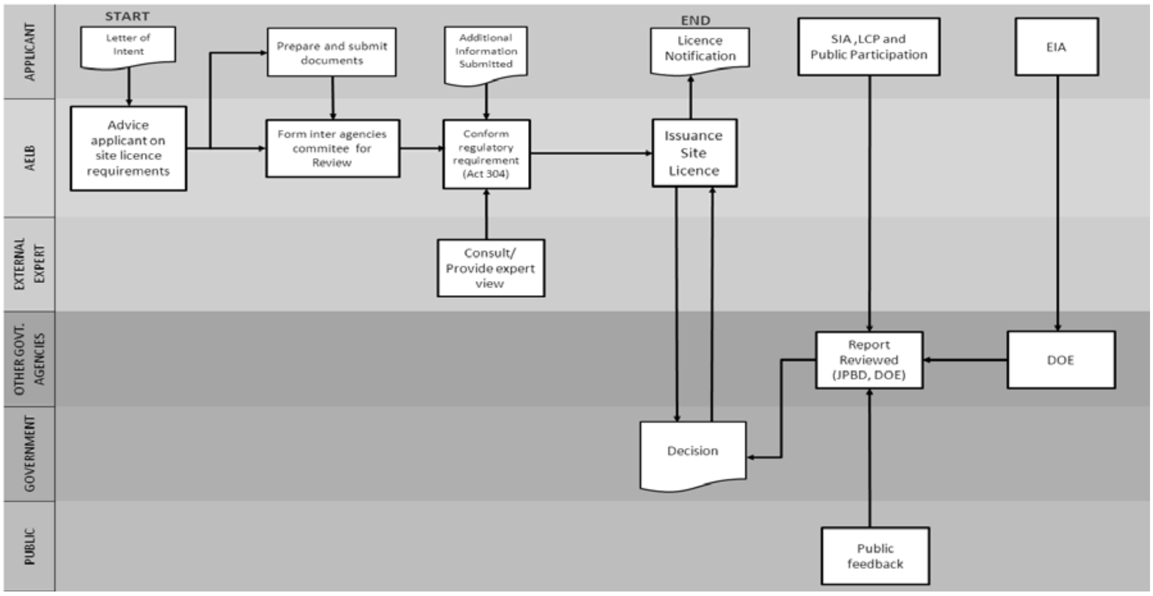
In an effort to eliminate potential regulatory risks for licensing, construction and ultimately the operation of a any nuclear installations, including a power plant in Malaysia, AELB is preparing new regulations on nuclear installation licensing, in addition to the Radiation Protection (Licensing) Regulation 1986, that by the way, is also currently being revised. This Regulation was initially developed almost 25 years ago, this process, which we will using it for the first time including for a for nuclear power plant if Malaysia decided to embark upon a nuclear power programme, involves “site licensing”, “construction licensing” and “operation licensing”.

In reviewing a site license application, AELB considers site safety issues, environmental protection issues, security measures and plans for coping with emergencies, and these independent of the review of a specific nuclear plant design. AELB will have to review the site evaluation report including characteristics of the site, its surrounding population, seismology, meteorology, geology and hydrology. During this process, it is believed that AELB through Radiological Impact Assessment (RIA), Department of Environment through Environmental Impact Assessment (EIA) reports and local authorities’ requirements including Social Impacts Assessment (SIA) will ensure public and other stakeholder involvement through public comments of the these reports before the issuance of a site license is considered. Figure 1 shows a possible licensing process for issuance a nuclear power plant’s site licence considering involvement of relevant parties mentioned above. Site license will be followed by construction and operation licence.

3.2 Proposed regulatory criteria for NPP’s site evaluation

Site selection is not regulated under the Atomic Energy Licensing Act, 1984 and this therefore not addressed in this paper. It is acknowledge that an important stage in the development of a nuclear power project is the evaluation of a suitable site to establish the site-related design inputs for the NPP. Therefore, AELB is in the process of developing new guidelines on the Nuclear Power Plant’s site evaluation.

Figure 1: A possible licensing process for a Site Licence



The greatest challenges facing AELB and other authorities may involved is to develop the guidelines and criteria to ensure the selected site is considered acceptable from the safety and security point of view with the application of appropriate exclusionary, avoidance and suitability criteria taking into consideration the following four (4) main aspects:

a) **Health and Safety**: geology/seismology related hazard (potential ground motion, soil stability, surface faulting and deformation, capable tectonic structure), atmospheric condition, cooling water requirement (water quality and availability), flooding (due to seismically, water control structure, river, coastal and wind generated induced flood), nearby existing and projected hazardous facility (military bases, refineries, dock and anchorage), population (density and distribution); potential radionuclide pathway (air, surface and underground water, during transport);
b) **Environment protection**: protection of area within *Environmental Sensitive Area,* disruption of protected habitat and endangered species in aquatic and terrestrial ecology, protection of marine park, wetland, forest including environmental and radiological impact assessment;
c) **Land use and Socioeconomics**: development of area within National Physical Plan, local and structure plan, consideration of existing and projected site vicinity development, aesthetic factor, proximity to public amenity;
d) **Public Participation:** to consult with general public and stakeholders early in the site evaluation process and to consider social impacts assessment before any substantive decision are made.

3.3 Other challenges

Future work projections indicate that AELB and other relevant authorities need more trained human resources, but many factors limit the ability to rapidly increase these human capital resources required e.g. financial constraints, an unclear government policy on NPP. It has been estimated by the IAEA that the envisaged regulatory authority requires at least 40-50 persons especially for regulating and supervising a nuclear power plant. Within AELB for instance, approximately 5 % of AELB resources will be in eligible for mandatory retirement in the next 10-15 years. Another area of future challenges are related to developing licensing framework and expertise necessary for reviews of the new advanced nuclear technologies and spent fuel and nuclear waste management regulatory policy in Malaysia.

4. Conclusion

The important stage in the development of a nuclear power project is the evaluation of a suitable site to establish the site-related design inputs for the NPP. The evaluation of suitable site is the result of a process to ensure adequate protection of workers, public and the environment from the undue risk of ionizing radiation arising from NPP taking into account impact to the social communities and public acceptance, thus it will depend on the regulators to ensure a high level safety standards and security are met, in addition to its peaceful uses. Development of regulatory criteria for the site evaluation is a pre-initiatives licensing work for a possible nuclear power plant to performing effective nuclear safety and security reviews in an efficient and timely manner regardless whether Malaysia embarks on a nuclear power programme with anticipating challenges, learning from others' experiences in preparing for the demands for new licensing processes by collaborating internationally, in an expanding global environment.

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