SAFETY, LICENSING AND REGULATION OF INNOVATIVE REACTORS

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Chairpersons

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Abstract

The paper addresses a general approach for the preparation of the design safety requirements using the IAEA Safety Objectives and the strategy of defence in depth. It proposes a general method (top-down approach) to prepare safety requirements for a specific kind of small or medium size reactor using the IAEA requirements for Nuclear Power Plants as starting point through a critical interpretation and application of the strategy of defence in depth. The method is general and can be applied to any kind of reactor. An activity is being carried out at the IAEA to prepare the technical basis for the development of safety requirements for Modular High Temperature Gas Reactors in order to show the viability of the method.

1. INTRODUCTION

There is a large variety of proposed Small and Medium Sized Reactors (SMRs) that use different design approaches, technologies and safety features. The safety assessment and the licensing of these reactors may require specific considerations and the current safety requirements may need, in some cases, interpretation or adaptation. It is expected that SMRs, apart from electricity generation, will also be used for non-electrical applications such as desalination and district heating. This requires the siting of the reactors in the vicinity of populated zones with additional implications on safety, licensing and acceptability.

The SMRs to be built in the future will have to meet at least the level of safety required for the best nuclear power plants currently in operation or being designed. It is expected that the safety will be even more transparent and easier to prove. The expected level of safety will also be independent of the particular type of reactor or technology.

The technical characteristics and safety features of some SMRs indicate that the full application of existing safety requirements, mostly developed for large water cooled reactors, would not be completely appropriate. There is the need to develop a tailored set of safety requirements derived from the general consolidated principles of nuclear safety that better incorporates the specific characteristic of SMRs. The IAEA Safety Standards and the work on implementation of defence in depth for different type of reactors provide a useful starting point and a suitable framework for this purpose.

2. GENERAL SAFETY ASPECTS OF NUCLEAR POWER PLANTS

There are three safety objectives from which all safety principles and requirements are derived:

**General Nuclear Safety Objective:** To protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards.
Radiation Protection Objective: To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents.

Technical Safety Objective: To take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

The safety objectives shall be achieved through the application of the defence in depth strategy [1] that will continue to be the overriding approach for ensuring the safety of workers and the public, and for protecting the environment. This strategy has been proved to be effective in compensating for human and equipment failures, both potential and actual. The concept is based on several levels of protection, including successive barriers that prevent the release of radioactive material to the environment. However, its efficacy depends on rigorous implementation.

TABLE 1, LEVELS OF DEFENCE IN DEPTH (FROM INSAG-10)

<table>
<thead>
<tr>
<th>Levels of defence</th>
<th>Objective</th>
<th>Essential means</th>
</tr>
</thead>
<tbody>
<tr>
<td>Level 1</td>
<td>Prevention of abnormal operation and failures</td>
<td>Conservative design and high quality in construction and operation</td>
</tr>
<tr>
<td>Level 2</td>
<td>Control of abnormal operation and detection of failures</td>
<td>Control, limiting and protection systems and other surveillance features</td>
</tr>
<tr>
<td>Level 3</td>
<td>Control of accidents within the design basis</td>
<td>Engineered safety features and accident procedures</td>
</tr>
<tr>
<td>Level 4</td>
<td>Control of severe plant conditions including prevention of accident progression and mitigation of the consequences of severe accidents</td>
<td>Complementary measures and accident management</td>
</tr>
<tr>
<td>Level 5</td>
<td>Mitigation of radiological consequences of significant releases of radioactive materials</td>
<td>Off-site emergency response</td>
</tr>
</tbody>
</table>

This implies a determined effort to make the defence effective at each level, particularly for accident prevention and accident mitigation. There is not a unique way to implement defence in depth, since there are different designs, different safety requirements in different countries, different technical solutions and varying management or cultural approaches. Nevertheless, the strategy represents the best general framework to achieve safety for nuclear power plants. In general, strong implementation of defence in depth requires a determined and constant effort from the design phase, to construction and operation in order to provide graded protection against a wide variety of transients, abnormal occurrences and accidents, including human error and equipment failures within the plant, and events initiated outside the plant.
FIG. 1. Logic flow diagram of defence in depth

Figure 1. shows in detail how defence in depth works in practice. The objective is always to maintain the plant in a state where the fundamental safety functions (confinement of radioactive products, control of reactivity and heat removal) are successfully fulfilled. The success criteria are defined for each level of defence in depth.

The possible challenges to the safety functions are dealt with by the provisions (inherent characteristics, safety margins, systems, procedures) of a given level of defence. All mechanisms that can challenge the successful achievement of the safety functions are identified for each level of defence. These mechanisms are used to determine the set of initiating events that encompass the possible initiations of sequences. The initiating events are, in general, the same for all the levels of defence. According to the philosophy of defence in depth, if the evolution of a sequence is not controlled by the provisions of a level of defence it will be by the subsequent level that comes into play.

2.1. Current safety approach

Operating nuclear plants are largely designed following the defence in depth strategy. This means that the plant is deterministically designed against a set of normal and accident situations according to well established design criteria in order to meet the radiological targets. The current design approach has been shown to be a sound foundation for the safety and protection of public health, in particular because of its broad scope of accident sequence considerations, and because of its many conservative assumptions which have the effect of introducing highly conservative margins into the design that, in reality, give the plant the
capability of dealing with a large variety of sequences, even beyond those included in the design basis.

The deterministic approach is complemented by probabilistic evaluations with the main purpose of verifying that the design is well balanced and there are not weak areas or systems which could allow for the possibility of risky sequences. Probabilistic safety assessment is recognized as a very efficient tool for identifying those sequences and plant vulnerabilities that require specific preventive or mitigative design features.

This safety approach is reflected in the existing IAEA Safety Standards for the design.

3. THE IAEA SAFETY STANDARDS SERIES

The Agency produces many documents related to nuclear safety, the most important of which are those now included in the Safety Standards Series (SSS), formerly the Safety Series, which included the NUSS programme.

![Hierarchy of the IAEA Safety Standards Series](image)

An extensive process was established few years ago to review all NUSS publication to produce a better organised and consistent set of documents. The SSS comprises three levels: Fundamentals, Requirements and Safety Guides. Several of these publications have been released and are available. They include the Safety Fundamentals, three Requirements (Legal and Governmental Infrastructures, Design [2] and Operation) and some Safety Guides. The remaining publications are still in a draft form at different grades of preparation.

3.1. Safety Fundamentals (SFs)

Currently, there are three SF documents, but they will be soon combined into a single document. SF are the first documents in the hierarchy; they present basic objectives, concepts and principles to ensure safety in the development and application of atomic energy or radioactive material for peaceful purposes. The SF documents constitute the reasons why activities must fulfil certain requirements; they do not state what these requirements are, they are self-sufficient and do not include a list of references. In the SF on Safety of Nuclear Installations (SS-110) there are 25 fundamental principles grouped into four main areas, related to the Legislative and Regulatory Framework, the Management of Safety, the Technical Aspects of Safety and the Verification of Safety.
3.2. Safety Requirements (SRs) and Safety Guides (SGs)

Supporting the SFs are Requirements (formerly termed Codes, Standards or Regulations). In the nuclear safety area there will be six main areas: Legal and Governmental Infrastructures, Siting, Design and Operation of water cooled nuclear power plants, Quality Assurance and the Research Reactor Series which has two SR documents. The SRs set out in more detail what is required by Member States to ensure safety in a particular area, and they are governed by the content of the SFs. SRs do not generally present recommendations on or explanations of how to meet the requirements. This more detailed aspect is covered by the third level in the hierarchy, namely, the Safety Guides. The SGs present recommendations on the basis of international experience, of the measures to be followed to meet the requirements set out in the SR documents.

Because of the process for the preparation that involves several experts inside and outside the IAEA and the review and approval mechanism that involves all the Member States of the Agency, the safety Standards are documents of large consensus that reflect established and well accepted safety rules.

4. MAIN TENETS TO DEVELOP THE SAFETY REQUIREMENTS FOR THE DESIGN

The Requirements for the design play an important role in establishing the safety level of the product. They have also great impact on the cost of the plant and operating procedures.

The general logical process to generate the Safety Requirements for the Design is represented in Figure 3 and briefly described below.

**FIG. 3. Logical process to generate the safety requirements**

The Safety Requirements can be derived from a set of limited safety principles which directly descend from three well established safety objectives. The safety objectives define the general targets that shall be achieved by a nuclear installation to protect the operators and the population. They are the same for all nuclear installations including nuclear reactors and are independent of the kind or size.
For nuclear reactors, the compliance with the safety objectives is achieved assuring that the three fundamental safety functions *Confinement of radioactive material, Control of the reactivity* and *Removal of the heat from the core* are fulfilled for all the plant operational, accidental and post accidental conditions in the respect of the radiological targets.

The extensive implementation of the strategy of the defence in depth ensures that the fundamental safety functions are reliably achieved and with sufficient margins to compensate for equipment failure and human errors.

Defence in depth has been proved to be generally applicable and very effective in assuring safety in Nuclear Power Plants. It can be used as main guidance for the preparation of safety requirements. As a matter of fact as it has been shown by INSAG [3], that there is a clear correspondence between each of the five levels of defence in depth and the requirements. It is reasonable to assume that this correspondence is maintained for all kind of reactors regardless of their size or specific safety features.

The deterministic concept of defence in depth is integrated with PSA considerations (e.g. system reliability, probabilistic targets, etc.) that also provide input for additional requirements and ensure a well balanced design to cope with all Postulated Initiating Events (PIE).

The actual level of safety is the result of the compliance of the design with detailed criteria and requirements (deterministic and probabilistic). In other words, the level of safety depends on the way defence in depth is implemented in the design taking into account the implications of the specific features and technology.

4.1. Top-down approach

The proposed top-down approach consists of a systematic review of the existing requirements for Nuclear Power Plants [2] starting from the most general (applicable to all nuclear plants) and down to the most specific and more technology dependent. This process is schematically presented in Figure 4.

The Requirements for a specific type of reactor are generated through a critical interpretation of the ‘objectives’ and ‘essential means’ associated with each level of defence in depth (see Tab.1), and the full understanding of the safety features of the specific reactor.

The safety requirements for Nuclear Power Plants have reached the current status through a long development process which incorporated the results of the extensive operating experience and the experience gained from the errors of the past. The current safety requirements define the safety approach developed and refined in many years. Although they are mostly been developed for large water cooled reactors, it is reasonable to assume that they are a very good starting point for the preparation of the design requirements for SMRs including non-water cooled reactors such as MTGRs. It can be expected that the requirements for these reactors that makes extensive use of inherent safety features could be less demanding than those for large water reactors.

The mechanism for judging the applicability or adequacy of a requirement for existing NPP to a SMR should be based on the full understanding of its contribution to defence in depth. The “transfer function” (central box in Fig. 4) that establishes the requirements for a generic
nuclear reactor plant from the requirements for existing NPP, should not simply be interpreted as a filter to accept or not a requirement but as a mechanism to generate new requirements if they are necessary because of the features of the specific nuclear reactor plant. For example, an inherent feature that fulfils a safety function in a very reliable way could allow for a relaxation of the requirements for a safety system or even to the possible elimination of the safety system that performs that function for water reactors. On the other hand, specific features or materials could possibly initiate some events for which adequate preventive or mitigative measures could be necessary.

This process will lead to the compilation of a consistent set of requirements organised in a hierarchical way with the general requirements at the top and the more specific at the bottom like those for Nuclear Power Plants.

**FIG. 4. Generation of Requirements for a SMR**

### 4.2. Main characteristics and safety features of future SMRs

Because of the large number of kinds of SMRs it is quite arduous to make safety and licensing considerations valid for all reactors without being superficial. Specific precise considerations can only be done for specific reactors. However there are common aspects that deserve to be mentioned.

The limited size of the reactors will allow, for example, to achieve the decay heat removal function with simple and reliable systems. A major effort devoted to enhance accident prevention through the improvement of the strength of the first level of defence in depth will
lead, in general, to the relaxation of the requirements on expensive mitigative systems. This can be achieved through technical solutions that eliminate some accident sequences by design, using the intrinsic capability of the systems in a more effective way, reducing the power density, increasing the design margins, increasing the time constants for the overall reactor system in order to slow down the transients response of the system. These measures allow more time for automatic control and operator actions, they simplify the design of control systems and simplify the actions required of operators (Level 2 of defence in depth), and they decrease the number and severity of challenges to structures and safety systems (Level 3 of defence in depth).

4.2.1 Severe accidents

For new reactors an important goal is to further reduce the potential radiological consequences of accidents. Severe accidents have to be considered systematically from the early design phase. The likelihood of core damage is expected to be very low and this should be demonstrated in a clear and convincing manner for example also through integral reactor testing.

The target proposed by INSAG (CDF< 10^-5 together with the practical elimination of sequences that could lead to large early radioactive release) can be used as reference and this will lead to the elimination of the need for any prompt off site response.

4.2.2 Risk-informed decision making

The challenge for the future is to develop more confidence in the PSA tools and to demonstrate that sufficient defence in depth can be achieved through simpler and cheaper technological solutions. Risk informed decision making plays an important role in the development and optimization of future reactors to achieve high levels of safety and reduce the cost in particular through simplification of safety systems and a sound safety classification of structures systems and components.

4.2.3 Simplification and use of passive features

The simplification of plant systems is another well established trend for future SMRs, especially with regard to safety systems. This goal of greater design simplification goes hand-in-hand with the goals of increased design margins, robustness and response time.

A simplified system is one that is more easily operated and maintained, which has reduced the number of components to the minimum necessary to provide all safety and performance functions (thereby reducing the number of failure points and modes), and which will be resilient to human errors in operation.

Passive systems offer the opportunity to eliminate complex active systems that rely on a large number of safety grade support systems by applying the advantages of simple gravity driven or thermal gradient driven safety systems. The challenge is to demonstrate the capability and the reliability of these passive systems and to deal with their longer time response.

The extensive use of passive components poses some problems for performing reliable PSAs. The safety of these reactors is determined by Initiating Events of very low probability. The consequences of these events that can be very serious, are determined by the direct
phenomenological response of the plant to these events, rather than by a sequence of failures of systems, which individually have higher probabilities and which can be analyzed and modeled with much less uncertainty.

4.2.4 **Digital instrumentation**

Systems based on digital technology have demonstrated very high reliability in many industrial applications, including NPPs allowing for a very good supervision and control of the plant. New generation designs offer the opportunity to fully implement this technology and benefit of the associated advantages.

4.3. **The objective-provisions tree**

The method of the objective-provisions tree represents a preliminary attempt to systematically address the “critical review” of the implementation of the defence in depth as indicated in the of Figure 4.

The logical framework of the objective-provisions method is graphically depicted in terms of a tree such as shown in Figure 5. At the top of this tree is the level of defence in depth that is of interest followed by both the objectives to be achieved and the barriers to be protected.

![FIG. 5. Defence in depth objective-provisions tree](image)

The objectives can be directly derived from those of Table 1. For example the main objective for Level 3 is to achieve the control of accidents within the design basis. This main objective can also be expressed in terms of more specific objectives such as: (a) limit the damage to fuel, (b) avoid any consequential damage to RCS, (c) maintain the confinement of radioactive products.
To ensure the objective of each level of defence there are a number of safety functions that should be assured. (E.g. shutdown the reactor, maintain the reactor in safe shutdown conditions...). At this point the challenges to the fulfilment of the safety functions can be identified. These challenges are general processes or situations that can prevent adequate performance of the safety functions (e.g. reactivity excursions that could damage the fuel before the shut down). The challenges arise from a variety of mechanisms which have also to be identified. The identification of the mechanisms that can challenge the delivery of a safety function is, an essential task in the development of the logical framework for inventoring the defence in depth capabilities of a nuclear power plant. Once the mechanisms are known it is possible to determine the necessary provisions necessary to prevent and/or control these mechanisms.

This methodology is being currently applied at the Division of Nuclear Installation Safety of the IAEA to prepare the guidelines for the development of Safety Requirements for Modular High Temperature Gas Cooled Reactors. If completed successfully the same approach will be extended to other reactors including Research Reactors.

CONCLUDING REMARKS

The SMRs to be built in the future will have to meet at least the level of safety presently required for the best large nuclear power plants currently in operation or being designed.

The safety requirements for the design of SMRs can be generated through a review process of the requirements established for current power plants. This process will be based on a critical application of the strategy of defence in depth. The adoption of technical solutions allowed by the size of the reactor and the specific technology will probably lead to the enhancement of prevention improving the strength of the first level of defence with the consequent relaxation of the requirements for expensive mitigative systems.

The use of small independent modules rather than a single large plant can also represent a viable solution to achieve simplification and cost reduction (e.g. heat removal function with simple passive systems).

The challenge for the future is to develop more confidence in the PSA tools and to demonstrate that sufficient defence in depth can be achieved through simpler and cheaper technological solutions. Risk informed decision making will play an important role in the development of future reactor of any kind It will help to achieve high levels of safety and reduce the cost in particular through simplification of safety systems and a sound and a well balanced safety classification of structures systems and components.

The standardization, prefabrication and modularity of the facilities and the simplification of the licensing procedures through a certification process are suitable means to reduce the costs.

Additional benefits can be obtained through harmonizing licensing criteria procedures used by the nuclear community to the greatest possible extent, based on worldwide scientific resolution of technical issues and accepted standards of safety adequacy.
REFERENCES


SAFETY BY DESIGN: A NEW APPROACH TO ACCIDENT MANAGEMENT IN THE IRIS REACTOR

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Abstract

Safety by design is an improved approach to safety, which is being fully implemented in the IRIS reactor. It means to prevent the accident through eliminating by design the probability of its occurrence rather than engineering how to cope with its consequences. The integral reactor vessel configuration of IRIS is an ideal layout for implementing this approach. A brief review of how various accidents can be handled in IRIS is given, followed by a detailed discussion of the IRIS response to small-to-medium LOCAs. An innovative containment design coupled with the integral vessel allows the core to remain safely covered for days under the worst LOCA conditions, without any safety injection. Details of the response to various postulated LOCAs are given. A brief review of the Core Melt Exclusion Strategy (CMES) is given; its application to IRIS will demonstrate a very significant improvement in reactor safety, licensing and economics which can possibly be extended to other advanced reactors.

1. INTRODUCTION

IRIS (International Reactor Innovative and Secure) is an integral primary system configuration, light water cooled, modular reactor of small-to-medium power (100-335 MWe/module) which is being developed by an international consortium of 13 organizations (industry, utility, academia and laboratory) from six countries. A brief description of the IRIS reactor is provided in a companion paper in this symposium [1]. As a help to the reader, however, the IRIS vessel layout is again shown in Fig. 1.

The IRIS integral vessel houses the reactor core and support structures, core barrel, upper internals, control rod guides and drivelines, steam generators, pressurizer, heaters and primary coolant pumps. Another characteristic feature of IRIS is the long life (8 years) straight burn core with no shuffling or refueling to satisfy proliferation resistance considerations.

A third fundamental feature of the IRIS design is its approach to safety which we call “safety by design” and which was developed to fulfill the enhanced safety requirements stated by the DOE NERI program.

The “safety by design” approach and its implications for licensing is the focus of this paper.

The work is far from being complete, but the overall approach has been established and important results have already been obtained, as will be discussed in the following sections.
2. THE “SAFETY BY DESIGN” APPROACH

The so-called Generation II reactors (e.g. current LWRs) cope and interfere with accident sequences through active means to assure that the consequences of the accident remain within specified acceptable limits. Generation III reactors (AP-600 and similar designs currently available for deployment) do the same, but with passive means. Generation IV nuclear energy systems are currently being designed to satisfy requirements of improved economics, enhanced safety, proliferation resistance, and reduced nuclear waste stream. In answering to the requirement for enhanced safety, IRIS has been the first of the Generation IV nuclear energy systems to formulate the philosophy that Generation IV systems should leverage their novel design and operational characteristics to prevent accidents, to the largest extent possible. That is, the “safety by design” approach is prevention of the accident through
eliminating by design the possibility for the accident to occur or, if that is not possible, limiting by design its consequences to an acceptable level.

Table I summarizes how the IRIS designer can take advantage of the IRIS characteristics to enhance safety.

**TABLE I. IRIS SAFETY BY DESIGN**

<table>
<thead>
<tr>
<th>Design Characteristic</th>
<th>Safety Implication</th>
<th>Related Accident</th>
<th>Disposition</th>
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<tbody>
<tr>
<td>Integral configuration</td>
<td>No external loop piping</td>
<td>Large LOCAs</td>
<td>Eliminated</td>
</tr>
<tr>
<td>Tall vessel with elevated steam generators</td>
<td>High degree of natural circulation</td>
<td>LOFAs (e.g. pump seizure)</td>
<td>Either eliminated (full natural circulation) or made acceptable (partial natural circulation)</td>
</tr>
<tr>
<td></td>
<td>Can accommodate internal control rod drives</td>
<td>Reactivity insertion due to control rod ejection</td>
<td>Can be eliminated</td>
</tr>
<tr>
<td>Long life core</td>
<td>No partial refueling</td>
<td>Refueling accidents</td>
<td>Reduced probability</td>
</tr>
<tr>
<td>Large water inventory inside vessel</td>
<td>Slows transient evolution Helps to keep core covered</td>
<td>Small-medium LOCAs</td>
<td>Core remains covered with no safety injection</td>
</tr>
<tr>
<td>Reduced size, higher pressure containment</td>
<td>Reduced driving force through primary opening</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Inside the vessel heat removal</td>
<td></td>
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</table>

It is well known that large LOCAs are not possible in an integral vessel configuration, where the steam generators, reactor coolant pumps and the pressurizer are inside the vessel, because there is no external large loop piping. IRIS, like other integral designs, takes advantage of this design configuration. In addition, the integral configuration results in a tall vessel with elevated steam generators, which is an ideal layout for natural circulation. Trade-off studies [2] were performed by the IRIS team to determine the optimum level of natural circulation. It was found that full natural circulation is best for low reactor power outputs. At power levels close to 100 MWe, the vessel height required to attain full natural circulation exceeds 30 m and the reactor is uneconomical. The IRIS team does have a small reactor design, for powers equal to or less than 50 MWe, called IRIS-50, which features full natural circulation and which was submitted to DOE for consideration in their study of small reactors for remote locations.

However for the “mainstream” commercial IRIS we have adopted “partial natural circulation,” where the reactor flow is driven both by pumps head (forced circulation) and fluid density differences (natural circulation). The reactor is designed such that the amount of natural circulation is sufficient to prevent exceeding the transient design limits during the worst postulated LOFA.
Another interesting feature of the integral design is that, by locating the steam generators in a peripheral annular configuration for ease of maintenance and for allowing core changeover without prior removal of the steam generators, there is an open space above the core extending in excess of 10 m. This space rather than containing the control rod drivelines, as shown in Fig. 1, could accommodate the control rod drive mechanisms (CRDMs). This will automatically negate the reactivity insertion accident due to a control rod ejection, because there is no penetration of the vessel head since all the control systems are inside the vessel.

Actuation of the control rods can be through internal electromagnetic motors (a system proposed for the Japanese MRX vessel [3,4]) or externally controlled hydraulic systems (the system proposed for the Argentinian CAREM [5] and Chinese NHR [6,7] reactors). We have submitted a proposal to DOE to study these systems for their applicability to IRIS. This is clearly a safety by design feature made possible by the integral vessel configuration, and which IRIS intends to pursue. However, until the proposal study is performed and the practical deployment assessed, IRIS will continue to feature traditional drives as shown in Fig. 1.

The probability of refueling accidents (another class IV accident) is significantly reduced by the elimination of fuel shuffling and infrequent refuelings which are made possible by the straight burn long life core.

The most innovative IRIS feature in terms of safety by design is the handling of small-to-medium LOCAs which are historically the accidents yielding the worst consequences. The IRIS approach is to reduce the pressure differential between vessel and containment following a LOCA, thus reducing the driving force across the rupture and ultimately the coolant loss.

This is accomplished through 1) a high pressure containment which increases the pressure downstream of the break and 2) an efficient heat removal system inside the vessel which reduces the pressure upstream of the break. The integral vessel design enables reduction of the containment size to about half the diameter needed in a comparable LWR. Thus, at the same metal thin shell stress, the spherical IRIS containment can take a pressure at least four times higher than the cylindrical containment in a loop reactor. Actual calculations [8] indicated that the IRIS containment peak pressure is limited to ~ 1 MPa (130 psig) by using a pressure suppression pool with an air volume of 300 m$^3$. The heat removal inside the vessel is provided by the multiple steam generators and the large water inventory inside the vessel provides a grace period by slowing the transient evolution.

Finally, the reactor vessel is located within an open cavity extending axially above the top of the core to collect any liquid discharged or condensed inside the containment. The cavity will flood up and refill the reactor vessel, thus terminating any LOCA at or below the core elevation. It will also assure that the outside of the reactor vessel is sufficiently cooled to prevent vessel melt-through following postulated severe accidents.

3. IRIS RESPONSE TO SMALL-TO-MEDIUM LOCAS

Because of the design features discussed in the previous section, the IRIS core remains safely under water, without any core water makeup or safety injection, for an extended period of time (days) under the worst (in terms of size and location) hypothetical LOCA.
Analyses were performed for an IRIS reactor power of 100 MWe. Initial analyses were conducted, using a proprietary code of the Polytechnic of Milan (POLIMI), for the hypothetical worst combination of the largest postulated penetration pipe size break (4 in.) and the worst axial location (above the top of the core and just above the reactor vessel cavity, 7 m above the bottom of the vessel), assuming no water makeup or safety injection.

**FIG. 2. The IRIS Reactor Vessel/Containment Pressure Differential Equalizes Quickly and Actually Reverses Early in the Transient.**

**FIG. 3. Coolant Flow through the Rupture showing Flow Reversal in the Early Phase of the Transient.**

As shown in Fig. 2 the reactor vessel/containment pressure differential quickly equalizes in the first 15 minutes. Afterwards, a very interesting phenomenon occurs for about one hour: the containment pressure is higher than the vessel pressure due to the fact that while there is very limited air cooling of the containment, there is vigorous heat removal inside the vessel by the steam generators, with consequent steam condensation. Since the reactor vessel
pressure is now lower than the containment pressure, the break flow reverses and steam, along with non-condensable gases, is drawn back into the vessel as shown in Fig. 3. The consequence of this flow reversal is that right after the initial depressurization, in the critical early phase of the transient, the coolant level in the vessel increases rather than decreases. Afterwards, the heat removal rate from the containment exceeds the vessel heat removal via the steam generators and a slow, small coolant outflow results. The core remains safely under water for the entire duration of the transient time considered.

FIG. 4. The IRIS Core Remains under 2 Meters of Water after more than 2 Days, during the Worst Postulated Small LOCA.

FIG. 5. Coolant Level during a Vessel Break inside the Cavity.
Another series of analyses were conducted using the GOTHIC code, to confirm the POLIMI results and to extend the transient beyond the 12 hour period previously analyzed. A more realistic case was run first: still a 4” break, even though no penetration larger than 2.5-3” is envisioned in IRIS, but occurring at a location 12.5 m above the bottom of the vessel, which is the axial location of the lowest vessel penetration. The POLIMI results were confirmed and, as shown in Fig. 4, the top of the core was still under 2 meters of water after 2.5 days into the accident, without any safety injection or water makeup. Another case considered was a 2 in. break occurring at 6.5 m above the bottom of the vessel. This corresponds to the postulated break of a long-term core makeup pipe (which actually is expected to have a diameter less than 1 in.), which is located near the top of the core, but within the vessel floodup cavity. The results are shown in Fig. 5; because of the low break location, the mass loss is initially quite large until the vessel/containment pressures equalize. For this break location there is a significant reverse flow, as the vessel cavity becomes flooded and water flows back into the vessel via the break. The large flow reversal through the break results in the reactor vessel liquid level increasing to almost 7 m, which is the elevation of the cavity flood up level. Thus, after approximately 5 hours the transient is practically terminated. The core remains safely covered throughout the entire transient.

Another case analyzed was a 1 in. break at the bottom of the vessel to simulate a beyond-design-basis failure of the reactor vessel; the results were similar to the previous case, i.e. the core remained covered. Overall, we investigated break sizes of 1, 2 and 4 in at various elevations: bottom of the vessel, above the core (inside and outside the cavity) and 12.5 m from the bottom of the vessel.

Reported in Table II are the Class IV accidents examined during the licensing of the AP-600 and their current resolution in IRIS. The safety by design approach of IRIS eliminates from consideration or positively addresses over 60% (5 of 8) of the AP600 Class IV Limiting Faults.

We have so far only addressed in detail the LOCAs and LOFAs, in addition to assessing the rod ejection and refueling accidents. The other Class IV accidents: steam and feed line break, steam generator tube rupture, as well as other accidents such as ATWS, station blackout and loss of heat sink still have to be addressed under the safety by design approach.

<table>
<thead>
<tr>
<th>Accident IRIS</th>
<th>Safety by Design</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Steam system piping failure (major)</td>
<td>Not yet addressed</td>
</tr>
<tr>
<td>2. Feedwater system pipe break</td>
<td>Not yet addressed</td>
</tr>
<tr>
<td>3. Reactor coolant pump shaft seizure or locked rotor</td>
<td>Eliminated or degraded</td>
</tr>
<tr>
<td>4. Reactor coolant pump shaft break</td>
<td>Eliminated or degraded</td>
</tr>
<tr>
<td>5. Spectrum of RCCA ejection accidents</td>
<td>Can be eliminated</td>
</tr>
<tr>
<td>6. Steam generator tube rupture</td>
<td>Not yet addressed</td>
</tr>
<tr>
<td>7. Large LOCAs</td>
<td>Eliminated</td>
</tr>
<tr>
<td>8. Design basis fuel handling accidents</td>
<td>Reduced probability</td>
</tr>
</tbody>
</table>
4. CORE MELT EXCLUSION STRATEGY

This section discusses the basis of a safety approach applicable to the design and/or the assessment of the defenses implemented for the prevention and the management of severe plant conditions through a Core Melt Exclusion Strategy (CMES).

The Core Melt Exclusion Strategy [9] is an attractive accident management strategy, since the core melt progression and the consequent phenomena threatening the containment integrity are excluded. The CMES will replace the currently adopted core melt management and will lead to improve the plant safety level and possibly public acceptability. It is particularly applicable to low power density, medium size reactors such as IRIS.

The main objective of our effort is to ensure the coherence of this approach with the currently available safety related recommendations for future concepts. A Line of Defense method (LOD) is proposed which could be applied as a basis for the safety demonstration. It is important to stress that the CMES is in principle applicable to most future nuclear plants. Its application to IRIS will also be a demonstration of such viability.

Background

A comprehensive set of safety principles and recommendations for future power plants is currently under discussion at an international level. Even though a conclusion has not yet been reached, we believe that it will include: an increased effort in plant simplification and implementation of functional redundancies using passive systems, when necessary; and, the adoption of the Defense in Depth principle (DiD), with an increased effort on prevention, improvement of the forgiving plant inherent characteristics, reduction of the core melt frequency, confinement improvement and rejection of any cliff edge effect\(^1\) (i.e. all the DiD levels shall be considered).

The design effort should implement three defenses: comprehensive, homogeneous and progressive. The comprehensive defense requires that an effort be made to prevent all the plausible initiating events including those that can lead to the core degradation. The homogeneous defense requires that great differences be avoided among the contributions of the different event families, and/or the plant status, to the “severe plant condition” frequency.

Finally, the progressive defense requires avoiding short sequences that can lead to significant releases. The objective is a design with inherent characteristics and automatic implementation of the safety systems.

Among the recommendations by the European Technical Safety Organizations (TSO), some are applicable to the CMES: ...Accident situations which would lead to large early releases have to be “practically eliminated”...while recognizing that There is a need to develop adequate guidance to clearly establish when “sufficient design and operation provisions” have been taken to practically eliminate a severe accident sequence.

The current probabilistic objectives to obtain the “practical elimination” of accidents are that the cumulative frequency for severe plant conditions – coping with the allowable design

\(^1\) This risk corresponds to the mobilization of a potentially unacceptable source term - severe plant conditions – with the simultaneous loss of the containment (releases much higher than those which are acceptable).
release – shall be lower than $10^{-5}$/reactor year; and, the cumulative frequency of exceeding the design release shall be lower than $10^{-6}$/reactor year.

To satisfy the latter when considering a dozen of Postulated Initiating Events (PIE) families, each asking for some safety function achievement, leads to an objective of a fraction of $10^{-7}$/reactor year, per family and per safety function.

**CMES Approach**

CMES relies on Lines of Defense (LOD), which are any inherent characteristic, equipment, system, implemented into the safety related plant architecture as well as any procedure consistent with the General Rules for Plant Operation (e.g. human actions) which accomplish a given safety function.

In order to define this concept more precisely, two types of LOD are considered:

- The strong lines (called “a”) with a probability of failure of the order of $10^{-3}$ to $10^{-4}$ per year or per demand, and
- The average lines (called “b”) with a probability of failure of the order of $10^{-1}$ to $10^{-2}$ per year or per demand.

As a design goal, accident situations which would lead to large early releases have to be practically eliminated. As stated by TSO “when they cannot be considered as physically impossible, design provisions have to be taken to design them out”. TSO also stressed that the “practical elimination” of such accident sequences is a matter of judgement: each type of accident sequences has to be separately assessed. Moreover, the “practical elimination cannot be demonstrated by the compliance with a general cut-off probabilistic value”. The IRIS “Safety by Design” approach squarely addresses the TSO requirement.

Meeting the objective of $10^{-7}$/reactor year, per family and per safety function to prevent severe plant conditions, at least 2 “a” LODs should be implemented to manage the Design Basis Conditions. The key condition for the applicability of this rule is the effective independence of the LODs. When implemented, they must fulfill the principle of functional redundancy, i.e. once the upstream LOD fails, the one downstream is still able to achieve the requested function.

Moreover, as previously seen, the design of the plant must also satisfy the requirements of homogeneity, progressiveness and completeness. Taking into account both the probabilistic objective and the homogeneity and progressiveness principles, leads us to recommend the architecture shown in Fig. 6, where LOD4 represent all the inherent characteristic, equipment, system, procedure, implemented to practically achieve the CMES objectives. The safety related functions for the CMES are tentatively represented in Fig. 7. Knowing that the robustness of the chain is defined by its weakest link, one can stress the fact that the CMES design and assessment shall proceed with an homogeneous process ensuring the final needed quality: each of the different LOD4/i should be at least equivalent to a strong LOD, i.e. their reliability shall be assessed to ensure that the corresponding order of magnitude is coherent with the $10^{-3}$ to $10^{-4}$ per demand. All the IAEA generic safety requirements, e.g. use of proven technology; general basis for design; classification of components, structures, and systems; reliability targets, should be carefully applied in order to meet these technical objectives.
FIG. 6. Example of LOD Implementation.

FIG. 7. Safety functions to be assured simultaneously to guarantee the CMES.
Previous studies have indicated the attractiveness of passive LODs [10] and have outlined the design process [11]. The IRIS Safety by Design takes the LOD approach to a further level by eliminating the physical possibility of some accidents. The preliminary safety analyses indicate that the strong LODs related to maintaining the core integrity are fully satisfied in IRIS. Actually, as shown in Sections 2 and 3, large LOCAs cannot occur and for small-to-medium LOCAs it appears that core recovery and hence severe core damage do not occur. Thus, there is no need for LOD 4/4. The other LODs are provided directly in the design.

CONCLUSIONS

IRIS has taken maximum advantage of its integral configuration to successfully implement a safety by design approach where accidents are eliminated by design. Analyses conducted so far indicated that for IRIS both LOCAs (of any size) and LOFAs are no longer a safety concern. In IRIS the containment is designed such that, in addition to its normal “containing” role, proactively contributes to manage the small/medium LOCAs. The CMES strategy finds a natural resonance with the IRIS safety approach since strong LODs are germaine to the safety by design. When fully demonstrated, CMES will represent a tremendous improvement in licensing and economics not solely for IRIS but possibly also for other advanced reactors.

REFERENCES

SAFETY MARGIN EVALUATION IN THE NEXT GENERATION OF NUCLEAR REACTORS

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Abstract

The new concepts of nuclear reactors, including the IV generation ones, are based on the passive and inherent safety concepts. The evaluation of the safety margins of these plants is a challenge due to the special characteristics of the systems to be analysed, based on passive system for the core cooling and decay heat removal. The working principles of the new emergency safety features (ESF), based on weak driving forces, required new and advanced models for the evaluation of plant safety margins with respect to the proven system code. In the present paper, this problem will be investigated using new models, partially qualified on the basis of the results of some experimental apparatus. In particular, the stronger coupling that exists in these reactors between primary system (PS) and containment in accident conditions with respect to the present generation ones, requires a new evaluation of the safety margins in order to get reliable information, including new experimental data to acquire a more in-depth knowledge. As a consequence, it is highlighted as the traditional methodology of analysis may be not adequate when applied to the next generation reactors; this is also carried out on the basis of experimental data obtained in different facilities devoted to the phenomenological analysis of the new passive ESFs. A new integrated model for the evaluation of the safety margins is highlighted; the main results of the code validation programme, including analysis of experimental data on containment stratification using a methodology for dealing with complex, interacting physical processes, based on a general scaling criteria are also presented.

1. INTRODUCTION

Proposed new reactors have challenged code developers with the special characteristics of systems to be analysed, in particular concerning the thermal-hydraulic simulation; furthermore, particular attention has been devoted to the intrinsic and passive safety characteristics of new plants, which are credited to increase reliability and public acceptance [1]. These concepts have led to an extensive use of passive and fail-safe ESFs and several kinds of passive safety systems have been designed and investigated [1], [2] and [3]. As is clear from the working principles of these new ESFs, the stronger coupling that exists between PS and containment phenomena during accident conditions with respect to the present generation plants, requires an integrated simulation of both systems in order to get a reliable assessment capability. This is essential because, in the next generation reactors, almost all accident sequences are changed to low-pressure scenarios by PS depressurisation procedures. Accident analyses which treat PS and containment phenomena as completely separated are not adequate: in fact, with the traditional sequential calculation approach, which uses the results of a PS calculation to define the mass and energy releases to be used in the containment calculations, approximate boundary conditions are applied to the PS and inaccurate sources are obtained for containment pressurisation especially when, after depressurisation, sub-critical flow conditions are possible. Only a continuous interaction between PS and containment models can lead to the evaluation of the tight coupling existing between these two systems. As a first approach, PS codes adopted for advanced reactors are being adapted to represent the containment system in a simplified way. The “informatic” coupling of well assessed PS codes with proven containment ones was also proposed as an additional way to achieve understanding of interactions occurring during postulated accidents, at least for short term analyses [4]. The FUMO integrated code [5], presented in the following, is an example of this integrated approach, supplying to a well assessed containment code the lacking information on boundary conditions.
In relation to the new ESF concepts, several facilities have been built to investigate their behaviour, both for PS and containment phenomena, and there has been interest in performing scaling analysis of such facilities and to provide data to qualify best estimate (BE) codes for licensing purposes [3], [6]. These scaling analyses have often been based on the US Nuclear Reactor Commission (NRC) Hierarchical Two-Tiered Scaling (HTTS) methodology [7], specifically intended to provide guidance in the design of new experimental programmes or in the applicability investigation to a specific plant for available experimental data. This methodology was applied by DIMNP [6], [8] to the scaling analysis of stratification tests performed in two facilities, at different scales, including the effects of an Innovative Containment Cooling System (ICCS) proposed by ENEL. In order to evaluate the behaviour of the dimensionless parameters governing the stratification phenomena, FUMO [9] was employed; the results of these analysis about the containment stratification phenomena are also summarised in the following.

2. SAFETY CHARACTERISTICS OF ADVANCED REACTORS

Advanced reactors represent an evolution of the actual proven reactors in which the aspects related to passive and inherently safe characteristics of the overall plant are strongly enhanced [1]. Thanks to the simplifications introduced in the plant itself and to the adoption of protection systems based on readily understandable principles, they may promote a greater public acceptance of the nuclear power. Furthermore, the adoption of simplified systems accomplishing the basic safety functions (i.e. the control of the coolant inventory, the control of reactivity, decay heat removal and fission product retention), making use of natural forces only, helps to increase the so-called “walk-away time”, that is the time period during which no external action is needed to maintain the plant in safe conditions. These passive safety features should not have the characteristics of “first of a kind” systems but should have a well known behaviour. Nevertheless, as they are intended to withstand a wide range of accident conditions, they must be thoroughly analysed, both experimentally and theoretically, to assess their characteristics. In particular, as the driving forces available to perform their safety functions are generally weaker than in corresponding active systems, studies must be carried out to make sure that they are strong enough to cope with postulated accidents. The use of these passive safety systems is not limited to small and medium sized advanced reactors due to their weak natural driving forces but due to their necessary scaling. Technical limitations exist, e.g. due to the available elevations and to the available heat transfer areas, but their application to larger advanced plants (as the European Pressurised Reactor), where there is an increase of the decay power to be removed by these passive mechanisms, is merely an economical problem [3].

2.1. Main Phenomena

A consequence of the introduction of passive devices to perform the core cooling action is the decrease of the PS pressure value that must be reached to obtain this long term plant cooling. While, in proven reactors, a depressurisation down to about 2 MPa was enough to start the flooding by the active low pressure systems, in new reactors a pressure equalisation between PS and containment is needed to start the water injection from the large capacities included inside the containment itself. Therefore, an Automatic Depressurisation System (ADS), as previously employed in BWRs, must be introduced to change every high pressure sequence into a low pressure one, with the aim to let a quicker injection driven by the gravity force. One of the results is the mentioned strong influence of the containment behaviour on the PS evolution.
Other phenomenologies that have to be considered in the evaluation of the safety margins for the new reactors will be highlighted in the following, also on the basis of numerical and experimental tests. Evidence of the influence of the containment system on the PS was obtained in some experimental tests performed in the past at the DIMNP (Department of Mechanical, Nuclear and Production Engineering) Laboratories of Pisa University on the PIPER-ONE apparatus [19], a 1/2200 volume scaled simulator of General Electric (GE) BWR-6 plant. Three tests were performed, mainly differing for the pressure imposed inside the GDCS tank. Figure 1 reports the rod surface temperature trends: note the effect of the increase in GDCS pool pressure on the core rewetting time. The PIPER-ONE facility was also employed [12] to study the long term heat rejection capabilities of passive Residual Heat Removal (RHR) systems, capabilities that strongly depend on the size of the pool where the final heat exchanger (HX) is immersed and on the location of the HX itself inside this pool. In fact, the stratification of the pool temperature may decrease the actual amount of pool water involved in the heat transfer process. In these tests, a HX was deliberately put in the upper part of the pool to enhance the water temperature stratification; the result was a strong temperature difference between the top part of the water mass, at boiling conditions, and the bottom part of the pool, remaining at the initial temperature throughout the whole experiment, drastically changing the heat transfer conditions.

![FIG 1: PIPER-ONE - rod surface temperatures.](image)

From the aspect of containment phenomena, several of the new plant concepts are characterised by the use of innovative cooling systems, able to remove the heat released inside the containment following a hypothetical accident, through passive heat transfer mechanisms. The installation of such a system requires a reliable knowledge of the temporal and spatial distribution of non-condensable gas inside the containment atmosphere, both to ensure the heat removing efficiency of the system, and to guarantee that the local concentration of combustible gases, such as hydrogen, does not reach dangerous values during the ESF operation. The problem of the weakness of the driving forces for passive ESFs is particularly highlighted in both AP-600 and SBWR plants where, after a postulated accident, the decay heat is rejected from the containment to the external environment only by these passive mechanisms, including natural circulation and steam condensation. In particular, in AP-600, condensation occurs on the internal surface of the containment liner and the condensate is routed towards the reactor cavity. An external spray system enhances the heat transfer from the outer liner surface in the medium term, while air natural circulation provides a sufficiently effective heat removal mechanism in the long term. The lower containment compartments are designed to collect water in order to maintain a minimum level above the main cooling lines: this creates natural circulation flow paths involving both PS and containment which
accomplish long term core cooling and decay heat removal. Also in the AP-1000 design, the natural convection phenomena and the heat removal from the double concrete containment are strongly influenced from the adoption of localised ESFs. In SBWR, steam presents inside the dry well is condensed in HXs having the drain lines connected with the GDCS pool, while the vent lines discharge noncondensable inside the wet well pool. From the GDCS pool, the water is continuously fed to the reactor vessel where it is vaporised and again discharged to the containment through the postulated break and valves.

3. MODELLING REQUIREMENTS

Present PS codes are not qualified to cope with containment phenomena and difficulties were found even in their use for the simulation of advanced reactor scenarios [1], especially when low pressures and noncondensables are involved. Moreover, they are characterised by computational efficiencies which hardly allow the analysis of transients lasting some days or more, i.e. the duration required to demonstrate long term cooling capabilities. On the other hand, containment analysis tools have generally no capability to adequately represent the reactor system. The optimum solution [10] could be the development of a new code, or the qualification of an existing one, in order to get a tool capable to treat simultaneously the two systems with reasonable computational efficiency. Another possible choice [4], the coupling of a qualified PS code with a qualified containment code, continuously exchanging data by informatics way along the calculation, shown some problems in its practical application. In the case in which only PS or only containment phenomena are mainly addressed, approximate representations of the containment behaviour, in the former case, or of the PS response, in the latter, may be also acceptable. So, existing qualified codes may be coupled with simplified modules supplying the lacking information on boundary conditions. Such an approach, more attractive if fast-running performances are needed, has been adopted in the past at Pisa University [5], as briefly described in the following paragraph.

3.1. The Integrated FUMO Code

The integrated approach was started by upgrading the FUMO code [9], initially developed at DIMNP to analyse DBA and SA transients in LWR containments, with the introduction of “ad hoc” PS module. In this new integrated tool [10], containment models are the same implemented inside FUMO stand alone. Various models can be used to define the thermodynamic conditions of control nodes: the common choice is to consider a two-region model, consisting of a homogeneous air-water mixture and a liquid pool with allowance for the presence of six non-condensable gases. The effect of hydrogen production by corium-water interaction and its combustion are also taken into account. A new model for the simulation of natural circulation flows is also implemented: it allows to account for the buoyancy driven flows due to temperature and/or concentration differences among the compartments, using an algorithm based on an “electrical network” analogy [11]. Conductive heat transfer mechanism inside structures is evaluated with a fast-running “coarse-mesh” method, based on a cubic polynomial approximation of the temperature profile, and the different heat transfer coefficients are calculated with several of the most reliable correlations or models available.

The reactor coolant system is simulated by a simplified module, called SIMSIP; homogeneous equilibrium balance equations are solved, simulating - at the same time - the operation of the various injection devices (e.g. ESFs, level control systems, etc.) and the different connections with the containment. Time discretization of the balance equations is performed on the basis
of an “implicit” coupling between pressures and flow rates. Both homogeneous conditions and vertical phase separation are considered inside a control volume. Countercurrent flow at junctions is dealt on the basis of Wallis’ type correlation. Critical flow at valves/breaks is evaluated by adopting the Henry Fauske models for sub-cooled and low quality fluid and the HEM for high quality fluid, with an appropriate transition. A full-range heat transfer package which allows for six pre-CHF and post-CHF regimes is employed to calculate heat transfer coefficients. Point neutron kinetics, with one group of delayed neutrons, and a decay power model simulate power transients. The ESFs operation is simulated by several lumped parameter models; passive injection devices involving water tanks are simulated by a quasi-steady model which takes into account water discharge from a tank with a pressure behaviour calculated by the containment module. A quasi-steady model is also used for the Isolation Condenser (IC), in which natural circulation through the heat exchanger and the related heat transfer are evaluated.

FIG. 2: PIPER-ONE - Lower plenum pressure.

FIG. 3: PIPER-ONE - IC power.

FIG. 4: SPES-2 - PRHR flow-rate.
3.2. Code Validation Aspects

Validation of computer tools for long lasting transients is a difficult task due to the lack of applicable down-scaled experimental data, especially for combined behaviour of PS and containment [10]. As a consequence, no direct validation of an integrated code can be presently performed. The alternative is to separately validate the modules, obtaining a qualitative idea of the accuracy in the prediction of known phenomena. In this respect, the FUMO containment module has been thoroughly validated by DIMNP in the past [11]. Concerning the PS module, the usual assessment procedure, based on analysis of separate effect and integral tests is not applicable. This is due to the model simplicity and in consideration of the module purpose, being to supply the containment module with realistic boundary conditions especially during long term analysis.

So, a model shakedown was initially carried out, by comparing its results with those by other system codes [5] for some relevant transients. A further assessment, based on the analysis of reactor coolant system simulators, has been also performed, for both SBWR and AP-600 plants. These assessment analysis were related to the previously mentioned tests on IC performance [12], carried out in DIMNP PIPER-ONE facility (Figure 2 and Figure 3) and to the test S0110 on a steam generator tube rupture sequence (Figure 4), carried out on the SPES-2 facility [13], both with a satisfactory behaviour of the simplified module.

The new integrated code has been extensively employed in the past [14] by DIMNP, in collaboration with ENEL, for the licensing analysis of the two most famous projects for passive reactors: Westinghouse AP-600 and GE SBWR. The relevant phenomena and conclusions highlighted in these integrated studies are briefly summarised in the following part of the paper.

3.3. Relevant SBWR Phenomena

The relevance of an integrated approach on the SBWR analysis is highlighted in a sensitivity LOFW sequence with conditions of degraded ESF performance [15]. Figure 5 compares PS results obtained by integrated FUMO and by SIMSIP stand-alone, against a reference GE TRAC run. Core rod temperatures show the effect of the different assumptions adopted for the containment pressure on the GDCS intervention time and, consequently, on the core cooling conditions. In the stand-alone PS module and TRAC analysis, the containment pressure was set at a constant value drawn from previous separate containment calculations; on the other hand, in the integrated run, the actual containment pressurisation history was simulated, leading to a very strong reduction in the dry-out relevance.

![FIG. 5: SBWR LOFW - Core rod temperature.](image)
Figure 6 shows the differences between integrated and stand-alone calculations on the side of the containment phenomena. The feedback of containment pressurisation on the flow rate of depressurisation valves results in different dry well and wet well pressure histories. In particular, the equalisation of the dry and wet well pressures is predicted earlier for the integrated code respect to separated calculations.

Sensitivity studies were also performed, always using the Integrated FUMO, to understand the relevance of the heat removal mechanisms in a reference MSLB sequence [10]. These mechanisms are, as previously said, the only available way for the removal of the decay heat from the primary containment system towards the external environment. During this SBWR transient, firstly, a dynamic phase is experienced in which the vent clearing occurs and the dry well steam is condensed in the suppression pool, causing its heat-up; at the same time, most of dry well noncondensable gases are transferred to the wet well free space. Later on, different heat transfer mechanisms come into play [14], determining the overall containment pressure trend in the long term phase. As main result of these past analyses, it was verified as the target design pressure (483. kPa) of the SBWR primary containment remains well above the maximum pressure reached at the end of the three days period, even with very conservative assumptions on the heat transfer mechanisms and on the Passive Containment Cooling System (PCCS) efficiency, having a very particular configuration in this plant. This confirmed furthermore the strong safety margins considered in the SBWR design.

3.4. Relevant AP-600 Phenomena

The spectrum of the AP-600 sequences studied by DIMNP with the Integrated FUMO code has permitted [14] to highlight some typical AP-600 plant characteristics; these characteristics are common with other containment designs where the long term cooling action is based on passive (and so necessarily “weak”) mechanisms. In this plant, the containment (and PS too) pressure level that is achieved in the long term phase of the accident (again three days) is a direct function of the equivalent heat transfer coefficient on the external side of the containment steel liner, due to the external spray or to the natural convection flows. Pressure peaks inside the containment are rapidly reduced by the PCCS (containment external spray) and, even in the worst case for the pressure load, i.e. the Large LOCA sequence, the pressure is kept well below the required design limits. But there is of course an intrinsic limit in the long term pressure reduction, linked to the absence of any active cooling systems. A high temperature, and as a consequence, a high containment pressure, is the only driving force available for removing decay heat towards the external environment so, in the long term, the...
pressure trend is almost flat and, unless some active system is started, can only slowly decrease, following the decay heat generation history.

The purpose of these past DIMNP analyses [10] on this particular plant design was just to assess the long term behaviour of the whole system in order to check if the proposed PCCS concept is able to guarantee the required safety margins also in the long term phase of the transient, until one week.

Following the break opening and the subsequent reactor scram, the PS rapidly depressurises from the nominal pressure of about 15 MPa to about 0.25 MPa and in a very short time attains the containment pressure that is meanwhile increasing due to the strong blow-down release from the break (Figure 7). This pressure trend is characterised by two peaks, the first one linked to the main coolant release, the second one to the release of the latent heat from the steam generators. The containment pressure is rapidly decreased by the PCCS operation and, in the phase between 10,000 and 40,000 s, it becomes very small, due to the lack of a significant steam source [10]. In the long term phase (100,000 s), an almost stationary pressure level is reached; this pressure level is not strongly dependent on the accident history but it mainly derives from the evaluation of the global heat transfer resistance given by all the mechanisms which play a role in carrying the decay heat outside the containment. In this phase, the decay power is in equilibrium with the exchanged power (Figure 8) so the thermal-hydraulic conditions are strongly dependent on the PCCS operation conditions.
4. CONTAINMENT INTEGRITY CHALLENGES

During a SA, large amounts of hydrogen could be released into the containment and its integrity could be challenged by certain hydrogen combustion modes if no mitigating measures were available. International consensus [16] is that a local knowledge of the containment thermal-hydraulics is necessary to analyse the hydrogen mitigation methods and considerable efforts have been undertaken to better understand these phenomena, conducting a large number of experiments and then subjecting the test results to extensive analytical assessment. [16]. In this context, the prediction of the local distribution of hydrogen, steam and air is a key issue.

The containment cooling following a SA obtained through passive processes, such as natural circulation and condensation is an additional challenge, considering also as the European Utilities Requirements (EURs) asked for a rugged containment which has to be able to prevent damage from external hazards and internal effects, maintaining the characteristics of the last barrier between plant and environment. In particular, EURs pointed out the double concrete containment as the preferred solution for the future European plants. In relation to these new concepts, several facilities have been built to investigate the behaviour of passive containment heat removal systems and there has been interest to perform scaling analysis of such facilities and to provide data to qualify BE codes for licensing purposes.

These scaling analyses have often been based on the NRC HTTS methodology [7], specifically aimed at providing an effective method to evaluate the relative importance of physical processes in a specific system and to specify the required level of detail for scaling and investigation of each phenomenon. It is based on a development of the analysed system proceeding through a preliminary top-down analysis followed by a bottom-up process-and-phenomena approach. Applying this methodology, the system is progressively subdivided into subsystems, modules, constituents and phases. For each geometrical configuration of a phase, the governing equations describing the time behaviour of mass, momentum and energy balances can be established and handled to obtain a number of dimensionless parameters. This scaling methodology was applied [6], [8] for the analysis of mixing phenomena in presence of localised passive ESFs inside the double concrete APWR-1000 containment.

In the following the influence of a localised ESF on the thermal-hydraulic containment transient will be analysed. Of course this item is not the unique “open issue” on containment thermal-hydraulics but other aspects, as the weakness of the traditional system code – also if in new and revised versions – in describing phenomena as the velocity fields and the gas distribution inside the system or the heat transfer conditions outside the system, have to be considered.

4.1. Influence of a Localised ESFs

The Large Scale Containment Test Facility (LSCTF) programme, aimed to verify the performances of a new ICCS, was a joint effort between Westinghouse and ENEL with the co-operation of Pisa University [6]; the main items of experimental investigations were the natural convection phenomena and the effects of a light gas release or of a localised ESF on the resulting distribution.

The LSCTF (Figure 9) is a cylindrical vessel (6.1 m tall and 4.6 m diameter wide), originally utilised to simulate the AP600 containment [17] and, appropriately modified [18], represents a
1/10th scaled of the APWR-1000 double concrete containment. ENEL proposed a new innovative ESF to assure the containment heat removal [3], able to operate in a fully passive way and allows for long term cooling and depressurisation also of a double concrete containment and the final part of the wide assessment programme [18] was carried out on LSCTF. Each module of the ICCS contains an internal HX constituted by a finned tube bundle located inside the containment dome and an external HX, connected by an intermediate loop. Containment pressurisation is mitigated by the steam condensation on the tubes of internal HXs; latent heat causes a partial evaporation of the water flowing by natural circulation inside the intermediate loop and this two-phase mixture transfers its energy to external HXs, immersed in an external pool initially covering HX tubes, by condensation. During the tests, an auxiliary cooling system was located outside the vessel to provide and control the water flow to the primary side of the four internal HXs. This system allowed to impose HXs boundary conditions similar to those postulated during ICCS operation.

![Diagram](image)

**FIG. 9: LSCTF and main instrumentation locations.**

The key parameters selected to fix some reference operating conditions were the removed power and the atmosphere pressure, while the amount of helium injected was chosen to reproduce, with the proper scaling, the hydrogen generated by more or less extensive oxidation of the existing Zircaloy in the reference plant. Different ICCS heat removal efficiencies were investigated and the test matrix included ten experimental tests, each characterised by several steady state conditions [6]. The measurements provided extensive mappings of temperatures, velocities and steam/air concentrations. As example, the M06-Run34 test reproduces low power and medium pressure conditions, which should be typical of a long term SA scenario. Injection of helium starts during a very poor steaming phase, in which the steam blow-down is reduced to almost zero. After this, a new steam blow-down is applied to verify if natural circulation starts again.

The experimental results of the LSCTF test campaign, particularly focused on the characterisation of the atmospheric mixing inside the containment during a wide range of test conditions selected to be representative of plant states which are expected to occur during bounding accident sequences, confirmed a good performance of the proposed ICCS solution both in term of the cooling capability and in term of avoiding a dangerous stratification inside the facility. In all the investigated test conditions the light gas undergoes a near homogeneous distribution: this very good mixing experienced by the helium indicates that the localised ESF
promotes efficient convective motions inside the containment atmosphere, giving a positive indication for the related safety concerns.

The NRC HTTS methodology was applied [6], [8] to LSTCF tests by DIMNP in order to provide a systematic approach in the assessment of the scaling effects arising during these tests and to allow an extrapolation of the ESF results to the prototype plant. Scaling analyses were performed comparing the behaviour of two dimensionless parameters [8] deduced from the experimental data (label Exp.) with FUMO calculations for the facility and the real plant (label LSCTF Calc. and APWR Calc. respectively).

Figure 10 shows as the experimental “HX condensation number” is well reproduced by FUMO during the whole transient in the facility calculation. On the contrary, higher values are calculated in the scaled reproduction of the transient in the prototypical containment. Higher condensation rate prediction in the full-size containment is related to the dome pressure and temperature which result 20% higher in the reference plant, due to the lower ratio of the wall surface area to the free containment volume.

The comparison of the “densimetric Froude number” during the test first phase (Figure 11) shows a general agreement between the experimental results and the FUMO calculation for LSCTF, while higher values are evaluated for APWR, where a stronger density gradient exists. A different behaviour is shown during the second steady-state phase, when FUMO calculates a larger homogenisation of the LSCTF atmosphere as a consequence of the steam
injection, with a general under-estimation of the Froude number with respect to the experimental trend.

The distortion of these dimensionless parameters were also evaluated [8] as the ratio between the values related to the LSCTF facility (experimental or calculated ones) and those, obviously only predicted, for the prototypical containment. This methodology is a powerful tool able to provide major insight about the adopted code capabilities in simulation of experiments performed at different scaling levels and, at the same time, about the possibility to correctly extrapolate experimental results to a full-size plant.

Also in another test, the HX condensation number presents a similar distortion in the scaled facility, with respect to APWR, both in the FUMO simulated and in the experimental data, confirming that the down-scaled LSCTF represents in an approximate way [6] the phenomenon and FUMO does not introduce any additional distortion. On the contrary, the densimetric Froude number shows [8] a substantial distortion during the initial pre-heating phase both in the calculation and in the experimental data; a better experimental reproduction of the atmosphere density is progressively obtained during the test (a value very close to unity - no distortion - is obtained during the last phase, when a compensation among distortions relating to heat transfer to HX and heat losses through walls probably takes place). Globally, while the analysis of HX condensation number highlights a not perfect LSCTF scaling in reproducing this physical aspect of the APWR containment, not perfect scaling also highlighted by the FUMO code results, analysis of the densimetric Froude number emphasises that, after the reinjection of the steam, also due to possible favourable compensations, the LSCTF facility represents the reference plant behaviour in a better way than the code [6].

CONCLUSIONS

Several of the new plant concepts presently being considered are characterised by the use of innovative cooling systems, able to perform the core cooling action or to remove the heat released inside the containment following a hypothetical accident, through passive heat transfer mechanisms. In the previous sections the main characteristics of these new ESFs have been summarised and general design features were briefly outlined.

The strong coupling existing between PS and containment has been the main outcome of the discussion. The need for integrated calculation tools able to analyse the whole plant is the obvious conclusion on the side of accident analysis. The numerical results obtained by DIMNP with the development of a simplified PS module to be introduced inside the FUMO containment code, show the result of an effort aimed at investigating innovative reactors basing on a realistic approach. Nevertheless, the results obtained in SBWR and AP-600 applications show that indications on the overall plant behaviour can also be obtained. This information could be used as a basis for further and more sophisticated system analyses. But, in these new plants, there is an intrinsic limit in the pressure reduction, consequent to the absence of active systems: high temperature, and in consequence high pressure, is the only driving force for carrying outside the decay heat; so, in the long term, pressure trend is almost flat, and, unless some containment heat removal system is activated, can only decrease following the decay heat generation.

The second part of the paper has highlighted that the installation of such an ICCS in a containment system requires a reliable knowledge of temporal and spatial distribution of non-condensable gas inside the containment atmosphere, both to ensure the heat removing
efficiency of the system, and to guarantee that the local concentration of combustible gases does not reach dangerous values during the system operation. The LSCTF experimental results presented indicates that the localised ESF promotes efficient convective motions inside the containment atmosphere, giving positive indication for related safety concerns. A hierarchical methodology based on a general “top-down” scaling criteria, also intended to provide a guidance in the design of a new experimental programme, has been also shortly presented. This methodology has been applied to the experimental investigations of mixing phenomena inside the atmosphere of an insulated containment system, equipped with a localised ESF for the decay heat removal, representative of a double concrete containment system.

REFERENCES

SAFETY AND REGULATORY ASPECTS OF THE SMART REACTOR

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Abstract

A pressurized water reactor of 330 MW thermal power, named SMART (System integrated Modular Advanced Reactor), is under development at the Korea Atomic Energy Research Institute (KAERI) for seawater desalination application and electricity generation. The plant is expected to be installed near the population zone. Thus, the public around the plant should benefit from in-depth protection from the possible release of radioactive materials, and also the fresh water generated should be protected from radioactive contamination. In parallel with the design development, regulatory research is being conducted to identify safety concerns of the nuclear desalination plant. The safety concerns have been generally identified based on the new safety features of SMART design and current safety requirements. The items discussed in this study include the use of proven technology, enhanced containment function, event categorization and selection, effects of desalination plant, and maintainability of major components. As a result, this cooperative research in the design stage is expected to provide an opportunity for early resolution of the safety concerns, and eventually the licensing stability of the SMART design.

1. INTRODUCTION

The 330 MW thermal power of the integral type pressurized water reactor (PWR), which is called SMART (System integrated Modular Advanced Reactor), is under development for seawater desalination and electricity generation at Korea Atomic Energy Research Institute (KAERI) [1,2] in the Republic of Korea. Figure 1 shows the development structure of the SMART reactor.

FIG. 1. Research and Development Structure of SMART Reactor.
KAERI is a leading institute being responsible for developing the SMART design and some institutes and industry are involved in the related technology development. The Korea Institute of Nuclear Safety (KINS), a regulatory body, is also involved for SMART-related regulatory research, especially for identifying and resolving safety concerns for new design of the SMART reactor in the design development stage.

The SMART desalination plant will produce about 40,000 m³/day of potable water from about 10% of total thermal power along with about 90 MWe of electricity generation. Thus, the final plant products become electricity and fresh water unlike the commercial nuclear power plant (NPP) for only electricity generation. The conceptual design of the SMART has been completed and the basic design is currently in progress. Basically, it aims at improving the safety, reliability, and economics. To achieve these design goals, it adopts inherent and passive safety features and modular design concepts of main systems and components.

However, the SMART reactor and desalination facilities are expected to be located near population areas because of the high cost of heat/water transport system. It implies that the public and environment around the plant facility should be protected in depth from the possible release of radioactive materials under any plant condition. In addition, the fresh water generated and seawater should be protected from radioactive contamination for public safety. Thus, in this safety and regulatory research, which is carried out as a part of the SMART design development program, the safety characteristics and defense in depth strategy of the SMART desalination facilities are discussed. Also, based on the new design features and current safety requirements, general safety concerns to be considered in the design development stage are discussed. Because the SMART reactor has significantly different safety features from the current light water reactor (LWR) designs, the efforts to identify and resolve the safety concerns in the design stage would provide an early assurance of its safety.

2. 2. SAFETY AND REGULATORY RESEARCHES

2.1. Safety Characteristics of SMART Reactor

In the Republic of Korea, 16 commercial NPPs for electricity generation are in operation and 4 NPPs are under construction. The provisions on nuclear safety regulation and radiation protection for the NPP are systematically established in Atomic Energy Act [3] with enforcement decree and regulation of the act, and also the specific regulatory requirements and technical standards are prescribed in the Notice of the Minister of the Ministry of Science and Technology (MOST) [4]. The SMART reactor is developed based on the design and operational experiences on the existing NPPs, however it has also significantly different safety characteristics from the existing NPPs.

(1) First, the SMART reactor has another application to utilize the generated steam as a heat source for seawater desalination. Then, the SMART and desalination facilities are expected to be installed near the population area and coastal area because the fresh water is a final product as well as the electricity. Therefore, the highest level of safety standards for the SMART design is required to assure an adequate protection of public health and safety. Besides, the means to prevent the fresh water and seawater from the radioactivity contamination should be required in the plant design.

(2) Second, the SMART design adopts many unique design features to improve safety, reliability, and economics. They include the self-pressure control, boron-free operation, core decay heat removal using natural circulation, passive safety systems, safeguard
vessel added to the current containment structure, and so on. According to the principle of use of proven technology in nuclear installations, the safety performance of those new design features should be required to be demonstrated through analyses, tests, or experiences.

From these different safety characteristics of SMART desalination plant, additional safety requirements or guidance may be necessary for the comprehensive safety assessment and design, in addition to the current regulatory requirements for the LWR design.

2.2. Safety Requirements for SMART Design

The safety requirements for the SMART desalination plant could be classified into three parts, i.e. the safety requirements for the nuclear facility, desalination facility, and interface system. Basically, the existing safety and regulatory requirements for the LWR design and operation are applicable to the SMART reactor because there are lots of common design concepts. However, some safety requirements related to new safety features must be additionally considered to appropriately assess safety. Especially, the safety requirements for the interface systems and desalination facility must be newly established because there are no design and regulatory experiences of nuclear desalination plants in the Republic of Korea. Thus, the following top level of safety requirements are considered in the development of the regulatory requirements as well as the design of the SMART.

a. SMART reactor should be designed and operated with a higher level of safety than current LWR designs to ensure an adequate protection of public health and safety.

b. Radioactive contamination of the product water and seawater should be prevented under any normal and abnormal plant conditions.

c. The desalination plant should be managed and operated without severe impacts on nuclear plant during operation of NPPs.

d. The interface systems should be designed and operated to prevent the release of radioactivity into the desalination system and the ingress of brine into the steam supply system.

These safety objectives are similar to the technical and safety requirements for desalination plants in user requirements documents for SMR of IAEA [5]. The specific safety requirements and guidance for the SMART desalination plant will be identified and developed referring to the top level of safety requirements, based on current regulatory provisions and safety characteristics of the SMART design.

3. IMPLEMENTATION OF DEFENSE IN DEPTH CONCEPT

As a fundamental strategy to achieving the nuclear safety, the defense in depth philosophy has been applied in the design and operation of NPPs in the past. Recently, the IAEA has emphasized that a systematic and comprehensive implementation of defense in depth concept would improve the safety for future reactor designs [6].

The SMART design has several levels of protection and multiple barriers to prevent releases of radioactive materials and to minimize the possibility of failures leading to significant radiological consequences. The overall design concept of the SMART is shown in Fig. 2 and major design parameters are represented in Table I. Basically, SMART enhances the defense in depth strategy adopting inherent and passive safety concept, enhanced containment function, and simplified systems.
3.1. Enhancements of Physical Barriers

The defense in depth concept is generally structured in four specific barriers against the release of fission products and five specific levels of defense in depth to fully exploit these redundant barriers [7]. The current physical barriers in LWRs are in the form of the fuel matrix, fuel cladding, reactor coolant system (RCS) pressure boundary, and containment. In the SMART design, the defense in depth concept is additionally strengthened to protect the public and to prevent the fresh water from the radioactivity contamination. In the following sections, the safety enhancement items are discussed in detail in view point of five levels of defense in depth.

TABLE I. MAJOR DESIGN PARAMETERS OF SMART REACTOR

<table>
<thead>
<tr>
<th>Items</th>
<th>Design Values</th>
<th>Items</th>
<th>Design Values</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor type</td>
<td>Integral PWR</td>
<td>Design pressure (MPa)</td>
<td>17</td>
</tr>
<tr>
<td>Thermal power (MWt)</td>
<td>330</td>
<td>Operating pressure (MPa)</td>
<td>15</td>
</tr>
<tr>
<td>Boron reactivity control</td>
<td>No</td>
<td>Core inlet/outlet temperature (°C)</td>
<td>270/310</td>
</tr>
<tr>
<td>Vessel length/diameter (m)</td>
<td>9.8/3.96</td>
<td>Feedwater pressure (MPa)</td>
<td>5.2</td>
</tr>
<tr>
<td>Active fuel length (m)</td>
<td>2.0</td>
<td>Feedwater temperature (°C)</td>
<td>180</td>
</tr>
<tr>
<td>No. of fuel assembly</td>
<td>57</td>
<td>Steam pressure (MPa)</td>
<td>3.0</td>
</tr>
<tr>
<td>Core power density (w/cc)</td>
<td>62.6</td>
<td>Steam temperature (°C)</td>
<td>274</td>
</tr>
<tr>
<td>Refueling cycle (year)</td>
<td>&gt;3</td>
<td>Design life time (year)</td>
<td>60</td>
</tr>
</tbody>
</table>
3.2. Enhancements of Five Levels of Defense in Depth

3.2.1 Enhancements to level 1 of defense in depth

Measures at level 1 aim at confining radioactive material and minimizing deviations from normal operating conditions including reactor shutdown states. Level 1 mainly provides the design basis for protection against external and internal hazards. The SMART design enhances the level 1 measures through new design concepts such as integrated arrangement of the RCS as well as the traditional conservative design. The integrated arrangement simplifies the RCS layout and allows the elimination of large pipes so that the large break LOCA is excluded. Also, the reactor pressure vessel (RPV) is relatively larger in size and then sufficient coolant inventory is assured to cover the core in any LOCA condition. The RPV penetrations are also designed to minimize and the core power density is relatively low. In addition, the self-controlled pressurizer eliminates the active mechanism such as pressurizer heater and spray, and the canned type main circulation pump (MCP) excludes a pump seal failure. Additionally, a physical barrier between the nuclear system and the desalination system is designed, and a continuous radioactivity monitoring system is installed in the interface and desalination systems to monitor the leakage of radioactivity.

As a result, the SMART design provides sufficient design margin and response time before deviation from normal operating condition by low power density, large RCS inventory and simplified RCS design.

3.2.2 Enhancements to level 2 of defense in depth

Measures at level 2 aim at bringing the plant back to normal operating condition from abnormal operation, i.e. anticipated operational occurrences, as soon as possible. The SMART design inherently has a strong negative moderator temperature coefficient due to the absence of soluble boron and that yields beneficial effects on self-stabilization and limitations of reactor power. It also adopts reliable digital instrumentation and control systems that provide sufficient information on plant status to operators in the control room design, such as on-line core monitoring and protection system.

3.2.3 Enhancements to level 3 of defense in depth

Measures at level 3 aim at preventing an evolution toward a severe accident and confining the radioactive materials within the containment system following the postulated events. Level 3 mainly provides the design basis for engineered safety features and reactor protection systems to perform the intended safety functions. The SMART design enhances level 3 through passive safety features instead of traditional active features, as shown in Table II. The passive residual heat removal system (PRHRS) removes the core decay heat by natural circulation to the SGs and ultimately atmosphere through the PRHR cooldown tank. It removes decay heat for 72 hours after an accident without any operator action. The emergency core cooling system (ECCS) injects the water into the RCS by a compressed gas force and gravity to make up the RCS during the LOCA condition. In addition, the reactor overpressure protection system (ROPS) is used to rapidly decrease the high RCS pressure.

These design features are passively activated only by the process parameter variation after an isolation valve opening at certain set point. These passive systems will simplify the plant design and operation, and provide the high reliability of the systems.
TABLE II. SAFETY SYSTEMS OF SMART REACTOR

<table>
<thead>
<tr>
<th>Accidents</th>
<th>Safety Functions</th>
<th>Safety Systems</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design Basis Accidents</td>
<td>Reactivity control</td>
<td>Control rods</td>
</tr>
<tr>
<td></td>
<td>Coolant inventory control</td>
<td>Passive ECCS (2x100%)</td>
</tr>
<tr>
<td></td>
<td>Decay heat removal</td>
<td>Passive RHRS (4x50%) + Cooldown tank</td>
</tr>
<tr>
<td></td>
<td>Fission product containment</td>
<td>Safeguard vessel and containment</td>
</tr>
<tr>
<td></td>
<td>RCS overpressure control</td>
<td>ROPS + Internal shield tank</td>
</tr>
<tr>
<td>Severe Accidents</td>
<td>Containment heat removal</td>
<td>COPS + PRHR cooldown tank</td>
</tr>
<tr>
<td></td>
<td>Hydrogen control</td>
<td>Igniters, Recombiners</td>
</tr>
<tr>
<td></td>
<td>Leak tightness</td>
<td>Safeguard vessel and containment</td>
</tr>
<tr>
<td></td>
<td>Corium control</td>
<td>Internal and external shield tank</td>
</tr>
</tbody>
</table>

3.2.4 Enhancements to level 4 of defense in depth

Measures at level 4 aim at preventing core damage and mitigating the consequences of severe accidents. This level has been recently enhanced considering the severe accident prevention and mitigation features in the design stage for advanced light water reactors. In the SMART design, the containment overpressure protection system (COPS) with cooldown tank is used to reduce the atmospheric temperature and pressure in the containment. Additionally, the reactor vessel cooling system protects against core damage severe accidents, hydrogen igniters limit the hydrogen concentration, and a shielding tank for attenuation of fission products all feature as severe accident mitigation features. One of the most enhanced measures in this level will be safeguard vessel surrounded the RPV, installed in addition to the current containment. Therefore, the dual containment system of SMART will significantly limit the release of fission products into the environment for beyond design basis accident (DBA) as well as DBA.

3.2.5 Enhancements to level 5 of defense in depth

Measures at level 5 aim at mitigating the radiological consequences due to releases of radioactive materials into environment. So, the off-site emergency plan should be prepared based on realistic or conservative accident source terms. Because each level of defense in depth is improved in the SMART design as described above, the current emergency planning is expected to simplify, such as no notification or exercises. However, the simplification of the emergency plan is carefully considered in aspects of maintaining and enhancing the level 5 of defense in depth.

4. SAFETY CONCERNS OF THE SMART REACTOR

As discussed above, the SMART reactor is expected to achieve a high level of safety and reliability through inherent and passive means to accomplish its safety functions, and simplified reactor coolant systems. However, the SMART design also has many different safety features from the existing LWR designs. So, it is needed to identify safety concerns and resolve them in the design development stage for future licensing stability for its design. In this section, some general items to be considered in the safety aspects are discussed based on the conceptual design of the SMART reactor and current regulatory requirements. The detailed safety review as a licensing procedure will be conducted after filing the application for safety evaluation of the basic design.
4.1. Use of Proven Technology

SMART utilizes inherent and passive means to accomplish its safety functions, such as natural circulation, negative reactivity coefficient and passive residual heat removal. It also has just one reactivity control system using only control rods without a safety-grade boron injection system. The passive system will definitely provide greater simplicity and higher reliability of the systems. However, in principle, the technologies incorporated in the reactor design should be proven or qualified by experience, testing, or analysis and, if possible, the equipment should be designed according to the applicable approved standards. Therefore, the new design features important to the safety shall be introduced after through research and/or prototype testing at the level of component, system, or plant.

According to these requirements, the performance and reliability of the new safety features in the SMART design should be demonstrated through analysis, test programs, experience, or a combination thereof. Particularly, because the driving force in the passive fluid systems and their flexibility is lower in abnormal condition than those of the active systems, the safety performance and interdependent effects among the safety features should be demonstrated to be acceptable. In addition, sufficient data should exist on the safety features to assess the analytical tools used for safety analysis.

4.2. Event Categorization and Selection

All possible events in the NPP should be categorized according to expected frequency of occurrence, and acceptable criteria for each event category should be established in terms of core damage and dose limits. These events and sequences could be deterministically selected to supplement the insights from the probabilistic risk assessment depending on the design specific data. In the SMART design, the postulated initiating events and accident sequences could be different from those of the current NPP. For example, large break LOCA and coolant pump seal failure are excluded because of the integrated arrangement of RCS and canned motor MCP. Meanwhile, the turbine trip due to the desalination system failure, an inadvertent actuation of the passive system, or steam line break outside the safeguard vessel could additionally take place.

As a result, some of current DBAs could be excluded and some of new initiating events could be added in the safety analysis of the SMART design. Thus, all the possible events should be comprehensively categorized and analyzed according to the safety priority, and also the acceptable criteria for each event should be established based on current regulatory requirements. In this event analysis, the beyond DBAs should be also considered to confirm the sufficient safety margin of the SMART design. The beyond DBAs could be analyzed using best-estimate methods as the current analysis method.

4.3. Containment Function

The SMART design adopts a safeguard vessel surrounding the RPV to prevent releases of radioactive materials in addition to the current containment building. It also has a physical barrier between the nuclear system and the desalination system to protect the fresh water and seawater from the radioactive contamination. They will obviously provide an efficient barrier to confine the radioactive materials under normal and accident conditions. However, the safeguard vessel volume is markedly smaller than that of current LWRs, and there is no active safety-grade containment coolers or spray systems for cooling atmosphere of the safeguard...
vessel. Then, the safeguard vessel may be maintained at high pressure and temperature for a long time during accident condition.

As a result, there is a high potential for release of fission products and challenge to operability of safe-related equipment installed inside safeguard vessel. According to current safety requirements, the safe-related equipment in the safeguard vessel must be designed to provide reasonable assurance that they will operate in the accident environment. Therefore, although the safeguard vessel is expected to provide more efficient leak-tight function for the radioactive materials, the new concept of dual containment need to be evaluated in detail in the future.

4.4. Effects of Desalination Plant

In the nuclear desalination plant, radioactive contamination of the product water is not allowed under any plant condition. Also, the radioactive concentration of the brine to be used for desalination process should be maintained below an acceptable level. To meet these safety requirements, the release of radioactivity into the desalination facility and coastal sea should be prevented in the design and operation stage. In the SMART desalination plant, a physical barrier between the steam supply system and the desalination system and a continuous radioactivity monitoring system in the interface and desalination systems are considered. However, the nuclear system could be affected with an unstable operation of the desalination plant. In practice, a desalination facility with less qualified equipment and shorter lifetime could cause an unnecessary initiation of nuclear plant transients. Therefore, such transients in the nuclear plant should be limited and prevented, if possible.

As a result, it is essential that two the plants be designed and operated as an integrated plant in terms of hardware and software to share and exchange the essential information in each plant to minimize the effects of improper operation of desalination plant. Because there is no regulatory and design experience of coupling a nuclear plant with a desalination plant, these safety concerns will be reviewed in depth with further design details in the future.

4.5. Simplification of Emergency Planning

In general, offsite protective measures should be prepared when dose consequences are estimated above a low level of emergency protective guidelines, 1 rem for whole body and 5 rem for thyroid at the site boundary after any accident [8]. However, the SMART design with enhanced safety features, such as low power density, passive safety systems, and strengthening containment function, is expected not to exceed the low level protective guideline at the exclusive area boundary. The emergency planning zone is also expected to reduce as compared to the existing NPPs. It implies that some protective measures such as a rapid notification, detailed evacuation planning, and periodic exercise for the public may not be required by regulation or design.

This safety concern is strongly related to the accident analysis methodology and source terms following the core damage accidents. The accident source terms are usually used to calculate the radiological consequences and the emergency planning zone including the exclusive area boundary. That accident source is generally dependent on the performance of the fuel, reactor, and containment as well as the transport mechanism of fission products. Thus, new accident source terms may need to be developed based on the realistic behavior of fission products of SMART design. As a traditional approach, the current conservative source terms can be also used to evaluate the radiological consequences and emergency planning. Therefore, the
adequacy or simplification of the emergency planning and the related regulatory requirements are needed to review with the concerns on the use of realistic source term in the future.

4.6. Other Concerns

In the SMART design, digital instrumentation and control (I&C) is adopted as an advanced technology because of the ease of data processing. However, the reliability of software and the common mode failure of redundant equipment have been raised as an important safety issue in the digital I&C system of current NPP. This safety concern will be reviewed with design details and recent regulatory guidance.

The integrated arrangement of the RCS components provides a simple RCS design, but the internal components of reactor pressure vessel become much more compact. So, space to access for maintenance or test of main components may be not sufficient. In addition, the new design of steam generator with helically coiled tubes, control element drive mechanism with fine movement, and canned motor type MCP may require some different test and maintenance methods or acceptable standards. Therefore, the testability, inspectability, and maintainability of the integral SMART should be comprehensively reviewed to ensure the reliability of the major RCS components.

SMART is designed to operate with fewer operators than current NPP. The start up, shutdown, and refueling processes are also different from the existing NPP. So, the operational safety will be reviewed with design details in the future.

CONCLUSIONS

The integral pressurized water reactor of 330 MW thermal power, named SMART, is under development at KAERI for seawater desalination and electricity generation. The final product is electricity and fresh water, and the plant is expected to be installed near the population areas and coastal seas. Thus, the public around the plant should be protected in-depth from the possible release of radioactive materials, and the fresh water and seawater should be protected from radioactive contamination.

In parallel with the SMART design development, regulatory research is being performed to identify and resolve the safety concerns related to the SMART desalination plant. In this study, the safety characteristics of the SMART design and some safety concerns identified in the development stage were discussed. The general safety concerns are the use of proven technology, event categorization and selection, strengthening containment function, effects of desalination plant, maintainability of major components, and so on.

As a result, cooperative research with the regulatory body in the design stage is believed to provide an opportunity to early resolve the safety concerns and eventually the licensing stability of the SMART design. In the future, the specific safety concerns related to the SMART-unique design will continue to be identified and, if necessary, the related regulatory requirements and guidance will be developed.

ACKNOWLEDGEMENT

This study was been carried out under the nuclear research and development program supported by Ministry of Science and Technology (MOST) of the Republic of Korea.
NOMENCLATURE

COPS: Core Overpressure Protection System  DBA: Design Basis Accident
ECCS: Emergency Core Cooling System  IAEA: International Atomic Energy Agency
KAERI: Korea Atomic Energy Research Institute  KINS: Korea Institute of Nuclear Safety
LOCA: Loss of Coolant Accident  LWR: Light Water Reactor
MCP: Main Circulation Pump  MOST: Ministry of Science and Technology
NPP: Nuclear Power Plant  PRHRS: Passive Residual Heat Removal System
PZR: Pressurizer  RCS: Reactor Coolant System
ROPS: Reactor Overpressure Protection System  RPV: Reactor Pressure Vessel
SG: Steam Generator  SMR: Small and Medium Sized Reactor
SMART: System Integrated Modular Advanced Reactor

REFERENCES

**CAREM-25 ACCIDENT ANALYSIS**

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**Abstract**

This paper describes the response of the CAREM reactor prototype, CAREM-25, and its safety systems to representative accidental initiating events. A nodalization of the primary circuit including the steam generators was developed for RELAP5 and RETRAN02 codes to perform the accident analysis. Also a simplified two zone non-equilibrium model was developed to calculate long term reactor behavior. Secondary Shutdown, Residual Heat Removal and Emergency Injection systems were also modeled in detail. The reactor steady state was calculated at full power. The results agree quite well with design values, and this condition was used as reference for the accident analysis. Several initiating events were considered for the accident analysis. They were grouped into Reactivity Insertion, Loss of Heat Sink (LOHS) and Loss of Coolant Accidents (LOCA). As there are no primary pumps, Total Loss of Flow Accident (LOFA) is not applicable in this case. Results show that the design requirements are verified and the reactor reaches a safe condition, without need of active intervention. As a general conclusion, it could be said, that due to the large coolant inventory in the primary circuit, the system has a large thermal inertia and long response time in the event of transients or severe accidents.

1. **CAREM SAFETY SYSTEMS**

The CAREM-25 [1] is an indirect cycle integral reactor of 100 MWth with some distinctive and characteristic features that greatly simplify the reactor design and also contribute to a higher level of safety. The primary system is integrated, cooling is by natural circulation and self-pressurised. Safety Systems (SS) are based on simple, passive features and must guarantee no need of active actions to mitigate accidents for a long period. They are duplicated to fulfill the redundancy criteria. Table 1 shows the safety functions performed by the different safety systems. A diagram is shown in Figure 1.

**TABLE I. CAREM-25 SAFETY SYSTEMS FUNCTIONS**

<table>
<thead>
<tr>
<th>Safety Function</th>
<th>Safety System</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactivity Control</td>
<td>First Shutdown System: Safety control rods</td>
</tr>
<tr>
<td>Primary Pressure Limitation</td>
<td>Safety Relief valves</td>
</tr>
<tr>
<td>Primary Depressurization</td>
<td>Emergency Condenser</td>
</tr>
<tr>
<td>Primary Water Injection</td>
<td>High Pressure: Second Shutdown System</td>
</tr>
<tr>
<td>Residual Heat Removal</td>
<td>Low pressure: Emergency Injection System</td>
</tr>
</tbody>
</table>

The shutdown system must be diversified to fulfill regulatory requirements. The First Shutdown System (FSS) consists of neutron absorbing rods, driven hydraulically by a device that is located inside the reactor pressure vessel (RPV) dome. The absorber rods drop into the core by gravity action when required. The Second Shutdown System (SSS) injects borated water into the primary circuit in case of FSS failure. The system consists in a pressurized tank that contains a boron solution. Each tank is connected, from the top, to the RPV steam dome, to equalize pressure and from the bottom to the core inlet, through a pipe located inside the RPV in the cold leg. Whenever the system is demanded, the solution is injected by gravity
driven force. It is required that the discharge time should belong enough to get boron homogenization in all the primary circuit. The FSS and SSS are designed independently to stop the nuclear reaction and maintain the core in a safe condition during cold shutdown. The SSS has also another safety function that is to inject water into the primary system at any pressure in case of Loss of Coolant Accidents (LOCA).

The Residual Heat Removal System (RHRS) consists of an arrangement of tubes connected to the RPV dome by two common heads. The vapor, which comes from the dome, condenses inside the tubes, transferring heat from the primary system to a pool located in the upper level of the containment. The condensed fluid returns to the RPV. This system has a heat removal capacity to reduce the primary circuit pressure to avoid safety valves opening and to lead the reactor to hot shutdown conditions. The nominal power is 2 MW, removed by means of 2 independent modules.

The Emergency Injection System (EIS) is an accumulator that discharges coolant in case of LOCA when primary circuit pressure falls below 1.5 MPa. It provides water enough to maintain the core covered for a long period.

2. ACCIDENT ANALYSIS CAREM REPRESENTATION: DETAILED ODALIZATION

2.1. Primary circuit

A one-dimensional nodalization of the CAREM-25 primary circuit including the steam generators was developed for RELAP5 [2] and RETRAN02 [3] codes to perform the deterministic accident analysis.
The fission energy is calculated from the solution of the point kinetic equations coupled with the fission product decay curves. Contributions to the system reactivity include feedback effects caused by Doppler and water density changes (that includes temperature and void fraction variations). Also reactivity from the FSS or SSS is considered. The fuel elements are modeled as heat structures, representing the fuel, gap and cladding.

The steam dome is modeled with two volumes. One represents the liquid-vapor-stratified region that includes the upper chimney, the SG inlet and a portion of the vapor zone. The other represents the upper zone of the RPV that contains only vapor and where the connections of the safety systems steam lines are located. Vapor condensation on the absorbing elements hydraulic device, RPV wall and on the liquid surface is modeled. These mechanisms are included as they govern the vapor generated in the core, which travels along the chimney (riser) to the dome. This amount of vapor affects the hot leg density and therefore the buoyancy forces that, together with pressure losses, determine the primary circuit mass flow. A more detailed nodalization of the dome is being developed and analyzed.

Down-comer and chimney are modeled using a fine mesh discretization in order to follow properly the thermal fronts, minimizing numeric diffusion. On the other hand, components such as the lower plenum are modeled with a unique volume to represent the mixture effect of water, before coming into the core.

One equivalent steam generator (SG) represents the twelve SGs with a flow and heat transfer area equal to the sum of all of them. SG primary and secondary sides are modeled in detail. The secondary system is represented by mass flow (SG feed-water) and pressure (turbine inlet) boundary conditions, which can eventually be changed with time.

RPV and in-vessel structures are modeled to evaluate their thermal capacitance during transients and thermal shortcuts between different volumes.

The feed and bleed of the Absorber Rods Hydraulic Device and of the Purification and Control Volume System is included through boundary conditions.

### 2.2. Safety systems

FSS is modeled as reactivity as function of time, evaluated considering single failure of the safety absorber rods. It is triggered when any of the set point of the trip parameters is verified.

In the RELAP nodalization the SSS and RHRS are modeled in detail. SSS model includes the tank, valves and piping that are connected to the reactor dome. For the RHRS, the condenser tubes, headers, piping and pool are modeled. Both systems can also be modeled through boundary conditions, depending on simulation or analysis requirements, by means of specifying fluid properties in the extraction and injection points. For SSS, boron concentration must be also provided and for RHRS the vapor condensation rate is evaluated through a correlation that relates the power removed as function of the RPV pressure. EIS is modeled with an accumulator component. The safety valves are also modeled.
2.3. Simplified model for long term simulations

A simplified two-zone (vapor and liquid) non-equilibrium model, VERT code [4], was developed to calculate long term reactor behavior in the event of LOCA or LOHS where detailed codes calculations are prohibitively. Also, due to the expected phenomenology it is reasonable to make simplifying hypotheses such as assuming homogeneous temperatures in the liquid and vapor zones. The SSS and RHRS are modeled as boundary conditions of mass flow and removed power, respectively, while an adiabatic accumulator represents the EIS.

2.4. Steady state

Reactor steady state at full power was calculated considering heat losses from the RPV to the containment and vapor condensation on the absorbing rods hydraulic device. The results are shown in Table 2. Figure 3 shows the liquid and saturation temperature distribution along the primary circuit from the core inlet, following fluid circulation direction. The cold leg temperature is 557 K and saturation temperature at dome pressure is 599K. A temperature increase along the chimney can be observed and is due to condensation of the sub-cooled vapor generated in the core.
TABLE II. REPRESENTATIVE VARIABLES VALUES CALCULATED IN STEADY STATE CONDITION

<table>
<thead>
<tr>
<th>Variable</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>RPV dome pressure</td>
<td>12.25 MPa</td>
</tr>
<tr>
<td>Mass flow</td>
<td>410.6 Kg/s</td>
</tr>
<tr>
<td>Liquid level over SG active zone</td>
<td>1.04 m</td>
</tr>
<tr>
<td>Thermal power</td>
<td>100.4 MW</td>
</tr>
</tbody>
</table>

FIG. 3: Steady state: liquid and saturation temperature distribution along primary circuit.

3. ACCIDENT ANALYSIS

Several initiating events were considered for the accident analysis. They were grouped into Reactivity Insertion (RIA), Loss of Heat Sink (LOHS) and Loss of Coolant (LOCA) accidents. As there are no primary pumps Total Loss of Flow Accident (LOFA) is not applicable in this case.

3.1. Reactivity insertion accidents

As the innovative hydraulic control drive for the Fast Shutdown System and the Adjust and Control System is located inside the RPV, the Rod Ejection Accident is avoided and only inadvertent control rod withdrawal transients are postulated. For the present analysis the maximum control rod worth at normal withdrawal velocity is adopted, that is a reactivity ramp of 0.018 $/s$. It is assumed conservatively that the rod is initially fully inserted considering the reactor at full power. Two scenarios considering FSS or SSS actuation were modeled.

As a consequence of the positive reactivity insertion the reactor power increases. After 10 s the high power set point (108 MW) is reached and the FSS is demanded. After a delay conservatively assumed to be 1 s, safety rods are introduced, and the power diminishes abruptly reaching decay values, as it is shown in Figure 4.

When failure of FSS is postulated SSS trip is produced at 12.7 s when the very high power set point is verified (115%). The equalization lines valves are opened and after a delay of 1 s the discharge lines valves are opened, beginning the boron solution injection, through a pipe, into core inlet. Approximately 8 s later the boron solution arrives in the core, reducing the power to decay values. The average mass flow of boron solution injected is 2.5 kg/s and the elapsed discharge time is 800 s approximately, verifying the design criteria in order to provide boron homogenization in the primary circuit.

The results of the accident with FSS actuation show that safety margins (see Figure 5) are well above the critical values (DNBR and CPR). In the case with FSS failure and SSS success, safety margins reach minimums higher than 1.1 for a very short period and therefore, no core damage is expected.
3.2. LOHS

CAREM-25 reactor and Residual Heat Removal System (RHRS) behavior to mitigate a loss of heat sink accident (LOHS) is described and the correspondent design requirements are analyzed.

The RHRS design requirements, to be fulfill for this accidental sequence, are:

- **Short-term**: primary circuit pressure must remain below safety valves opening set point and condensers must not flood in order to avoid instabilities.
- **Long-term**: must allow attainment of the hot-shutdown condition (primary circuit pressure below 2.3 MPa).

### 3.2.1 Short term analysis

Short-term reactor behavior was simulated using RELAP5 with a detailed nodalization of the primary circuit and RHRS. Engineering or safety factors were established in order to have a conservative simulation where reactor pressurization is enhanced by means of reducing energy storage capacity or increasing power imbalance. In this sense the main considerations assumed are:

- FSS is trigged by primary circuit parameters and not by secondary system ones, with a delay of 1s
- In order to reduce vapor condensation in the dome no heat losses to the containment are included and the RPV dome structures are not modeled
- The primary system mass flow is increased by 20% in order to increase the downcomer temperature, and thus to reduce primary system thermal or energy storage capacity
- SG feed water is interrupted abruptly.
It is important to notice that this accidental sequence is more severe than a blackout one because FSS and RHRS will be demanded just after the cut-off of the external electrical power supply.

SG feed water is interrupted at \( t = 0 \). Due to the loss of the heat sink, the down comor temperature increases, leading to a decrease in the water density. Therefore the steam in the dome is compressed and pressure increases, Figure 6. When it reaches 13 MPa, the FSS is tripped. As a consequence of the power reduction and the subsequent sub-cooled void collapse, there is a temporal pressure decrease. As there is no power removal, the temperature goes on increasing in the downcomer and in the whole circuit and the pressure increases again. In spite of the coolant heating all the liquid inside the RPV remains in the subcooled condition because of the pressurization. When RHRS trip set point (13.8MPa) is reached, this system is tripped. Steam is removed from the dome and condenses inside tubes located in a pool, to return, finally, to the RPV. As primary water is subcooled there is no steam reposition into the dome and therefore a sharp decrease in pressure is observed. Then liquid in all the primary system reaches saturation and the pressure stops falling. From this moment on, pressure begins to be ruled by saturation conditions, steam reposition into the steam dome (generated in the core) and steam condensation in the RHRS.

![Figure 6: LOHS short-term: pressure evolution.](image)

It is verified that, with the RHRS removing 2 MW at nominal conditions, the maximum primary pressure after RHRS trip is 1Mpa below the safety valves opening set point. In the case of a hypothetical RHRS failure, it would take around 50s to reach the safety valves opening pressure value. No flooding in the condenser return line is observed and so no flow instabilities are expected.

3.2.2 Long-term analysis

As long as the primary circuit is in the saturated condition in the medium and long term, reactor performance is simulated with VERT model. This condition is expected during RHRS
operation. Also RPV and internals structures are included in the model. The RHRS is represented by means of an efficiency curve of removed power vs. RPV pressure. This is previously generated with the RELAP detailed RHRS model for different RPV pressure values in the steady state condition.

Like in short-term simulation, SG feed water is conservatively interrupted at $t = 0$ s. Due to the loss of heat sink, pressure increases. When it reaches 13 and 13.8 MPa, FSS and the RHRS are demanded, respectively. When the power removed by the RHRS is greater than the decay power, pressure begins to decrease, as it can be observed in Figure 7. The reactor reaches hot-shutdown in approximately 32 hrs in a safe condition, verifying the design requirements.

![FIG. 7: LOHS long-term: pressure evolution.](image)

3.3. LOCA: parametric analysis

A parametric study considering several pipes break diameters (12.7, 19.0, 25.4, 38.1 and 50.8 mm) was performed to analyze CAREM-25 reactor response to LOCA. The 50.8 mm break diameter is outside the design bases as a single rupture, but it was considered to model in a conservative way the coolant looses through both ends of a 38.1 mm inner diameter pipe (i.e. SSS lines). A discharge coefficient ($C_d$) equal to one was used. A total SG feed-water loss and the Purification and Volume Control System unavailability are postulated when SCRAM occurs for a conservative calculation. Short-term modeling was done with RETRA02 code and long-term with VERT code.

Two analyses were carried out in order to verify, in the first one, the safety systems performance assuming the successful operation of one of the redundancies and in the second, the inherent reactor response postulating FSS success and a hypothetical failure of all the safety systems related with reactor cooling.

Table 3 shows maximum loss of coolant flow, reactor power, safety systems trip parameter, its time of activation and core uncovering time for the different breaks analyzed. For break equivalent diameters equal or lower than 12.7 mm, the reactor pressure diminishes slowly and
FSS trip is produced due to the low water level inside the RPV. For medium size diameters, the RPV depressurization is greater, and the FSS is tripped by low pressure (11.7 MPa). For diameters around 38 mm and higher, the pressure decrease causes the chimney water to boil, leading to the consequent increase in the buoyant force, so the mass flow increases originating a core overcooling. As core coolant average temperature diminishes, fission power rises due to the negative coolant temperature reactivity coefficient. This coupled phenomena between thermal hydraulic and neutronic behavior causes the reactor power to rise significantly, and the FSS trip is due to the detection of high power or high neutron flux (108% FP). This effect is self-limiting without reaching core thermal limits.

TABLE III. LOCA PARAMETRIC STUDY: RESULTS

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>12.7</td>
<td>½</td>
<td>2.32</td>
<td>Low water level</td>
<td>666</td>
<td>103.3</td>
</tr>
<tr>
<td>19.0</td>
<td>¾</td>
<td>5.22</td>
<td>Low pressure</td>
<td>51</td>
<td>102.3</td>
</tr>
<tr>
<td>25.4</td>
<td>1</td>
<td>9.28</td>
<td>Low pressure</td>
<td>23</td>
<td>104.3</td>
</tr>
<tr>
<td>38.1</td>
<td>1 ½</td>
<td>20.77</td>
<td>High power</td>
<td>4</td>
<td>108.2</td>
</tr>
<tr>
<td>50.8</td>
<td>2</td>
<td>36.52</td>
<td>High power</td>
<td>3</td>
<td>108.4</td>
</tr>
</tbody>
</table>

TABLE IV. LOCA RESULTS OF SEQUENCES WITH AND WITHOUT ACTUATION OF SS RELATED WITH REACTOR COOLING

<table>
<thead>
<tr>
<th>Break Hydraulic Diameter [mm]</th>
<th>With SS</th>
<th>Without SS</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>SS and RHRS Trip [hh:mm]</td>
<td>EIS injection trip [hh:mm]</td>
</tr>
<tr>
<td>12.7</td>
<td>00:39</td>
<td>5:60</td>
</tr>
<tr>
<td>19.0</td>
<td>00:16</td>
<td>2:20</td>
</tr>
<tr>
<td>25.4</td>
<td>00:08</td>
<td>1:15</td>
</tr>
<tr>
<td>38.1</td>
<td>00:03</td>
<td>0:30</td>
</tr>
<tr>
<td>50.8</td>
<td>00:02</td>
<td>0:15</td>
</tr>
</tbody>
</table>

Safety Systems, related to core cooling (SSS, RHRS and EIS), are demanded by the Reactor Protection System in case of LOCA, when Emergency Cooling Criteria (ECC) is verified. That is when very low RPV water level “or” very low RPV pressure “or” high level of radioactivity in the containment, is verified with a 2 of 3 logic. As the containment is not modeled, ECC is considered to occur conservatively when very low RPV water level “and” very low RPV pressure are verified. So when this criteria is verified SSS is demanded and starts the boron solution drainage. Simultaneously the RHRS is demanded, enhancing RPV depressurization by means of condensing vapor from the reactor dome. When pressure falls below 1.5 MPa, EIS starts injecting water into the primary system. First it is initially injected with a massive flow rate. When 5 m³ are injected, the system switches to a lower mass flow, around 0.4 kg/s. Table 4 summarises the trip time for cooling related SS, minimum water level over the core reached during the first moments after EIS trip and core uncovery time -if
it is produced before 55hs-, or water level over core active zone -if it is produced later-. The analysis shows that there is not an early core uncover, therefore there is no need of a high-pressure injection. The reactor remains cooled during that period.

Collapsed water level inside the RPV is shown in Figure 8 for the different break diameters analyzed. When the EIS starts, the mass flow injected into the RPV is greater than the loss through the break, then the water level begins to increase quickly. When the EIS switches to reduced injection, water level continues rising but at a lower rate. The injection finishes around 24 hs, depending on break area. Core uncovery begins after 48 hs. Increasing EIS accumulator volume, this time could be extended to longer periods.

Finally it the reactor inherent response to LOCA was analyzed, considering FSS success and failure of the SS related with core cooling. In the last column Table 4 is shown the core uncovery time. Due to the large water inventory over the core and the small penetrations diameters through the RPV, the core unfolds after 2.5 hs, this means that large LOCA phenomenology is not present and allowing enough time for SS recovery actions.

**FIG. 8: RPV collapsed water level.**

**FINAL CONCLUSIONS**

Results show that the design requirements are verified and the reactor reaches a safe condition, without need of active actions. As a general conclusion, it could be said, that due to the large coolant inventory in the primary circuit, the system has large thermal inertia and long response time in case of transients or severe accidents. The most remarkable aspect is that in the event of LOCA with the failure of SS related with reactor cooling, core uncovering occurs after 2.5 hrs.
REFERENCES


Abstract
In three decades that the Karachi Nuclear Power Plant (KANUPP) has been in operation considerable
development has taken place in analytical tools besides understanding of the phenomena affecting safety of
PHWRs. This paper summarizes and highlights the salient features of the extensive revamp of safety analysis,
which became necessary particularly from the perspective of the phenomena that were either not addressed or
altogether unknown at the time plant was built.

Included as such are the analyses performed for the evaluation of effect of impurities in heavy water coolant and
moderator on void reactivity. Additionally the impact of fuel string relocation in the aftermath of a postulated
LBLOCA has been examined from the viewpoint of qualification of shut down system.

In conclusion the results of presented analyses are used to demonstrate continuity of safe operation of this SMR
in the present times and projection of the same beyond its design life of 30 years.

1. INTRODUCTION
The Karachi Nuclear Power Plant is a 137 MWe Pressurized Heavy Water Reactor of
CANDU type. It was commissioned in 1971. The plant has to-date fed ~ 9900 GWhs of
electricity to the Karachi electricity grid and is as such contributing to the comfort and well
being of millions of this mega city dwellers. Being a SMR it also affords essential nuclear
power plant training of operators and maintainers besides facilitating tests and measurements
needed for the development of indigenous nuclear fuel and mechanical components for the
non-conventional plant systems.

Since reactors of this type are characterized by positive void coefficient of reactivity their
safety was the subject of in depth investigation particularly in the aftermath of Chernobyl
Accident. In consequence, the understanding and interpretation of the effect of various reactor
physics characteristics affecting their safety have undergone in-depth review [1]. This paper
examines not only those safety related features, the interpretation of which underwent
considerable modification on account of progressively evolving understanding of the
phenomenon involved, but also those which were not known at the time plant was designed
and built.

It is now known that the void reactivity not only depends on the fuel burnup as originally
thought but the impurities in the basic cell constituents also contribute in aggravating the
same. Notwithstanding this, an altogether new phenomenon of fuel string relocation in the
aftermath of LBLOCA came to be recognized of late and introduced an element of
uncertainty as to the adequacy of special safety systems in coping with the consequences of
the postulated maximum credible accident.

The CANDU’s have however a comparatively long prompt neutron lifetime and a large
delayed neutron fraction on account of the photo-neutrons that are present. Together they play
a dominant role in slowing down the effect of power rise in a reactivity excursion. Notwithstanding these inherent positive features, improved understanding of the progression
### TABLE I. SOME NUCLEAR DESIGN DATA FOR THE KANUPP REACTOR

#### Gross Properties of Core

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Radius</td>
<td>1.912 m</td>
</tr>
<tr>
<td>Number of Fuel Channels</td>
<td>207 (Channel G-12 Removed)</td>
</tr>
<tr>
<td>Number of Bundles per Channel</td>
<td>11</td>
</tr>
<tr>
<td>Reactor Length</td>
<td>4.877 m</td>
</tr>
<tr>
<td>Coolant Temperature (average)</td>
<td>269.5 °C</td>
</tr>
<tr>
<td>Moderator Temperature</td>
<td>60 °C</td>
</tr>
<tr>
<td>Fuel Temperature (core average)</td>
<td>687 °C</td>
</tr>
<tr>
<td>Total Fission Power</td>
<td>456.8 MW</td>
</tr>
<tr>
<td>Heat Removed by Coolant</td>
<td>432.8 MW</td>
</tr>
<tr>
<td>Unit Electric Output</td>
<td>137.2 MW</td>
</tr>
</tbody>
</table>

#### Channel Ratings

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Time Average Maximum Channel Power</td>
<td>2.8 MWth</td>
</tr>
<tr>
<td>Time Average Maximum Bundle Power</td>
<td>453 kW</td>
</tr>
</tbody>
</table>

#### Fuel Bundle and Channel Details

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel</td>
<td>Natural Uranium Dioxide</td>
</tr>
<tr>
<td>Number of Elements per Bundle</td>
<td>19</td>
</tr>
<tr>
<td>Element Outside Diameter</td>
<td>15 mm</td>
</tr>
<tr>
<td>Pellet Outside Diameter</td>
<td>14 mm</td>
</tr>
<tr>
<td>Bundle Length</td>
<td>495.3 mm</td>
</tr>
<tr>
<td>Average Uranium Dioxide Density</td>
<td>10.6 Mg/m³</td>
</tr>
<tr>
<td>Uranium Dioxide Area</td>
<td>30.36 cm²</td>
</tr>
<tr>
<td>Coolant Area</td>
<td>19.98 cm²</td>
</tr>
<tr>
<td>Uranium Dioxide Weight per Bundle</td>
<td>15.195 kg</td>
</tr>
<tr>
<td>Zircaloy Weight per Bundle</td>
<td>1.379 kg</td>
</tr>
<tr>
<td>Pressure Tube Material</td>
<td>Zr-2.5% Nb</td>
</tr>
<tr>
<td>Inner Radius of Pressure Tube</td>
<td>41.4 mm</td>
</tr>
<tr>
<td>Calandria Tube Material</td>
<td>Zr-2</td>
</tr>
<tr>
<td>Inner Radius of Calandria Tube</td>
<td>50.6 mm</td>
</tr>
</tbody>
</table>

of voiding and delayed neutron effectiveness in enhancing the worth of shutdown system and reducing the void reactivity worth at the same time too helped suppressing the amplitude of power pulse in the post LBLOCA scenario.

This paper addresses the concerns raised and explains how improved understanding of heat transport and shutdown systems response could still ensure safety against the design basis accident.
2. BASIC REACTOR SYSTEMS

The essential nuclear design features of KANUPP are summarized in Table I. A description of the main process system and its components together with that of the relevant special safety system has been included to the extent it is necessary for the understanding of subject under discussion. Moreover description of only those features is provided which are different from modern CANDUs.

2.1. Reactor Core

The reactor core contains an array of 207 pressure tubes (channel G12 has been removed) within a cylindrical vessel approximately 1.9 m in radius and 4.9 m long, called the calandria. Each pressure tube contains 11 fuel bundles of the 19 element type, surrounded by a calandria tube with the annular space filled by carbon dioxide. The calandria contains the heavy water moderator and reflector. High pressure (~ 10 Mpa) and temperature (~ 276 °C average) heavy water is circulated within the pressure tubes for the removal of heat generated within the fuel. The moderator is isolated from the heat transport coolant and it is operated at low temperature (~ 70 °C) and slightly above atmospheric pressure.

2.2. Shutdown System

The reactor is equipped with a single, moderator dump shut down system. The heavy water moderator in the calandria is supported by differential helium gas pressure. This arrangement permits reactivity control by moderator level variation besides providing for rapid shutdown by dumping the moderator into the dump space, below the calandria, following gas pressure equalization. The shutdown system is actuated upon detection of an unsafe process condition that causes at least one trip parameter in each of two-out-of-three, SDS logic channels to exceed its set point. When a trip is called for, the protective system rapidly shuts down the reactor by opening the dump valves, causing the moderator to drain from the calandria.

3. PHYSICS ASPECTS AFFECTING SAFETY

In order to develop an understanding of the subject it is necessary to have a brief description of the physics characteristics and phenomena affecting safety of CANDU PHWRs. While the kinetic parameters, responsible for the rise and decay of power in a reactivity excursion are only referred to, the effect of voiding is dealt with in some detail as it is the only coefficient of importance in the time scale of interest.

3.1. Reactor Kinetic Parameters

3.1.1 Prompt Neutron Life Time ($\lambda^p$)

The CANDU reactors are characterized by a prompt neutron lifetime ~ 0.9 s. Compared to the value of 0.03 s in light water reactors, it is 30 times longer. For reactivity transients well below prompt critical, the effect of this difference is small. However, for reactivity insertions at or near, prompt critical the larger $\lambda^p$ retards a power pulse significantly. In CANDU reactors this is an important consequence, since it reduces the demands placed on the shutdown system design to relatively modest performance requirements.
3.1.2 Delayed Neutron Fraction ($\beta$)

Detailed treatment of any reactivity excursion accident in a CANDU PHWR involves 24 neutron precursors (six from each of the fissionable isotopes U-235, U-238, Pu-239 and Pu-241) and nine families of gamma precursors for photo neutrons produced in heavy water. The photo neutrons constitute 17% of the total delayed neutron source in CANDU reactors. A larger value of delayed neutron fraction implies higher margin to prompt criticality in these reactors.

3.2. Core Voiding

The voids in the reactor may be formed, if either the moderator or heat transport system boils. An increase in heat generation, a decrease in coolant flow or a reduction in pressure due to a failure could cause this in the system. Generally, if a reactor is over moderated i.e. with moderator/fuel ratio in excess of that required to just thermalize the neutrons, then a void formed in the moderator or in heavy water heat transport fluid, will cause an increase in reactivity.

The coolant void reactivity is positive in CANDU reactors. The fast neutrons are moved from their fission spectrum to lower energies by scattering with coolant besides the moderator atoms. When this (coolant) is removed fast fission is more likely and fast-fission factor is increased. Moreover the loss of coolant results in a small decrease in the cell slowing down power and hence a small increase in resonance absorption in the fuel. However this is overcompensated by the much larger increase in the resonance integral (due to the increase in the mutual shielding between elements) the net result is a reduction in the resonance absorption and therefore an increase in $p$ (resonance escape probability).

Moreover, when the coolant is hot under normal steady state full power conditions, it tends to harden the spectrum of thermal neutrons coming from the moderator. When the coolant is lost this hardening effect no longer exists and there is a large gain in reactivity.

3.2.1 Effect of Fuel Irradiation on Core Voiding

In CANDU reactors the void reactivity coefficient is larger for fresh than for equilibrium fuel because when coolant is removed from the channel, the fuel experiences a softer neutron spectrum (a soft neutron spectrum comprises neutrons with lower temperature). A softer spectrum causes an increase in $\eta$ for fresh fuel where there is no plutonium. However for equilibrium fuel where there is a significant amount of plutonium a softer spectrum causes a decrease in $\eta$. The major reason for this is the significant decrease of the fission cross sections of plutonium-239 and plutonium-241 (fission resonances at higher thermal neutron energies) with decreasing neutron temperature. In the irradiated fuel due to the presence of plutonium the hardening of neutron spectrum already is a positive effect and reactivity change is relatively less on disappearance of coolant. This is the reason as to why for a fresh core the void reactivity is larger than it is for irradiated core and decreases with irradiation.

The design values of full core void reactivity calculated using MOLAR and FOG for the fresh KANUPP core was 11.4 mk but only 3.4 mk for the equilibrium core.
3.2.2 Effect of Absorbers on Coolant Void Reactivity

The magnitude of the coolant void reactivity is sensitive to the concentration of absorbers in the lattice cell. The two most important of these are boron in the moderator and ordinary water in the heavy water coolant i.e. downgraded coolant. The impurity in pressure and/or calandria tube would likewise produce a similar effect. As an example, the void reactivity is increased by approximately 0.7 mk if the absorption cross sections of the pressure and calandria tubes are increased by ten percent.

The void reactivity coefficient is higher with down graded coolant because the removal of downgraded coolant results in an increase in the thermal utilization factor. Since light water is much more absorbing than heavy water, the value of $\Delta f$ is sensitive to the isotopic purity of the coolant.

4. FUEL STRING RELOCATION AND ITS IMPACT ON SAFETY

The phenomenon of fuel string relocation and its effect on the safety has been described in an exclusive paper earlier [2]. It will therefore be dealt with only to the extent necessary from the viewpoint of completeness of the subject.

The fuel channels in the KANUPP reactor are fuelled in a direction opposite to the coolant flow. This means that the fuel irradiation is lowest at the coolant downstream end of the channel and highest at the upstream end. During the postulated Large Break Loss of Coolant Accident (LBLOCA), the fuel string is moved towards the upstream end of the channel which leads to a positive reactivity inserted by the movement of lower irradiation fuel bundles to a higher flux region of the channel and higher irradiation bundles to a lower flux region. This occurs in the broken pass of the single heat transport system loop that is in half of all fuel channels (104).

The magnitude of positive reactivity insertion on account of FSR depends on the axial gap between the upstream fuel bundle and part of the inlet end shield plug which blocks further fuel string motion, that is the distance which the fuel string can move during the postulated accident. The smaller the gap is, the smaller is the reactivity effect of the fuel string relocation. This gap in the case of KANUPP is 7.5 cm. On the basis of reverse hydraulic impact in the aftermath of LBLOCA, the string of bundles has been estimated to take 83 ms to move forward into this gap. The corresponding reactivity effect has been evaluated to be 0.9 mk.

5. ANALYTICAL TOOLS AND ASSOCIATED MODELS

The analysis reported in this paper made use of the computer codes currently employed in Canada for the design and safety assessment of CANDU reactors. These include (1) POWDERPUFS -V (PPV) [3] which was utilized for the static core calculations and (2) RFSP (Reactor Fuelling Simulation Programme) [4] for neutron flux and power distribution calculations. The dynamic core calculations were carried out making use of CERBERUS [5] computer code, which is basically a tool for solving the time dependent neutron diffusion equations in three spatial dimensions and two energy groups. The thermal hydraulic analysis was carried out with the help of computer code SOPHT [6]. The code employs a 3-equation model (conservation of fluid mass, momentum and energy) and is considered adequate for single phase analysis.
As described earlier, KANUPP reactor core has features which are different from modern CANDUs. These pertain to channel misalignment, top and bottom entering control absorbers, empty channel G-12 and moderator level variation for reactivity control. Appropriate modelling of these features was incorporated in the RFSP code. The thermal hydraulic part of the analysis was based on the computer code SOPHT. To develop the SOPHT model, the plant systems were divided into NODES and LINKS. Essentially the Primary Heat Transport (PHT) and Emergency Water Injection (IJW) only were modelled. Modelling of the controllers was not considered; for events during major loss of coolant accident occur much too fast for the plant regulation controllers to respond with any kind of action in the time frame considered. The core was divided into 22 zones with 11 axial nodes representing 11 fuel bundles. Considerations relating to direction of coolant flow, power rundown after the initiation of moderator dump and proximity of maximum power producing channels dictated partitioning of channels into various zones.

6. SYSTEM CHARACTERISTICS CONTRIBUTING TO SAFETY

In addition to some inherent safety characteristics contributing to safety of CANDU’s the following system features relating to KANUPP reactor also contributed in the qualification of its moderator shutdown system.

6.1. Dynamic Reactivity Worth of Moderator Shutdown System

In static calculations steady state is assumed and delayed neutron precursor densities are implicitly proportional to the flux shapes. This is so for both the normal core configuration and with the shutdown system inserted. The shutdown system worth obtained from these calculations is called the static worth. In dynamic calculations the effect of delayed neutrons is explicitly taken into consideration. In such a situation precursor shape lags behind the instantaneous or prompt flux. Even when the shutdown system is inserted there are delayed neutrons in the region of action of shutdown system which enhance the effectiveness of the shutdown system. The increased reactivity worth pertinent to this situation is termed as the shutdown system dynamic reactivity.

![FIG. 1. Reactivity variation as a function of moderator level in Calandria](image-url)
A comparison of static and dynamic reactivity worth curves is given in Fig. 1 indicating the enhanced effectiveness of the shutdown system in controlling the post LBLOCA power excursion. It should be remembered that depending upon the delayed neutron distribution prior to a reactivity excursion, the dynamic worth could increase by as much as 80 percent over and above the static worth.

6.2. Dynamic Void Reactivity

As explained in the foregoing section the delayed neutrons in the core (below the moderator interface) maintain the pre-shutdown shape and relative to static calculation have higher probability of leaking out hence the dynamic reactivity is higher. For the same reason of shape delay, the (positive) void reactivity worth is less than the void static worth. For that reason if one were to simulate the transient with a reactivity ramp derived from a succession of static simulations, the desired level of accuracy would not be obtained in the calculation of power amplitude and power shape. It is important to use dynamic reactivity.

6.3. Heat Transport System Response in Post LBLOCA Scenario

The thermal hydraulic response of the heat transport system assessed on the basis of SOPHT calculations indicated that the void fraction is still ~ 40% at the time of opening of shutdown valves in a worst case scenario of postulated 100% rupture in the inlet header. This is contrary to earlier KFSAR analysis which assumed insertion of full core void reactivity before the action of shutdown system.

6.4. Irradiation Creep in Heat Treated Zr-Nb Pressure Tubes

KANUPP employs heat-treated Zr-2.5% Nb pressure tubes. The elongation due to creep as measured in eight representative pressure tubes indicated an average length increase of about 3 mm only, which is much less than that measured for cold worked Zr-2.5Nb pressure tubes. In view of insignificant 0.6% elongation this has not been taken into consideration. Had the expansion been of the same magnitude as in the case of cold worked tubes, the effect of fuel string relocation on the post LOCA power pulse would be more severe.

7. RE-EVALUATION OF SAFETY SIGNIFICANT PHYSICS PARAMETERS

A comparison of temperature coefficients of reactivity, kinetic parameters and full core void reactivity for equilibrium condition and design core configuration obtained from PPV with those given in the previous KFSAR, is presented in Table-II. The void reactivity was also calculated for currently existing moderator and coolant isotopic.

The time average reactor core calculations performed making use of RFSP computer code in conjunction with PPV generated lattice parameters yielded for the design moderator and coolant isotopic a value of 7400 MWd/teU as against 8650 MWd/teU for the average discharge burnup, mentioned in the KFSAR. The effect of downgrading of moderator and coolant on such core performance parameters as average discharge burnup and bundle consumption rate was calculated to be consistent with the operating experience since commissioning of the plant.

The design values of full core void reactivity calculated using MOLAR and FOG for fresh KANUPP core, corresponding to 99.75 wt.% D2O purity both for the coolant and moderator was 11.4 mk. The revised calculations based on PPV code for identical purities put this estimate at 10.5 mk.
TABLE II. PHYSICS PARAMETERS AFFECTING REACTOR SAFETY

**Temperature Coefficient of Reactivity (Equilibrium Core)**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Design Value</th>
<th>Re-Evaluated Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Temp Coefficient</td>
<td>-</td>
<td>- 0.007 mk/°C</td>
</tr>
<tr>
<td>Coolant Temp Coefficient</td>
<td>+ 0.016 mk/°C</td>
<td>+ 0.0223 mk/°C</td>
</tr>
<tr>
<td>Moderator Temp Coefficient</td>
<td>+ 0.050 mk/°C</td>
<td>+ 0.040 mk/°C</td>
</tr>
<tr>
<td>Power Temp Coefficient</td>
<td>- 0.023 mk/°C R.P.</td>
<td>- 0.0293 mk/°C R.P.</td>
</tr>
</tbody>
</table>

**Reactor Kinetic Parameters (Equilibrium Core)**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Design Value</th>
<th>Re-Evaluated Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ave. Delayed Neutron Fraction</td>
<td>0.527%</td>
<td>0.547%</td>
</tr>
<tr>
<td>Prompt Neutron Life Time</td>
<td>0.785 ms</td>
<td>0.746 ms</td>
</tr>
</tbody>
</table>

The void reactivity predicted by PPV as a function of fuel burnup for the nominal KANUPP lattice is shown in Fig. 2. Evidently, for equilibrium burnup fuel, PPV predicts a substantially higher void reactivity (5.6 mk) than the design values (3.4 mk) computed with MOLAR. Similar calculations using WIMS-AECL have also been performed which suggest even higher value (6.7 mk). For the CANDU 6, 37-element fuel lattice, the comparisons of WIMS-AECL void predictions show similar features i.e. agreement for fresh and divergence as the fuel burnup increases.

![Fig. 2. KANUPP void coefficient of reactivity (Based on PPV Lattice Cell Calculations)](image-url)
Full core void reactivity was computed for various core conditions using the code PPV. The results are given in Table-III. The void reactivity corresponding to the current equilibrium core conditions viz. moderator (MH): 99.57 wt% and PHT: 98.8 wt% was evaluated to be 6.3 mk. These conditions are used in the CERBERUS code for the assessment of KANUPP shut down system.

TABLE III. PPV CALCULATED FULL CORE VOID REACTIVITY UNDER VARIOUS CORE CONDITIONS

<table>
<thead>
<tr>
<th>Core Conditions</th>
<th>Full Core Void Reactivity (mk)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Equilibrium Core (MH: 99.75%, PHT: 99.75%)</td>
<td>5.60</td>
</tr>
<tr>
<td>Equilibrium Core (MH: 99.75%, PHT: 98.80%)</td>
<td>6.02</td>
</tr>
<tr>
<td>Equilibrium Core (MH: 99.57%, PHT: 98.80%)</td>
<td>6.30</td>
</tr>
<tr>
<td>Equilibrium Core (MH: 99.75%, PHT: 97.50%)</td>
<td>6.60</td>
</tr>
<tr>
<td>Equilibrium Core (MH: 99.57%, PHT: 97.5%)</td>
<td>6.88</td>
</tr>
<tr>
<td>Equilibrium Core (MH: 99.75%, PHT: 99.75%) + 1 ppm B</td>
<td>6.12</td>
</tr>
</tbody>
</table>

Studies were also carried out to determine the void reactivity as a function of coolant density. The coolant degradation penalty on void reactivity was found to be + 0.4014 mk/atom%. On the basis of the currently prevailing rationale, the minimum coolant purity for KANUPP has been set at 97.5 wt% (97.3 atom%). Void reactivity corresponding to this purity below which the plant may be operating in an unanalyzed regime has been estimated by PPV to be 6.88 mk. An uncertainty allowance of the suggested 20% over and above results in a figure of 8.3 mk for the full core void reactivity. The 3D CERBERUS calculations require coolant purity corresponding to void reactivity as input. Using the estimated coolant degradation penalty of 0.4014 mk/atom%, the ultimate coolant purity for the purpose of accident analysis works out to be 93.8 atom%.

The effect of boron in the moderator on void reactivity was analyzed making use of the PPV code. It was determined that 1 ppm of Boron in the moderator would increase the equilibrium core void reactivity from 5.6 mk to 6.12 mk (~ 9.2%). For the purpose of safety analysis, the base case was defined as the one with maximum concentration of 0.5 ppm boron in moderator. Sensitivity analysis was however performed with 1.5 ppm boron as well.

8. REASSESSMENT OF SHUTDOWN SYSTEM

Taking into account presence of impurities in the PHT and MH and consequent adverse effects on void reactivity besides the impact of Fuel String Relocation the performance of slow acting moderator dump shutdown system was reassessed by carrying out the analysis for postulated guillotine (100% - double ended) rupture in the inlet header. The result of the analyses are summarized as follows:

A postulated LBLOCA with 0.5 ppm B in moderator, as stated earlier in the text, was used as the base case for determining the adequacy of the shut down system. The neutron flux transient with and without FSR for this case are shown in Fig. 3. With FSR there is a sharp increase in reactivity due to fuel string relocation. This results in exceeding the log N trip (second trip) set point of 15% at 753 ms into the accident. A peak reactivity ~ 4.9 mk is inserted until then. The reactor power peaks at 3.9 times the operating power. The negative
Fig. 3. Neutronic power transient (rih guillotine break LOCA) Sensitivity to fuel string relocation

Fig. 4. Large LOCA neutronic transient Sensitivity to boron in moderator

reactivity due to dumping of moderator makes the reactor sub critical at 1.7 s at which point moderator height in calandria is ~160 inches. The maximum integrated power of 285 kJ/kg during the transient upto 5.3 s is seen by an outer elements of bundle at position 6 in channel L09. Including the stored energy of 318 kJ/kg, the hot element enthalpy is computed to be 603 kJ/kg which is only 63% of 960 kJ/kg limit for center line melting. This fulfills the criteria as such for the qualification of shutdown system quite adequately.
In order to demonstrate the capability of shutdown system, its adequacy was also determined for an upper bound case of 1.5 ppm boron in moderator (Fig. 4) and tripping on third rather than second trip. The results indicated the system to be fully capable of containing even such an improbable situation. The adiabatic enthalpy of the fuel up to ~5 s into the overpower transient was found to be 692 kJ/kg which is well below the 960 kJ/kg limit for fuel center line melting.

9. DISCUSSION

The revamp of the KANUPP reactor physics analysis have indicated that the present design and analysis codes do not predict the same results as given in the design description with respect to certain physics design parameters e.g. the cell constants, temperature coefficients of reactivity and fuel average discharge burnup. The reason for this has been identified to be the use of correct input data and validated computer codes evidently based on improved methodology and updated nuclear data.

In spite of the fact that the full core void reactivity was revised considerably upwards and the consequences of downgrading of moderator and coolant as well as the concentration of boron in moderator and fuel string relocation were taken into consideration, the moderator dump shut down cooling system could still be qualified. This partly, is due to the fact that in the earlier analysis full core void reactivity was assumed to have been inserted before the action of shutdown system. As demonstrated by the current analysis, this is not what may happen in actual practice. In the worst circumstance of 100% rupture in inlet header, the void fraction is still ~ 40% at the time of opening of shutdown system valves. Moreover the realization that the effectiveness of the shutdown system substantially increases on account of the delayed neutrons has played a significant role in suppressing the amplitude of power pulse.

In conclusion the presented studies have demonstrated that in spite of some noticeable improvements in the understanding of some in-reactor phenomena and introduction of altogether new concepts impinging upon safety, the smaller first generation reactors of the seventies could still be operated into the new millennium with renewed assurance and confidence. It is pertinent that such safety reassessments as carried out for KANUPP will go a long way in designing and building better and safer SMRs

ACKNOWLEDGEMENTS

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DEPLOYMENT ENVIRONMENT FOR SMALL AND MEDIUM REACTORS

(Session 13)

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REQUIREMENTS FOR DEPLOYING SMRs IN DEVELOPING COUNTRIES:
THE KOREAN EXPERIENCE

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Abstract

Interest in deploying SMRs is growing in developing countries as they are more suitable for meeting the needs of relatively smaller distribution systems as well as being easier to finance and having a lower capital cost. Many developing countries are thus in an early stage of preparing for the introduction of SMRs. This paper describes the requirements for deploying SMRs in developing countries by dividing the requirements into four (4) categories, i.e. National Nuclear Energy Policy, Economics and Financing, Infrastructure, and Technology. As a specific example of the planning and implementation of nuclear power projects in a developing country, the Korean experience in nuclear power programs is described with respect to requirements for deploying nuclear power reactors. In addition, the top-level requirements imposed on SMART (System-integrated Modular Advanced Reactor), an integral PWR with the rated thermal power of 330MW, are discussed.

1. INTRODUCTION

SMRs have various merits such as inherent characteristics, passive safety features, low environment impacts, and low capital cost [1-2]. Many developing countries are thus concerned about the introduction of SMRs and consequently need assistance in establishing their own requirements[3]. Prior to making a decision on the deployment of SMRs, their long-term energy program, availability of natural resources, capital investment and financing, industrial infrastructure, technology transfer, and national participation should be evaluated. For this purpose, the establishment of user requirements will be very helpful for defining the needs for the planning and implementation of specific projects. In the early 1980s, utilities in the USA, the European Union (EU), Japan and the Republic of Korea developed their own user requirement documents for ALWRs and next-generation reactors. However, the requirements for developing countries may differ very much from those of developed countries especially in terms of infrastructure, technological capabilities, financing capabilities and other environments relevant to the introduction of SMRs. This paper addresses the user requirements such as National Nuclear Energy Policy, Economics and Financing, Infrastructure, and Technology Transfer for developing countries

During the past two decades, the Republic of Korea has accomplished outstanding achievements in facilitating nuclear power development. For a stable and economical electricity supply, nationwide efforts towards achieving self-reliance in nuclear power technology have been made. At present, sixteen (16) nuclear power plants are in operation with a total capacity of 13,716 MWe, which is about 40% of the national power need. In this paper, the Korean experience in nuclear power programs is described as a practical example of the introduction of nuclear power projects for developing countries. In addition, the top-level requirements imposed on SMART are discussed.
2. REQUIREMENTS FOR DEVELOPING COUNTRIES

For the introduction of SMRs in developing countries, the establishment of user requirements will be essential for defining the needs for the planning and implementation of their projects[2-3]. The environments relevant to the introduction of SMRs for developing countries are quite different from those of developed countries especially in terms of infrastructure, technological capabilities, and financing capabilities. The user requirements for developing countries are grouped and discussed with respect to national nuclear energy policy, economics and financing, infrastructure, and technology.


For the introduction of SMRs in developing countries, a desirable first step towards the planning of specific nuclear projects is to establish the national nuclear energy policy requirements. These requirements deal with security of national energy supply, electric demand forecast, energy planning for demand vs. capacity, and grid capacity. Furthermore, these requirements include technology self-reliance policy requirements, and commitments on liability, safety and safeguards for nuclear reactors. Therefore, it is very desirable to establish a national nuclear program to be implemented. For energy and nuclear power planning, the IAEA technical reports will be very helpful [2-3]. In addition, international treaties, legal obligations and conventions such as the Treaty on the Non-Proliferation of Nuclear Weapons (NPT), the Convention on the Physical Protection of Nuclear Material, and the Convention on Nuclear safety are also important to be considered.

2.2. Economics and Financing Requirement

Economic competitiveness is a decisive factor in decision making for the deployment of SMRs. Therefore, the economic target must be set up in advance. To compete with advanced fossil-fired power plants, SMRs should be developed to meet utility requirements at a minimum power generation cost by shortening the construction period, introducing modular fabrication and construction, reducing O&M and fuel cycle costs. Factors affecting the economics of SMR are design simplicity, mass production, standardization, and site cost. Economics and financing requirements include capital cost, O&M cost, fuel cost, financing, and techno-economical optimization. For the financing scheme, conventional financing approaches with export credits and commercial loans on sovereign guarantees as well as alternative financing approaches with build-operate-own (BOO), and/or build-operate-transfer (BOT) could be considered in deploying SMRs in developing countries.

2.3. Infrastructure Requirement

For the introduction of SMRs in developing countries, more industrial infrastructures for the organization, systems, and resources required to implement nuclear programs are involved than the general practice in developed countries. Therefore, it is very desirable to establish infrastructure requirements for the introduction of SMRs in developing countries. These infrastructure requirements refer to the electric grid, transportation, organizations for the project planning/implementation and plant operation, manpower development, technical support, regulation and licensing support. Other infrastructures requirements such as manpower development, national participation, and long-term assurance of fuel supply should be additionally considered for developing countries.
2.4. Technical Requirement

The technical requirements may directly influence the economics of the SMRs and are closely related to the other requirements including economic requirements and infrastructure requirements. These technical requirements should cover the whole spectrum of design development, manufacturing, project management, QA & QC, and safety/licensing. The specification of technical requirements depends very much on the type of reactor, the site characteristics and its applications. Technical requirements comprise requirements for safety, plant performance, and plant design of SMRs. In order to meet the safety requirements, the SMRs should have design features such as prevention of the core damage, degradation of an accident to severe accident condition, passive safety, and radiation protection. To meet the performance requirements, availability, reliability, maintainability and manoeuvrability should be considered in the SMRs design. In the plant design requirements, design lifetime, design simplicity, design margin, standardization, mass production, proven technology, constructability, and decommissioning should be considered.

3. KOREAN EXPERIENCE IN NUCLEAR POWER PROGRAMS

3.1. Nuclear Power Programs in the Republic of Korea

Nuclear power is an inevitable option in the Republic of Korea to overcome scarce natural resources and to reduce overseas energy dependence. Nuclear power generation has thus played a major role in electricity supply since 1978 when the Kori unit 1, the Republic of Korea’s first nuclear power plant, went into commercial operation. As a result of intensive nuclear power development programs based on the steady execution of an energy source diversification plan, the Republic of Korea now has sixteen (16) nuclear power plants in operation, four (4) units under construction, and an additional ten (10) units planned before 2015.

The self-reliance program in nuclear power technology from the 1970s to the present can be divided into three phases: namely, turn-key base, component approach, and self-reliance strategy. The first phase was characterized as a turn-key base contract by foreign suppliers on the first three units. During this period, all the work including design, manufacturing, and construction have largely been performed by foreign suppliers. Therefore, nuclear power technology could not be accumulated and localized. The second phase was based on a so-called component approach executed for nuclear unit Kori 3&4, Yonggwang 1&2, and Ulchin 1&2. In this phase, the utility was in charge of project management. The plant design and manufacture of the primary system was performed under contract by foreign suppliers. The third phase, adapting the self-reliance strategy, has been applied to the implementation of Yonggwang units 3&4. The most important objective in this phase is to accomplish complete self-reliance in nuclear power plant construction.

3.2. Self-reliance in Nuclear Power Technology

3.2.1 Industrial Infrastructure

In order to achieve nuclear power technology self-reliance, the Korean Government assigned technical responsibilities to nuclear organizations in 1985 [4]. KEPCO (Korea Electric Power Corporation), the nation’s sole utility company, undertook all responsibilities for project
management. KAERI (Korea Atomic Energy Research Institute) was responsible for the design of NSSS and nuclear fuels for PWR and CANDU. KOPEC (Korea Power Engineering Company) has been responsible for architectural engineering work. KNFC (Korea Nuclear Fuel Company) has produced all the locally needed PWR fuels. HANJUNG (Korea Heavy Industries and Construction Co. Ltd.) has been assigned to manufacture major components of NSSS and turbine-generators. As of the end of 1996, NSSS and fuel technology secured by KAERI was successfully transferred to the nuclear industries.

### 3.2.2 Self-reliance in Fuel Technology

The development of CANDU fuel has taken place since the mid-1970s at KAERI. After irradiation tests for twenty-four (24) fuel bundles at Wolsung unit 1 were successfully completed in 1984, locally produced fuels for Wolsung unit 1, designed and manufactured by KAERI, have been supplied since 1988 [6]. The development of an advanced CANDU fuel, CANFLEX (CANDU Flexible) has been performed jointly with AECL. The CANFLEX has been examined through extensive verification tests performed by KAERI and AECL including irradiation of twenty-four (24) CANFLEX bundles at the Point Lepreau Generating Station.

Upon its success in the development of CANDU fuel technology, the Korean Government also decided to localize fuel production for the PWR. To expedite self-reliance in PWR fuel, two technology inducement contracts were made in 1985 among Kraftwerk Union (KWU) of Germany, KAERI, and KNFC. As a new approach to self-reliance in PWR fuel design technology, a joint design was introduced by KAERI for the first time and has been found to be useful and efficient for technology transfer [5-6]. Through the joint design, three types of Korean Fuel Assemblies (KOFA) were developed; namely, 14 × 14 fuel assembly, 16 × 16 fuel assembly, and 17 × 17 fuel assembly. The locally fabricated PWR fuels, designed by KAERI and manufactured by KNFC with the help of KWU, have been supplied to domestic PWR plants since 1989.

### 3.2.3 Self-reliance in NPP Construction Technology

For effective technical self-reliance, a joint system design concept was also conducted between KAERI and CE in the NSSS system design and engineering work. The Korean Standard Nuclear Power plant (KSNP) concept has been developed by adapting several advanced technologies and is being applied to the construction of Yonggwang 5&6 and Ulchin 5&6. Also, KSNP technologies are now being applied to the first commercial nuclear power plant under construction in the Democratic People’s Republic of Korea. In addition, the Korean Next Generation Reactor (KNGR) as a 1,400 MWe advanced PWR has been developed since 1991. The KNGR will be potentially characterized with drastically enhanced safety, reliability, and operability as well as improved economy compared to currently existing plants. The first commercial operation of the KNGR is expected in the year 2010.

### 3.3. Prospects of Nuclear Power Program

While pursuing the self-reliance in nuclear power technology, a long-term R&D program was established in 1992 in order to upgrade the indigenous nuclear technology and to achieve innovations in nuclear technology [7]. These R&D programs have focused on improving the indigenous nuclear power technology such as improvements in safety and economy of KSNP, development of KNGR, SMART, and advanced fuels, and establishment of industrial codes & standards.
4. DESIGN REQUIREMENTS FOR SMART

The conceptual design of SMART with a desalination system was completed in March, 1999 [8]. The basic design for the integrated nuclear desalination system is currently underway and will continue until 2002. In the beginning stage of the SMART development, the top-level requirements for safety and economics were imposed for the SMART design features.

TABLE I. SMART DESIGN FEATURES RELATED TO SAFETY REQUIREMENTS

<table>
<thead>
<tr>
<th>Safety requirements</th>
<th>SMART design features</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety enhancement</td>
<td>✩ Integral primary circuit</td>
</tr>
<tr>
<td></td>
<td>✩ Low core power density</td>
</tr>
<tr>
<td></td>
<td>✩ Large negative MTC</td>
</tr>
<tr>
<td></td>
<td>✩ Large thermal margin</td>
</tr>
<tr>
<td></td>
<td>✩ High degree of primary circuit natural circulation</td>
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<tr>
<td></td>
<td>✩ Large pressurizer</td>
</tr>
<tr>
<td></td>
<td>✩ Large primary circuit water volume</td>
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<tr>
<td>Passive safety system</td>
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<td></td>
<td>✩ Passive residual heat removal system</td>
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<td></td>
<td>✩ Passive containment overpressure protection system</td>
</tr>
<tr>
<td>Radiation protection</td>
<td>✩ Safeguard vessel</td>
</tr>
<tr>
<td></td>
<td>✩ Removal of safety grade ADV</td>
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</table>

TABLE II. SMART DESIGN FEATURES RELATED TO ECONOMIC REQUIREMENTS

<table>
<thead>
<tr>
<th>Economic requirements</th>
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</tr>
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<tbody>
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<td>Capital cost</td>
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</tr>
<tr>
<td></td>
<td>✩ Design simplicity</td>
</tr>
<tr>
<td></td>
<td>✩ Longer plant life-time</td>
</tr>
<tr>
<td></td>
<td>✩ Reduced number of equipments</td>
</tr>
<tr>
<td>Fuel cost</td>
<td>✩ Higher fuel burn-up</td>
</tr>
<tr>
<td>O&amp;M cost</td>
<td>✩ No soluble born core</td>
</tr>
<tr>
<td></td>
<td>✩ Design simplicity</td>
</tr>
</tbody>
</table>

4.1. Safety Requirement

In developing SMART, safety enhancement is one of the most important considerations. The safety requirements on the SMART were top-tiered by the core damage frequency per reactor year less than $10^{-7}$ and the large off-site dose release rate of less than $10^{-8}$ per reactor year. These requirements have been achieved by applying the safety requirements on safety enhancement, passive safety system, and radiation protection to the SMART design features. Table 1 correlates these safety related design requirements with the major SMART system design features.
4.2. Economic Requirements

The top-tier requirements on economics were set up to achieve economic competitiveness at least equal to that of gas turbine plants. To achieve these design requirements, design simplicity, modular design, longer plant lifetime, ample design margin, reduced waste, and better maneuvering and transient response were adopted in the SMART design features. The economic requirements with respect to the major SMART system design features are presented in Table 2.

5. SUMMARY

The establishment of user requirements is essential for developing countries to define the needs for the planning and implementation of specific projects. However, the requirements are dependent mainly on the country specific conditions and needs of each developing country. These requirements for developing countries are thus very different from those of developed countries especially in terms of infrastructure, technological capabilities, financing capabilities and other environments relevant to the introduction of SMRs. The factors to be considered in user requirements for developing countries are described with respect to national nuclear energy policy, economics and financing, infrastructure, and technology.

As a specific example of the deployment of nuclear power project in a developing country, the Korean experience in nuclear power programs are described. The top-level requirements imposed on safety and economic aspects of the SMART development are also discussed.

REFERENCES

A CLOSE LOOK AT DEVELOPING COUNTRY CUSTOMER’S ESSENTIAL REQUIREMENTS FROM CHINA’S NUCLEAR POWER DEVELOPMENT

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Abstract

The status of nuclear power development in China was briefly introduced. The major features of China’s nuclear power program were summarized, which was a reflection of the nuclear power policies of the developing stage in the country. From China’s nuclear power development, the customer’s essential requirements of nuclear power in developing countries were drawn. Among them, safety, improved economy and broadening financing channels should be emphasized in order to promote nuclear power in developing countries.

1. THE STATUS OF NUCLEAR POWER IN CHINA

1.1. Nuclear Power Plants In Operation And Under Construction

China’s nuclear power program was launched in the 1980s. Currently, 3 PWR units are in safe and sound operation with remarkable business revenue. Qinshan Nuclear Power Plant is the first PWR unit in China, which was designed and constructed by the Chinese ourselves. The Daya Bay Nuclear Power Plant with 2 PWR units was imported from overseas. The nuclear installed capacity of 2.1 GWe is about 0.7% of the national total. 3 nuclear units delivered about 15 billion KWh of electricity to the national grid annually, around 1.2% of the country’s total power output. China’s installed power generating capacity has exceeded a record 300 GWe and the total power output in the country has reached 1.23 trillion kWh by the end of 2000. Meanwhile, a set of nuclear power plants with 300 MWe PWR was exported overseas, which operated commercially in September 2000. 2 CANDU units and 6 PWR units of 3 different types with a total installed capacity of 6.6 GWe are under construction now, they will be put into operation during the 10th five-year plan period (2001-2005), as shown in Fig.1. By that time, the nuclear power accounts for less than 2% of the national total. It will make a positive contribution to electric power for the economically developed coastal area in China, to ease coal transportation and to reduce acid rain.

<table>
<thead>
<tr>
<th>Project</th>
<th>Unit</th>
<th>Time Schedule For Operation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tianwan Project</td>
<td>Unit 2</td>
<td><img src="image1" alt="Graph" /></td>
</tr>
<tr>
<td></td>
<td>Unit 1</td>
<td><img src="image2" alt="Graph" /></td>
</tr>
<tr>
<td>Qinshan Project III</td>
<td>Unit 2</td>
<td><img src="image3" alt="Graph" /></td>
</tr>
<tr>
<td></td>
<td>Unit 1</td>
<td><img src="image4" alt="Graph" /></td>
</tr>
<tr>
<td>Lingao Project</td>
<td>Unit 2</td>
<td><img src="image5" alt="Graph" /></td>
</tr>
<tr>
<td></td>
<td>Unit 1</td>
<td><img src="image6" alt="Graph" /></td>
</tr>
<tr>
<td>Qinshan Project II</td>
<td>Unit 2</td>
<td><img src="image7" alt="Graph" /></td>
</tr>
<tr>
<td></td>
<td>Unit 1</td>
<td><img src="image8" alt="Graph" /></td>
</tr>
</tbody>
</table>

**FIG. 1. Commercial operation schedule of NPPs under construction**
1.2. Expected Nuclear Power Program For Future

**R&D**

The technology of 300 MWe and 600 MWe PWRs has been developed in China during the past 2 decades. Meanwhile, we also initiated R&D on the next generation advanced PWR technology to face the energy challenges of new century. Taking into consideration China’s reality, CNNC has engaged in the development of key AC-600 technology and upgrading its capacity form 600 MWe to 1000 MWe. The AC developments concentrate on advanced reactor and advanced core, passive plus active safety system, simplified system and reduced number of components, digital I&C, modular construction, etc. With the efforts of 2 five-year plans, the step-wise outcome has been achieved.

The High Temperature Gas-cooled Reactor (HTGR) of 10 MWe designed by Tsinghua University came into its first criticality by the end of 2000. HTGR is one of the 3 scientific projects in the nuclear energy field of 863 Hi-Tech Program in China. The others, Fast Breeding Reactor and Fission-Fusion Hybrid Reactor are under way.

**Technology Target**

The nuclear power project started up late in China. Even the TMI core melt accident in 1979 and Chernobyl disaster in 1986 did not affect China’s determination and confidence in the nuclear power program. After Qinshan and Daya Bay NPP were put into operation, the projects of 2 CANDU and 6 PWRs were approved by the government and the construction started in 9th five-year period (1996-2000). From the technical point of view, however, all of these units were essentially developed with 1970s/1980s technologies. Learning the worldwide lessons from the past and our own experiences from 3 operated units and 8 units under construction, China’s nuclear sector and the relative departments of the state council emphasize the proven technologies and localization for the proposed nuclear projects in the 10th five-year plan period. The emphasis must be placed on advanced technologies for the middle term development of the country’s nuclear power.

**Expected Installed Capacity**

When the 8 units currently being built are completed, the country’s installed nuclear power capacity will reach 8.7 GWe by 2005. The expected installed capacity might be 20 GWe by 2010 and around 40 GWe by 2020, according to expert’s estimation. However, the actual scale of installed capacity depends on carrying out policy to develop nuclear power as substituted energy and the uranium resource to a great extent. Based on the energy mix, China’s energy policy is to develop hydropower with main efforts, to optimize fossil-fueled power growth and to develop nuclear power commensurately. How many nuclear projects to be started in the 10th five-year plan (2001-2005) is still an unsettled question now. Besides the sites for the units being built in Qinshan, Lingao and Tianwan, in which there is a certain room to expand new units, up to now several reserve sites have been selected for future’s nuclear projects (See Table 1).
TABLE 1. RECOMMENDED SITES FOR NUCLEAR POWER DEVELOPMENT IN CHINA

<table>
<thead>
<tr>
<th>Recommended Sites</th>
<th>Site Capacity (MWe)</th>
<th>Scale of Phase-I Project (MWe)</th>
<th>Current Situation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Guangdong Yangjiang</td>
<td>4×1000</td>
<td>2×1000</td>
<td>Pre-primary feasibility study and project proposal</td>
</tr>
<tr>
<td>Zhejiang Sanmen</td>
<td>4~6×1000</td>
<td>2×1000</td>
<td>Pre-primary feasibility study and project proposal</td>
</tr>
<tr>
<td>Fujian Huian</td>
<td>4×1000</td>
<td>2×1000</td>
<td>Pre-primary feasibility study</td>
</tr>
<tr>
<td>Shangdong Haiyang</td>
<td>4×1000</td>
<td>2×1000</td>
<td>Pre-primary feasibility study</td>
</tr>
<tr>
<td>Jiangxi Pengze</td>
<td>4×1000</td>
<td>2×1000</td>
<td>Pre-primary feasibility study</td>
</tr>
</tbody>
</table>

1.3. Challenges: Nuclear Power Meets Competition From Power Market

The first 2 decades of the 21st century will be a critical period with challenges in the world nuclear history, because the most existing NPPs are to be decommissioned and NPP regeneration is upcoming. The further development of nuclear power faces two issues: nuclear safety and final disposal of high-level radwastes; economic competition with advanced conventional power generation technologies like clean coal, combined cycle, etc. Besides the common issues above, especially the country’s nuclear power meets the competition in the power market recently. The proposed west-to-east gas pipeline project and west-to-east electricity transmission strategy will provide the eastern regions with abundant energy/power supply. It might be occupy a part of power market rooms supplied by nuclear power. It is quite clear that China’s nuclear power has to meet higher and higher safety goal and much improved economy.

2. MAJOR FEATURES OF CHINA’S NUCLEAR POWER PROGRAM

2.1. Self-reliant Design And Construction Plus Import And Export Of whole unit

The technically proven PWR is adopted as the main stream reactor type and embodies the basic technology route for the development of nuclear power in China. The first NPP in China’s nuclear power history, Qinshan-I project, which was completed and put into commercial operation in 1994, during the 8th five-year plan period (1991-1995), is a self-designed and self-built NPP with a 300MWe PWR. The Daya Bay NPP with 2 units was an imported 900MWe PWR type. Both NPPs reflect the characteristics of nuclear power program in China in the start-up stage, i.e. the combination of self-reliant design and construction with introducing the whole foreign units. In addition, at very early stage of China’s NP program a 300 MWe PWR unit was successfully exported overseas.

2.2. Variety Of Nuclear Power Technologies Caused By Variety Of Financing Sources

A shortfall of initial capital investment is the other characteristics of China’s nuclear power program, i.e. the diversity of financing determines the variety of nuclear power technology. The 8 units being built have 4 types of technologies from 4 technical channels. Qinshan-II project: 2*600 MWe PWRs with self-reliant design and construction; Lingao NPP project: 2*900 MWe French PWRs; Qinshan-III project: 2*728 MWe PHWRs imported from Canada.
and Tianwan NPP project: 2*1000MWe PWRs (VVER) from Russia. Financing sources played an important part in decision making on the projects.

In the long term, over-diversification of technologies does no good for safety and economics. Following the criteria of the new generation nuclear power, it is an urgent task in choosing and deciding its future technology policy that is adaptive to China’s reality.

2.3. China’s Nuclear Power Of Developing Stage Facing Regeneration

Nuclear power needs a long construction period and long life span. The nuclear power plants being built in China are the technologies of the 1970s and 1980s that will be operated for a full life time in the new century, while advanced nuclear power countries have developed new generation NPPs for the new century. This forces us to face the world technology regeneration as soon as China steps into the development stage of nuclear power. It is imperative for China to build NPP with upgraded technological level. Otherwise, China’s nuclear power technology will fall behind and pay a price to upgrade to international level.

2.4. Good PA Environment

The 3 units in operation enjoy sound safety record. The improved air quality, the acquired economic benefits and the propelled local economy make the public accept the nuclear power as a “clean energy”. Compared to quite a number of other countries, China enjoys a better public environment. This is a shining feature.

The major features of China’s nuclear power program were summarized in Fig.2.

Variety of technologies caused by diversity of financing sources

Self-reliant design and construction plus import and export of whole unit

Facing regeneration of technologies

Advanced reactor technology self-reliant design and localization of equipment will be emphasized

FIG. 2. Major Features of China’s Nuclear Power Program
3. CUSTOMER’S ESSENTIAL REQUIREMENTS OF DEVELOPING NUCLEAR POWER IN DEVELOPING COUNTRIES

3.1. Major Constraints

Since the early 1980s, the Chinese nuclear sector has made several national nuclear power plans, with “planning goals” in nuclear power installed capacity that we failed to meet. The underlying reason is that the “planning goals” went beyond China's macroeconomic means, without considering the reality. The following inferences on major constraints to develop nuclear power in developing countries could be drawn from the problems met from developing nuclear power in China:

Lack Of Funding

Nuclear power needs a large amount of initial capital. Reducing initial capital causes a debt increase. Lack of funding poses as the main constraint to start nuclear power project.

High Construction Cost Affects Economic Competitiveness

Imported nuclear power units with high construction costs still lack in economic competitiveness currently.

Nuclear Power Projects Are Prone to Impact of International Politics

Nuclear power projects involve many parties. They are under the impact of factors related to international politics. Complication makes the decision timing of nuclear power projects a long process.

FIG 3. Customer’s Essential Requirement Of Nuclear Power In Developing Countries
3.2. Customer’s Essential Requirements in Developing Countries

In today’s world, some developing countries need nuclear power to support economy growth and reduce greenhouse gases. Generally speaking, the nuclear power program of developing countries could come across the similar problems in China’s nuclear power program, as mentioned above. Although URD and EUR developed in nuclear power countries have been disseminate worldwide, however, based on China’s nuclear power development, I would like to emphasize the customer’s essential requirements of nuclear power in developing countries.

(See Fig.3)

Much Safety

While the Golden Era of nuclear power development in 20th century is reviewed, nuclear power sector has to remember the lessons from nuclear accidents/events during a few past decades. The first challenge to further develop nuclear power worldwide comes from higher and higher requirements on reactor safety and final disposal of high-level radwastes of the public and governments. The core damage frequency (CDF) should be $10^{-5}/\text{reactor} \cdot \text{year}$ or lower and post-accident radioactive release probability is decreased to the negligible degree, less than $10^{-6}/\text{reactor} \cdot \text{year}$. These safety goals are much better than one of NPPs in operation currently. Generally speaking, the public and governments will pay more and more attention on transportation and treatment/disposal of spent fuel and radwastes.

Improving Economy

The imported NP units with high construction cost and duration very much lack in economic competitiveness currently. The construction cost should be decreased to the extent that nuclear power is economically competitive with conventional power generation technologies like clean coal, natural gas combined cycle, etc. For instance, a final construction cost less than 1,500USD/KW (specific capital investment) is accepted in China currently.

Advanced And Proven Technologies

Increased safety and improved economy entail adopting advanced and proven technologies. Only technological innovation breakthrough to prevent severe accidents, to mitigate accident consequences and to dispose of high-level radwastes can rescind the public’s fear of developing nuclear power. Generally speaking, the evolutionary and innovatory types of reactors have achieved good progress, which reduced the CDF and Post-accident radioactive release probability to an acceptable extent by the public. However, the public has been looking forward to a reliable technology that can be used for final disposal of high-level radwastes.

In the aspect of economics, to optimize design (specially innovation design conception), shorten the construction duration by new technological process, and improve uranium efficiency besides optimized financing can be done to make the capital cost and generation cost of nuclear power compatible to that of advanced conventional fossil-fired power.
Broadening Financing Channels

To finance favorably is a key to develop nuclear power in developing countries. Besides the initial capital investment, the huge debt caused by nuclear power projects makes government feel anxious. Generally speaking, the export credit interest rate for nuclear power equipment based on OECD regulation is much higher than the government credit interest rate that is easily accepted in developing countries. Looking at multi-national cooperation in the global resource allocation and economic activities, developing countries should seek the prospect that international capital enters nuclear power market in flexible forms to contribute to new century’s power growth.

To Secure Nuclear Fuel Supply And Spent Fuel Deal

Not only establishing a nuclear power project but also nuclear fuel supply and spent fuel deals are very prone to the impact of international politics. The majority of developing countries have no established nuclear industry system. After commercial operation of a nuclear power station, to secure a nuclear fuel supply and spent fuel deal will be one of the most important concerns. International nuclear sectors backed by their governments must secure nuclear fuel supply and spent fuel deal.

TABLE II. ACTUAL AND PROSPECTIVE OPTION OF SMALL AND MEDIUM SIZED UNITES FOR NUCLEAR POWER IN DEVELOPING COUNTRIES

<table>
<thead>
<tr>
<th>Type</th>
<th>Size (MWe)</th>
<th>Technical Situation</th>
<th>Practicality</th>
</tr>
</thead>
<tbody>
<tr>
<td>PWR</td>
<td>300-600</td>
<td>Proven; To be improved</td>
<td>Experienced</td>
</tr>
<tr>
<td>PHWR</td>
<td>~600</td>
<td>Proven; To be improved</td>
<td>Experienced</td>
</tr>
<tr>
<td>AP-600</td>
<td>600</td>
<td>Approved by NRC; Passive</td>
<td>To be demonstrated</td>
</tr>
<tr>
<td>AC-600</td>
<td>600</td>
<td>In developing; Passive + Active</td>
<td></td>
</tr>
<tr>
<td>VVER-640</td>
<td>640</td>
<td>Passive + Active</td>
<td>To be demonstrated</td>
</tr>
<tr>
<td>CANDU(NG)</td>
<td>In developing; Passive + Active</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HTGR</td>
<td>Modular Unit</td>
<td>In developing; Inherent safety</td>
<td></td>
</tr>
</tbody>
</table>

To Establish Technological Support System/Channels

In order to operate nuclear power station safely with a good business revenue, establishing strongly technological support system/channels is quite important. It is helpful to approach a higher availability of nuclear power unit.

3.3 Suitable Technologies To Developing Countries

According to the customer’s essential requirements mentioned above, the suitable technologies of small and medium sized reactor to developing countries could be summarized in table 2.
CONDITIONS NECESSARY FOR THE DEPLOYMENT OF NEW NUCLEAR PLANTS — WHAT WILL IT TAKE?

K.R. HEDGES, R.B. DUFFEY, W.T. HANCOX, D.F. TORGERSON
Atomic Energy of Canada Ltd, Canada

Abstract

In the 1970s and 1980s, throughout North America, Europe and several parts of Asia, there was wide development and deployment of nuclear power. In this paper we focus on the North American market but noting the international implications. In Canada the drivers for this deployment were the possibility of safe, reliable, economic power, which was independent of imported fossil fuels.

We stress, and it is interesting and important to note, that many of those drivers for building new plants in the 70s are still valid today. Simply stated, the challenge is for the nuclear industry to convince both investors and the public that new nuclear can meet the safety, reliability and cost requirements. Therefore, cheap, safe, reliable and medium sized nuclear power alternatives are needed to meet the economic and performance targets in the competitive power market of tomorrow.

The paper addresses why the subject of new nuclear plants is again being considered in North America.

The key factors leading to large-scale development of new reactors are proposed. These factors are:

- Excellent performance of existing plants;
- Competitive economics;
- Flexibility in introduction and plant size;
- Reducing the time to market;
- Reduced risk in the project approval process;
- Optimised site use;
- Public acceptance; and
- Emphasizing strategic benefits.

If these factors can be positively addressed then within 10 years it is possible to resurrect the nuclear plant supply business in North America and see a significant number of new SMRs being ordered.

1. INTRODUCTION

In the 1970s and 1980s, throughout North America, Europe and several parts of Asia, there was wide development and deployment of nuclear power. In this paper we focus on the North American market but noting the international implications. In Canada the drivers for this deployment were the possibility of safe, reliable, economic power, which was independent of imported fossil fuels.

These so-called first and second generations of plants were built with a backdrop of rapidly growing electricity demand which, when combined with the expected economies of scale, led to large units built in batches of 2 or 4 units in large markets such as Ontario Canada, several parts of the USA, France and Japan. In the latter part of the 80s slow downs in demand, and
other issues, led to forced delays in commercial operation in Canada and in the USA cancellation of plants.

The operating plants were relatively reliable & economic sources of power with clear economic and environmental advantages over coal. The early teething troubles were overcome. One general concern was the steadily increasing cost of power in part due to capacity factors, increased O & M costs, safety upgrades and other capital improvements.

The accidents at Chernobyl & 3 Mile Island raised significant concerns in the public’s minds, as did delays in establishing long-term strategies for fuel (high level waste) disposal. The result of various combinations of these factors is that no new plants have been committed for 20 years in North America. Nevertheless, energy demand, the power market, economic factors, and environmental concerns continue to grow. In fact, competitive power markets have already opened up new opportunities for existing plants, by adopting considerable business and operational consolidation, plant refurbishment, life extension and operating cost reduction.

Potential customers – the new generating companies ---have stated their expectations and requirements for new plants: lower generating cost, higher availability, reduced O&M costs, less capital funding, higher rates of return, smaller unit size, reduced risk, improved safety, and assured licensability.

We stress, and it is interesting and important to note, that many of those drivers for building new plants in the 70s are still valid today. Simply stated, the challenge is for the nuclear industry to convince both investors and the public that new nuclear can meet the safety, reliability and cost requirements. Therefore, cheap, safe and reliable nuclear power alternatives are needed to meet the economic and performance targets in the competitive power market of tomorrow.

2. WHY IS THE SUBJECT OF NEW PLANTS BACK ON THE TABLE?

There are many factors at work here, all the way from growing power demand, efficient operation of current units, stable licensing and operating conditions, and the simple need for a substantial portfolio of diversified generation options.

**Economics of the Current Plants**

Over the last decade the performance of existing plants in the USA has steadily increased driving down the operating cost of electricity generation. The average “going forward” generating cost has to be competitive with alternate sources, and usually that target can be met. In Canada and the USA the initial operation of plants was often excellent, This led to Utility practices which ultimately and paradoxically led to deteriorating conditions which have required substantial management intervention to restore performance. It is only over the last 3 years in Canada that corrective programs have been put in place, which should lead to performance matching that of the USA.

With the economics & reliability of the current plants improving one of the conditions necessary for new plant commitment is in place. The competitive going forward generating cost, the needed operating infrastructure and necessary management systems therefore exist
which enable the existing plants to supply power at competitive rates into the competitive market. For example, in Canada the generating cost is stated to be in the order of ~1c/kWh.

**Life Extension & License Renewal**

In Canada, as in the USA, the plant owners were faced with making investments to ensure long-term performance. They undertook evaluations of options ranging from plant shutdown to plant life extension. The initial work showed economic benefit in operating the plants well beyond the nominal plant design life. Ontario Power Generation is in the process of upgrading the 27 year old 4 Unit Pickering A plant for extended service and both Hydro Quebec and New Brunswick power are in the initial phase of life extension programs for their nuclear plants.

In the USA the life extension process is tied to license renewal and owners of about 30% of the nuclear fleet have now decided life extension makes sense. Again this commitment to future operation is part of the pre-requisites to, and paths towards new nuclear power plants.

**Environmental Issues**

In its latest sets of scenarios, the United Nations International Panel on Climate Change (UN IPCC) has included the necessity and effects of a growing nuclear energy component (at least until 2050 to 2100) in the search for reduced drivers to potential climate change. The potential nuclear energy contribution to reduced global emissions is significant.

Stringent limits have been put in place in North America, and other major industrial countries, on SOX, NOX, tailpipe and particulate emissions since the 1970s. Efforts to decrease urban smog and acid rain have been started. But growing total emissions from power generation and transportation have given rise to consequent environmental impact concerns, despite the trading of so-called emissions credits. With energy use expected to continue to grow in North America, emissions of particulates and gases (including greenhouse gases) are expected to increase. Canada and Canadians are concerned over deteriorating air quality, due to the possible health effects and the impact on wild life and forests.

Concerns on climate change are now seen to be real, but not high up on the list of political priorities. Being amongst the highest users of energy and emitters of gases and emissions, and producers of fossil fuels North America has resisted the wholesale imposition of limits or reduction on carbon energy use without allowing other offsetting measures.

Nuclear power is well positioned here in this changing climate of opinions to help the world manage its emissions. Although not recognised by its opponents as a “green” energy source, in North America the existing ~120 plants avoid and have avoided billions of tonnes of CO$_2$ and other particulates and gases, and is virtually and comparatively atmospheric emissions-free.

Giving credit (political, social or fiscal) for this major contribution is endlessly opposed by those who see this could delay or impede the large-scale introduction of other sources, such as wind and solar power. Nevertheless, the nuclear track record is good and the promise is substantial, so the relative contribution of nuclear to avoided emissions is a strong positive factor. If agreements are reached on implementing internal or external emissions trading schemes or offsets, these are also expected to favour nuclear energy.
At the same time, concerns about radioactivity in general have also lead to long delays in implementing geologic disposal of high level wastes, both at Yucca Mountain in the USA, and in Canada, There do not seem to be significant technical issues. In the interim, on-site storage of spent fuel is, and this should span the time needed for social acceptance of specific waste disposal sites, either permanent or monitored and retrievable. Thus, the necessary conditions for public acceptance will be met by using a viable technical approach to waste storage. The costs are absorbed easily within the energy cost. Since nuclear is the only energy source to already tackle the full energy cycle, it is ahead of the efforts to close the carbon cycle either by sequestration or “zero” emissions technology.

Hence a very positive environmental outlook is now seen for nuclear, despite the frustrating and long delays in implementing some of the technical strategies. The potential contribution of nuclear is recognised (if not agreed) and is present in the global debate about potential options and technologies.

**Energy Security & Sustainability**

Nuclear energy provides one way to help to ensure security and sustainability of energy sources, as exemplified by the living examples of France and Japan, and by the forward-looking plans of India, Korea and China.

North America, and most major industrial countries in Europe and many in Asia are now dependent on imported energy, usually as gas and/or oil. A vast network of pipelines has been established, which is both under guard and threat. The continued turmoil and political instability of key countries means that there is a high price to pay for both securing the energy supply and ensuring peace of mind. The net energy dependency of North America is expected to grow significantly and with that an attendant fiscal and a political price to pay.

Canada is a net exporter of energy, and has very large oil sands reserves (which exceed the oil reserves in Saudi Arabia) and already provide over 20% of the domestic production, and this source of energy will grow. Hydro power is subject to increasing environmental restrictions.

Security of supply and stability of price will become increasing factors in the future as energy policies and attitudes are developed. We note the present approach of the oil production market is to raise the price and control the supply in response to demand. But both oil and gas reserves and resources are finite and exhaustible, so North America must be concerned over the long term beyond the immediate market despite not having “energy planning” in any formal sense. Therefore, nuclear energy satisfies the criteria of providing competition and a secure energy option.

In the 1990s the term “sustainable development “ was developed as a guide to providing the economic, social and environmental needs and well being of humans, whilst still preserving the planets resources, including the global commons. This all-encompassing vision is enabled by nuclear energy over the next century or two by helping sustainable global economic and social development in two key ways:

1. Nuclear fission can provide both the time and the means for a graceful interval for other energy sources to develop, such as the introduction of hydrogen into transportation; and power the base load power and grid needed for interruptible and low load factor sources such as wind and solar to be integrated successfully;
2. Use of uranium and thorium fuel sources can provide energy for the world two to three hundred years (and are otherwise unusable), and enable the reuse of “spent fuel” and the introduction of fuel cycle flexibility without increased fears of proliferation.

For North America, all these considerations can be important, and are the factors, which enable nuclear energy to be a part of the debate on meeting the future needs for the world’s energy. The need for nuclear has been well stated the World Energy Council in its latest energy studies and predictions. It is easy to show that a nuclear energy portfolio of 10-20% can have a significant impact.

3. MARKET ASSESSMENT

The global energy market, and the future power market in North America are huge. New plants will be needed in North America in the next twenty years (over 300 GW(e) or 1,000 new plants) to meet new growth and replace aging and uneconomic existing plants alone. Traditional markets for power plants in North America were utilities which had access to low cost financing and in many cases monopoly over the supply in their local market. For example Ontario Hydro (now Ontario Power Generation “OPG”) supplied around 90% of the very large Ontario market, and in North America generally the utilities were local regulated monopolies.

These same companies are now becoming diversified energy companies competing in open power markets. Deregulation of the markets is leading to increase competition and less certainty of economic returns to investors. In this environment capital cost, project risk, cash flow, risk of stranded assets & ROI are the current market drivers. The emphasis switches from the obligation to supply to the need to demonstrate a large and rapid return on investment.

Over the last few years these factors have led to a combined cycle gas being by far the preferred option for new plant additions. Another factor reducing the desirability of new nuclear is steady move by the vendors to large plants in the 1000–1500 MW range. Plants in this size range need investments measured in billions $US which increases the project financing challenge. In addition project schedules varied widely and in comparison to other alternatives were generally long.

In response to these market barriers, energy producers in North America such as OPG and Exelon (and others) have stated what they see as the market drivers and Requirements for new nuclear plants, which include:

- Competitive generating cost,
- High availability,
- Low O&M costs,
- Reduced capital funding,
- High rates of return,
- Short introduction schedules
- Flexible and smaller unit size,
Reduced project risk,
Improved safety, and
Assured licensability.

This key list sets the tone for future nuclear designs and markets in North America, and we now discuss the details of how to meet and exceed these requirements.

4. KEY FACTORS LEADING TO DEPLOYMENT – WHAT WILL IT TAKE?

Excellence in Current plant performance

A large fleet of well operating plants, Figure 1 (US capacity factors vs. time), safely producing economic returns for investors is the best marketing tool for new plants. With operating costs around 1 cent per KW/hr and an expectance of this continuing for 20 plus years increases the residual value of the plants and will draw investment to the operating plants and start investors thinking of new plants.

![FIG. 1. US Capacity Factors vs. Time](image)

Competitive Economics

The purchase of operating plants, for relatively modest amounts, highlights the next barrier to new plants. The capital cost of new plants, Figure 2, differs widely between projects but in all cases is substantial above coal and gas plants.

![FIG. 2. Costs vs. Plant Size](image)

Comparison: Nuclear, CCGT, Coal

Although Canada has been relatively successful in selling SMRs (the CANDU 6 product) around the world, Figure 3 (CANDU 6 Sites) but in these cases it was not in direct competition to pipeline gas. Following from the CANDU 6 and 900MW Class reactors in Canada; Canada has developed the 1000 MW CANDU 9 which at around $1500US/KW is one of the most economically attractive advanced nuclear options. Despite the CANDU 9 good economics relative to other nuclear options it will likely still likely lose out to combined cycle gas in the North American market.
Several sources noted above and the US DOE have quoted a capital cost target for new plants in North America as of order or less than $1000 US/KW. The internal rate of return must be competitive with alternate investment options and strategies. For example, AECL has adopted this target and business approach for their Next Generation design for which conceptual design will be completed in 2002.

**Flexibility in introduction and Plant size**

To meet the $/Kw cost target, some nuclear designs moved to larger plant sizes (1400 MW(e) or more). Even meeting the capital cost target is not sufficient to get the economics right. Placing a large plant in many markets will disturb the local supply/demand balance and drive down energy pricing. For this reason an SMR has a substantial benefit over a larger plant in many markets. Fighting the economies of scale is difficult, but we believe the capital cost target given above must be met with a plant no larger than 700MW and preferably 600MW. Plants of this size have the added advantages of costing less than $1B US, meet with typical market growth increments and do not unduly disturb the local energy market.

![FIG. 4. Comparison of cash flows for two assumed project durations](image-url)
Reducing the Time to Market

Time to market is also a critical parameter, as Figure 4 shows illustrating the cash flow for nuclear and CCGT identical projects with 7 and 3 year project schedules respectively for at least three reasons. First, it significantly impacts cash flow and, secondly, it allows investors to delay making commitments until closer to when the power is needed. Finally, it is easier to predict the power needs 3 years ahead than 7 years ahead.

Reduced risk in the project approval Processes

The front end of a nuclear power plant schedule is taken up with the licensing and environmental assessment of the project. These activities have been traditionally long & very unpredictable. Regulatory review can be made less risky and hopefully shortened by upfront licensing. This approach has been used in Canada, and standardized licensing (certification) for new designs has also been adopted in the USA. Processes and/or legislation associated with environment assessments need to be adjusted to reduce the schedule risk they bring to new projects.

Optimised Siting Use

The current nuclear sites are beginning to be recognised as valuable assets. Many sites have infrastructure, which will support additional units, and many sites contain retired plants. We believe that utilization of these existing sites for new plants is often the most viable option in terms of both cost and public acceptance.

In practice, this means matching the plant “footprint” to the site, and taking maximum advantage of the existing infrastructure. These include everything from the water intakes and outfalls, the site power and grid, to the existing staff and other local resources.

Public Acceptance

Public opinion polls in Canada show that many people are still fearful and wary of nuclear power. In the USA a similar view holds, with a wide spectrum of opinions. The public’s view must be supportive to obtain widespread deployment of new nuclear power plants and this needs a multiple approach.

Improved Safety Perception

Following Chernobyl the public’s concern about the safety of nuclear power reached new heights. Safe operation of nuclear plants over the following period, and in the future, is a prerequisite to acceptance of nuclear powers safety but it is insufficient. The public must be presented with designs that have safety characteristics, which are clearly safer than the current generation of plants. It is doubtful the public can be convinced with books of detailed analysis on the existing generation of plants. The use of simple arguments is to be encouraged, and the safety of the plant through good design transparent to all. For example, in this regard the Candu has in-built passive safety systems, and making these visible in the public debate is crucial. In the USA the emphasis has also been on “passive safety” designs.

Visible Environmental Benefits

Degradation of air quality is a great concern. Throughout North America the public is becoming aware of the impact of coal-fired generation on smog and other environmental
conditions and emissions. Natural gas is seen as being clean but eventually the public must be convinced that CO₂ from this source of power is undesirable. Automobiles are understood to be major polluters and running them on fuel cells is seen as a potential answer. Eventually, the need to generate electricity and from that hydrogen to power fuel cells will be recognized. A study showed that eventually all Canadian automobiles could be powered by hydrogen fuel cells with the hydrogen generated with the electricity from about ~20 Candu reactors.

The potential for nuclear to assist improvements in air quality are crucial factors in offsetting negative perceptions of the potential large releases. At the same time, excessive or overly optimistic claims are not believable so the arguments must be credible and substantiated.

**Progress towards Waste disposal**

An important concern raised in the context of more nuclear stations is the long-term disposal of spent fuel. Progress in this area would remove or reduce another of the public's concerns. In North America the eventual (if glacial) movement towards approving sites for demonstration of concepts for final waste disposal will assist this, plus the assurance of safe on-site storage. The process must proceed and a long-term plan must be followed.

**Emphasising Strategic Benefits**

Each time there is a step increase in oil prices and each time there is a conflict in areas close to the main oil supply countries the public awareness of the vulnerability of North American life to the vagaries imported fossil fuels is raised. Also governments are again beginning to see the benefits of a diversified energy supply system. There is also a growing realization of a role for nuclear energy, and that it may provide one of the few technological solutions to the world energy needs, and also advance a nation's technical stature and capability.

The non-power uses of nuclear energy are also important in the dialog on social acceptance, for medical isotopes used for therapy and diagnosis, to the sterilisation of medical equipment, and the production of hydrogen fuels.

The disposition of nuclear weapons materials, the trend towards proliferation resistant fuel cycles, and the global dialog on climate change, are all areas where nuclear has a key contribution to make and needs to be heard.

**CONCLUSIONS**

For wide deployment of new nuclear power plants several events must occur and continue:

- Current plants must continue to run well
- Life extension of these existing plants must continue
- Economics for new plants must improve
- New plants must be SMRs to ease financing costs and avoid disrupting of the supply/demand balance in a particular market
- Project Schedules (time to market) must be shortened
- Public concerns over safety, environment, waste disposal must be addressed

Within the next 10 years it is possible to meet this challenge and see a resurrection of the nuclear plant supply business, in North America and in the world.
WASP STUDY FOR SELECTION OF PROSPECTIVE REACTOR TYPE FOR UKRAINE

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Abstract

In the frame of the IAEA TC project “Strategy for nuclear energy development” a WASP (Wien Automatic System Planning package) study was carried out for selection of prospective reactor types for Ukraine. The paper discusses the results of this study. Based on national operating experience and industrial situation, it is concluded that 500 – 700 MWe water reactors will be suitable for Ukraine such as WWER-640, AP-600, VPBER-600, CANDU-6, CANDU-9 and SWBR.

In the frame of the IAEA TC project “Strategy for nuclear energy development” the WASP (Wien Automatic System Planning package) [1,2] study, the selection of prospective reactor types for the Ukraine was carried out. The WASP model is designed to determine the economically optimal expansion programme of an electric generating system over a period of up to thirty years within user-specified constraints. It utilises a probabilistic simulation for estimating the system production costs, unserved energy and reliability, as well as a dynamic programming technique for optimisation of system expansion policies.

The operating units of the Ukrainian NPPs were commissioning in period between 1980 and 1995 (Table 1). The design life time of WWER unit operation is 30 years. Thus, in the period between 2011 and 2025 in the Ukraine it should be necessary carry out an operations for the decommissioning of the respective units.

The first units for planned shut-down in 2011-2012 are Rivne-1 and Rivne-2 with WWER-440 reactors on the Rivne NPP. Therefore, activities for life-time extension these reactors should be started as soon as possible.

The selection of prospective reactor types for the Ukraine is one of important directions of the nuclear energy development strategy. Preliminary work carried out was the analysis of the designs of a new reactor generation. The main conclusion is that done that all of the reactors have improved characteristic of safety. All of them have an increased operational life-time and reduced term for construction. The probability of serious accidents is considerably reduced. Based on reactor characteristics, which were estimated, and with allowance for capabilities of maximum engaging of the domestic fuel supplier and NPP equipment the following conclusions can be made:

- all reactor types are comparable according to design safety parameters. Taking into account the operation experience, more preferable are the following modern reactor types: WWER-640, SYSTEM80 +, AWBR, CANDU-6, CANDU-9;
• from the point of view of maximum engaging of the domestic producers and usage of national operation experience of NPP units more preferable are the light-water pressurized reactors: WWER-640, AP-600, SYSTEM80 +, VPBER-600;

• from the point of view of optimal combination of a national firms capabilities dealing with the nuclear fuel cycle and the Ukrainian NPPs more preferable are the design which use the fuel of minimum enrichment or the fuel similar already used: CANDU-3, CANDU-6, CANDU-9, WWER-640, VPBER-600;

• for maintenance of a manoeuvrability of the Ukrainian power grid it is more expedient to use the NPP unit with 500-700 MW electric capacity: WWER-640, AP-600, VPBER-600, CANDU-6, CANDU-9, SWBR.

### TABLE I. NUCLEAR UNITS IN NPP OF UKRAINE.

<table>
<thead>
<tr>
<th>NPP</th>
<th>Units</th>
<th>Reactor type</th>
<th>Installed capacity (MW)</th>
<th>Building start</th>
<th>Electricity production start</th>
<th>Designed stop time</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>5</td>
<td>WWER-1000/320</td>
<td>1000</td>
<td>07.1985</td>
<td>14.08.1989</td>
<td>14.08.2019</td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>WWER-1000/320</td>
<td>1000</td>
<td>06.1986</td>
<td>19.10.1995</td>
<td>19.10.2025</td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>WWER-1000/338</td>
<td>1000</td>
<td>10.1979</td>
<td>06.01.1985</td>
<td>06.01.2015</td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>WWER-1000/320</td>
<td>1000</td>
<td>08.1986</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>2</td>
<td>WWER-1000/320</td>
<td>1000</td>
<td>02.1986</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>3</td>
<td>600-1000 ?</td>
<td>600-1000 ?</td>
<td>03.1986</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>4</td>
<td>600-1000 ?</td>
<td>600-1000 ?</td>
<td>02.1987</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

* ChNPP-1 stopped 30.11.96, in decommission regime;  
** ChNPP-2 stopped 11.10.91 after fire in machine hall, in decommission regime.
The WASP study covered the period 2000-2025. The nuclear energy expansion plan expected the commission two 1000 MW new units during 2003-2006 [3], two units during 2007-2010 [4], life-time extension units with WWER-440 reactors up to 10-15 years and six unit commission during 2019-2025 [5] to replace decommissioned units. The capacity of new units should be selected over 600-1000 MW range during the WASP study.

REFERENCES

PROSPECTIVE OPPORTUNITIES FOR SMALL AND MEDIUM SIZE REACTOR UTILIZATION IN THE ARMENIAN NATIONAL GRID

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Abstract

The use of reactors of commensurable power in Armenian National Grid is economically and ecologically expedient. However, the commensurability of nuclear units with the operating capacity of the whole power system (in a series of regimes the capacity of NPP can reach 60-70% of operating capacity of the power system) dictates the necessity for carrying out the special researches of survivability of the power system, in connection with the reciprocal influence of emergency perturbations in the power system on parameters of operation of technological systems of the NPP, and vice versa.

1. INTRODUCTION

In 1998 the feasibility report to the Government of RA "Perspectives of development of electric power industry of the Republic of Armenia, including nuclear power engineering, till 2010, with an estimation for 2020" was prepared. In the Least Cost Plan the implementation of nuclear reactors of medium size (500-700 MW) in the Armenian power system is founded.

Within the frames of the IAEA project "Energy and Nuclear Power Planning Study for Armenia for the Period up to 2020" the works on the elaboration of the development plan of the whole energy sector of Armenia have been proceeding, based on computer-implemented planning and simulation models use, with the following categories included:

- Load forecasting and load shape models,
- Reliability analysis models,
- Generation resource and capacity expansion models,
- Production or operation simulation models,
- Transmission and interconnection planning models,
- Economic or financial analysis models.

A long-term study is necessary to examine the impact of different plant sizes and locations, the installation of a new voltage level for energy transmission, the regional electricity cooperation and integration, and/or changes in the contingency protection costs. Long-term planning is necessary because the long lead-time that is needed to plan, design, obtain necessary permits, procure materials and equipment and construct and test the new facility. Long-term studies also aid the system planners to optimize today's network operation and make decisions on tomorrow's electrical network expansion.
The research shows, that the use of reactors of medium size (500-700 MW) in the Armenian National Grid is economically and ecologically expedient. However, the commensurability of nuclear units with operating capacity of the whole power system (in a some of regimes the capacity of NPP can reach 60-70% of operating capacity of the power system) dictates the necessity for carrying out the research into the survivability of the power system, in connection with the influence of emergency perturbations in the power system on parameters of operation of technological systems of NPP [1-8].

The Survivability is understood as the ability of a power system to withstand inadmissible modifications of operation parameters.

The main principles for calculation of the numerical value of the survivability are given below.

2. DEFINITION OF NUMERICAL VALUES OF SURVIVABILITY

2.1. The methodology of survivability evaluation

A methodology of calculation of the numerical values of survivability with application of Matrix and Boolean algebra and Probability theory is offered [9]. The content of the method is brought to form some rectangular matrix of the response $F$, reflecting the condition of power system when different effects influence its elements.

Let us consider the one-linear scheme of the power system, consisting of $n$ numbers of nodes, for which $m$ types of emergency perturbations and $l$ of criteria of survivability are defined. We shall present the condition of the power system, when different emergency perturbations influence its nodes, as some matrix of the response $F$

$$F = \left[ f_{ij} \right]_{i=1,2,..n; j=1,2,..m}, \quad (1)$$

where $f_{ij}$ are Boolean Functions elements and are determined by the formula:

$$f_{ij} = t_{ij} \left( \prod_{l} R_{ijl} \right)_{i=1,2,..n; j=1,2,..m; l=1,2,..k} \quad (2)$$

where

$n$ - number of elements of the power system;
$m$ - number of influences;
$l$ - number of criteria of survivability;
$t_{ij}$ - indication of an admissibility of existence of $j$'s effect in the $i$'s element of an electric power system;
$R_{ijl}$ - indication of violation of admissible limits of $l$'s criterion of survivability when there is an availability of $j$'s effect in an $i$'s element of power system.

All elements of $t_{ij}$ can be complete in the matrix $T$

$$T = \left[ t_{ij} \right]_{i=1,2,..n; j=1,2,..m}; \quad (3)$$
The Boolean elements \( t_{ij} \) and \( R_{ijk} \) are accepted as 0 and 1 only:

\[
t_{ij} = \begin{cases} 
1, & \text{if in an i's node j's type of emergency perturbation is possible;} \\
0, & \text{otherwise} 
\end{cases}
\]

\[
R_{ijk} = \begin{cases} 
1, & \text{if from j's emergency perturbation in an i's node, k's criterion of survivability goes out of a limit edge} \\
0, & \text{otherwise} 
\end{cases}
\]

The Boolean function \( f_{ij} \) (outcome of the given experiment) reflects a reciprocal response of the power system on admissible j's emergency perturbation in an i's node on any of l criteria of survivability, and can be of two values:

\[
f_{ij} = \begin{cases} 
1, & \text{if at a j's type of emergency perturbation in an i's node, even one of l criteria of survivability goes out of an admissible value;} \\
0, & \text{otherwise.} 
\end{cases}
\]

For filling \( F \) up, it is necessary to carry out \( \text{num} = n \times m \) of calculations, which amount diminishes when an inadmissible type of emergency perturbation occurs in a node:

\[
\forall t_{ij} = 0 \iff f_{ij} = 0 \iff \text{num} = \sum_{i=1}^{n} \sum_{j=1}^{m} t_{ij} \leq n \times m .
\]  

\[
(4)
\]

Let us multiply, from the right, matrix \( F \) by its transpose matrix \( F^t \):

\[
F_n = F \times F^t = [f_{ij}^n]_{i, j = 1,2, \ldots, n}.
\]  

\[
(5)
\]

Each element \( f_{i1,i2}^n \) of a square symmetrical matrix \( F_n \) displays an amount of emergencies of the same type, both in a node \( i_1 \) and in a node \( i_2 \); diagonal elements – the total amount of emergencies in a node \( i \), disturbing even one of \( l \) criteria, and \( \max (f_{i1,i2}) \) indicates the most unfavorable node of the power system when there is an effect of emergency perturbations on this node of the electric power system, as a whole.

According to the rule of multiplication of matrixes, and definition of logic multiplication accepted in the Boolean algebra, it follows, that:

\[
\max(f_{i1,i2}^n) \geq \max(f_{i1,i2}^n).
\]  

\[
(6)
\]

Really, let \{a\} and \{b\} be a set of Boolean elements. Then:

\[
\forall \{a\} \subseteq \{b\} \subseteq \{c\} \Rightarrow \{a\} \leq \{c\}; \{a\} \leq \{b\} \Rightarrow \{b\} \leq \{c\}.
\]  

\[
(7)
\]

Having multiplied the scalar matrix \( F_n^d \), obtained from diagonal elements \( F_n \), by the scalar matrix \( M^d \) of \( n \) rate, which elements are conversely-proportional to an amount of admissible emergency perturbations in an i's node, i.e.,
\[ m_i = \frac{1}{m - \sum_{j=1}^{m} f_{ij}}, \quad (8) \]

we create a matrix \( P_n \), each element \( p_{in} \) of which is probability of violation even of one of \( l \) criteria under the emergency perturbations of all admissible types, in an \( i \)'s node.

Really, the probability \( p_{in} \) (\textit{“nodal probability”}), describing influence of a malfunctioning of a node onto the survivability of the whole power system, is the ratio of the algebraic sum of the elements of an \( i \)'s line of an initial matrix \( F \) to an amount of admissible emergency perturbations in the same node, i.e.

\[ p_{in} = f_{in}m_i = \left( \sum_{j=1}^{m} f_{ij} \right) m_i = \left( \sum_{j=1}^{m} f_{ij} \right) m_i, \quad (9) \]

By the similar transformations, when multiplying from the left matrixes \( F \) by its transpose matrix \( F^t \), and, then, by multiplying the scalar matrix \( F^t_a \), obtained from diagonal elements \( F^t \), by the scalar matrix \( N_d \) of \( m \)-order, which elements are inversely-proportional to an amount of nodes, where the \( j \)'s type of emergency perturbation is admissible, we obtain a scalar matrix \( P_e \), each element \( p_{je} \) (\textit{“emergency probability”}) of which is probability of violation even of one of \( l \) survivability criteria under the \( j \)'s emergency perturbation throughout the all nodes of the power system, and it characterizes influence of this emergency perturbation onto the survivability of the whole system.

From the theory of Matrix Algebra, it is known, that a track \( Sp(F_n) \) of a matrix \( F_n \) and track \( Sp(F_e) \) of a matrix \( F_e \) are identical and are equal \( S \):

\[ S = Sp(F_n) = Sp(F_e) = \sum_{i=1}^{n} f_{in} = \sum_{j=1}^{m} f_{je}. \quad (10) \]

With the help of a track \( S \), the probability \( P_{sys} \) is determined:

\[ P_{sys} = \frac{S}{\sum_{i=1}^{n} 1/m_i} = \frac{S}{\sum_{j=1}^{n} 1/n_j}. \quad (11) \]

**Generalized Whole System Accident Rate** \( P_{sys} \) (\textit{“Total System Accident Rate”}) characterizes the \textit{inability} of the Power System to resist the emergency perturbations influence. The more is the value of the \textit{“Total System Accident Rate”} \( (P_{sys}) \), the more are the control actions from the Anti-Accident control are required for introduction of a regime into the admissible area of functioning.

It is known [10], that if the probability of \textit{occurring} of some event is equal \( P \), the probability of it not occurring, \textit{non-occurring} is equal \( 1-P \). On the basis of probabilities \( p_{in} \), \( p_{je} \) and \( P_{sys} \), the \textit{“nodal survivability”} \( (p_{in}^{S}) \), \textit{“emergency survivability”} \( (p_{je}^{S}) \) and \textit{“Total System survivability”} \( (P_{sys}^{S}) \) are defined:
\[
\begin{align*}
    p_{i}^{ns} &= 1 - p_{i}^{n}, \\
    p_{j}^{es} &= 1 - p_{j}^{e}, \\
    P_{sys}^{s} &= 1 - P_{sys}. \\
\end{align*}
\]

(12)

It is obvious, that if the value of \( P_{sys} \) is higher, the required controlling effect is proportionally lower for the introduction of parameters of the power system into the admissible area of functioning, which aids in providing the survivability.

2.2. The Algorithm of Composition of the purposeful searching vectors for necessary control actions

The indexes of survivability allow estimating quantitatively the degree of reliable functioning of the power system; and, with the help of \( F_{n}^{d} \) and \( F_{e}^{d} \) a vector of purposeful searching for necessary control actions is defined. Having arranged indexes of diagonal elements of matrices \( F_{n}^{d} \) and \( F_{e}^{d} \) in a decreasing order of these elements values, we can get the vector of numbers of nodes \( V_{n} \), ranked by an accident rate of a node, and numbers of types of emergency perturbations \( V_{e} \), ranked by an amount of nodes affected:

\[
V_{n} = [v_{i}^{n}] (i = 1, 2, \ldots, n); \quad V_{e} = [v_{j}^{e}] (j = 1, 2, \ldots, m),
\]

(13)

where

\[
v_{i}^{n} \quad \text{and} \quad v_{j}^{e} \quad \text{- values of indexes of diagonal elements} \quad f_{i}^{n} \quad \text{of a matrix} \quad F_{n}^{d} \quad \text{, and} \quad f_{j}^{e} \quad \text{of a matrix} \quad F_{e} \quad \text{, accordingly.}
\]

Let us enter mathematical definition of operation “\text{index}”: Let “\( i \)” be an index of a variable \( b \) (\( b_{i} \)), then \( \forall e = \text{index}(b_{i}) = e = i \).

According to the above mentioned definition, we can write up the values \( v_{i}^{n} \) and \( v_{j}^{e} \) as:

\[
\begin{align*}
    v_{i}^{n} &= \begin{cases} 
\text{index} \left( \max \left( f_{i}^{n} \right) \right)_{i = 1} \\
\text{index} \left( \max \left( f_{i}^{n} \right) \right)_{i = 1} \\ 
\text{index} \left( \max \left( f_{i}^{n} \right) \right)_{i = 1} \\
\end{cases} \\
\text{index} \left( \max \left( f_{i}^{n} \right) \right)_{i = 1}
\end{align*}
\]

(14)

\[
\begin{align*}
    v_{j}^{e} &= \begin{cases} 
\text{index} \left( \max \left( f_{j}^{e} \right) \right)_{j = 1} \\
\text{index} \left( \max \left( f_{j}^{e} \right) \right)_{j = 1} \\ 
\text{index} \left( \max \left( f_{j}^{e} \right) \right)_{j = 1} \\
\end{cases} \\
\text{index} \left( \max \left( f_{j}^{e} \right) \right)_{j = 1}
\end{align*}
\]

(15)

The upper elements of the vector \( V_{n} \) indicate those nodes of the power system, where it is necessary first of all realizing the anti-accident measures; and the values of elements of vector \( V_{e} \) indicate how much more the given emergency perturbation effects unfavorably onto survivability of the power system, allowing to define its own type of the Anti-Accident control.
3. CRITERIA OF SURVIVABILITY, INCLUDING SAFETY OF THE NPP

As was noticed above, the assurance of survivability means exclusion of the large scale system accident development. Such emergencies are accompanied by cutting off a significant part of consumers and power plants, as well as/or by dividing the power system into the synchronously working separate regions, with the complete blackout of the areas linking them.

One of the most important criteria of survivability is the stability of the power system under dynamic disturbances. The symptom of the system instability is the unlimited increase of some part of relative angles of generators which are oriented towards the arbitrarily chosen synchronously rotating axes [11-14].

In the power systems with the large generation of electricity by the NPPs, it is necessary to consider the singularities of the NPP operation.

According to common principles, the NPP is considered safe, if during its long operation under all conditions, including emergency, serious damage of the fuel rod in the reactor core is eliminated; and also, the localization of radioactive emission, and appropriate NPP's personnel protection, as well as the protection of the neighboring population and environment from the radiation effect should be ensured.

The emergency perturbations in the power system can immediately result in origination of such emergency regimes of NPP operation, as:

- a regime accompanied with the emergency reduction of coolant flow, as well as of feed-and make-up water;
- cutoff of the NPP auxiliaries;
- operation of reactor facility under the unexpected dumping and increasing of an electrical load, and also, during other emergency situations at the power unit, which depend on the work conditions of the power system.

In this case, the survivability criterion for such kind of power system should be not only the stability of accident- and after-accident regimes, but also the keeping in admissible limits the basic technological parameters of the NPP should be such a criterion.

The reducing of system frequency, as well as of voltage on bus-lines of NPP, calls a diminution of turnovers of drives of NPP auxiliaries, and, as a result, reduction of the MCPs and FPPs capability.

When the frequency and voltage in a system are reduced, two factors should be indicated, which can cause the scram of the unit, or the reduction of its power, with the aid of technological protection system. It will aggravate even more the emergency situation in the power system:

1. Lowering the coolant flow-rate and, as a result, its temperature increase up to the emergency level on the output of the reactor, can cause the actuation of the emergency protection system, and the reactor capacity reducing.
2. The decreasing the water level in a steam generator when both the frequency and voltage fall low enough and this condition lasts long, can cause the steam generator emergency protection actuation, and reactor scram.

From the above-stated follows, that the important controlled parameters of the NPP are: the coolant flow, feedwater flow-rate, steam generator steam pressure, and a temperature difference in the reactor core.

Thus, the criteria of power system survivability, from the view of the NPP security assurance, are as follows:

I. The dynamic stability.
II. The frequency of the power system.
III. The neutron capacity of the reactor facility.
IV. The electrical capacity of the MCP.
V. The steam pressure in the steam generator.
VI. The temperature differences in the reactor core.

Thus, the dynamic regime does not influence the survivability of the power system, if during the transient process the values of parameters of any of above mentioned criteria of survivability do not go out of the admissible limits.

4. THE SURVIVABILITY OF ARMENIAN POWER SYSTEM

Using the above mentioned algorithms the Survivability of Armenian Power System was calculated for two scenarios:

1. The isolated regime of operation,
2. Parallel, with the power systems of the neighboring countries, regime of operation.

Researches show that the Total System Survivability of Armenian National Grid which includes NPP with the capacity of not more than 600 MW, in the first scenario is $P_{sys}^S=0.952$ for winter regimes, and $P_{sys}^S=0.948$ for summer regimes of operation.

We have better result in the second scenario, for which the Total System Survivability is $P_{sys}^S=0.960-0.992$ depending on the neighboring country's power system connection.

For both scenarios the introduction of the appropriate tools of anti-accident control allows to reach security of a sufficient level of survivability, as well as reliability and safety of operation of the reactors of such capacity.

CONCLUSION

The offered algorithms and criterion of survivability allow to analyse the problems of influence of the power system emergency situations on the reactor operation parameters. Such researches allow judging about the perspectives of operation of small and medium size
reactors within the Armenian National Power Grid, from the view of necessity of the ensuring the reliable and safe operation of such reactors in emergency situations of the power system.

Thus, basing on the analysis, we can conclude that the use of the reactors with capacities more than 500-600 MW in Armenian National Power Grid would be not expedient, taking into account the emergency situations in the power system.

REFERENCES


STRATEGY FOR INTRODUCTION OF SMALL AND MEDIUM REACTORS IN INDONESIA

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Abstract

The paper provides a perspective of the nuclear energy status and prospects in Indonesia. Since the late 1970s, the National Nuclear Energy Agency (BATAN) has been involved in convincing the Government of Indonesia on the need to embark on a nuclear power programme. However, no state-owned company is in a position to undertake large investments, hence no nuclear power plant could be built in Indonesia in the foreseeable future. A possible breakthrough would be the introduction of SMR in the form of the fourth generation of nuclear power plants whose goal is low generation cost, able to compete with gas combine cycle; high safety, free from catastrophic accidents; low amount of waste; and proliferation proof. A revival of the nuclear option would be possible under the following conditions: small or medium power reactors such as the South African HTR could be realized at the estimated costs claimed; private investment in nuclear energy is enabled through amendment of existing Indonesian laws; support in Parliament and from NGOs is available. Manpower development at an early stage is also essential for successful programme implementation.

1. INTRODUCTION

Slightly more than one billion people in the developed countries (about 20% of the world’s population) consume 60% of the total energy supply whereas just under five billion people in developing countries consume 40% of the total energy supply.

The challenge is to provide the minimum services, including energy services to allow those five billion people to achieve a decent standard of living.

The World Energy Council Statement 2000 is to examine what has happened since the WEC published its 1993 Report “Energy Tomorrow’s World, the Realities, the Real Option and the Agenda for Achievement” and to delineate the key contemporary policy activities.

Most current energy consumption derives from fossil fuels such as coal, oil and gas, the products of life forms that were buried and transformed into mineral resources from prehistoric times. Since these resources are finite and non-renewable, any civilization that depends on them will not be sustainable in the long run.

While the depletion of fossil fuels may not be a problem now, environmental degradation associated with the burning fossil is already apparent. Mounting fossil fuel consumption has caused acid rain, as a result of large quantities of sulfur and nitrogen oxides being spewed into the atmosphere. The burning of fossil fuels is necessarily accompanied by the emission of Carbon dioxide, of which mounting concentration will cause global warming, triggering climatic changes beyond human repair.

Modern civilization labours numerous contradictions, the classic one being the trilemma posed by the conflicting needs for economic growth, energy and resources and environmental conservation.
Everyone agrees that the solution to this trilemma must be found in a more sustainable resource use. When it comes to the supply side of energy, Greenpeace proposes a very rapid reduction in the consumption of fossil, banning hydropower and nuclear energy at the same time.

This idea would make us over-dependent on technological breakthroughs in the development of renewable energy options. Moreover, the advantages of some renewable options are accompanied by certain disadvantages. Renewable energy options (photo-voltaic, wind, biomass, etc.) definitely require further development if they are to make a significant contribution to the future energy supply mix, but during the transition period, the industry still will be needing fossil as well as nuclear energy.

This paper will describe the realities and facts the real options and plan for actions to introduce small and medium reactors in Indonesia.

2. FACTS AND REALITIES

The National Nuclear Energy Agency (known with the acronym BATAN), formerly named the National Atomic Energy Agency, has in the past been involved on several occasions in attempts to convince the Government of Indonesia on the need to embark on a nuclear power programme. The first was in 1979-1980 when the Government of Italy provided a technical assistance grant to conduct a feasibility study for the construction of the first nuclear power plant. Although economic conditions were then favourable, among others as an oil exporting country it was important that the international price of oil was at a high level which was the case at that time, other conditions were not too favourable: the Three-Mile Island incident had just occurred in May 1979, and the Java system grid peak load had not yet reached 3000 MW (although it was growing very rapidly at 15 per cent annually).

The second attempt was begun in 1984 with a request to the IAEA to update the earlier feasibility study, with concurrent technical assistance grants in 1985 from the Governments of France and of the United States implemented by Sofratome and Bechtel. This effort at first appeared promising, and the Java-Bali system was already 5000 MW, and it was still growing at 15 per cent annually. But it was frustrated by the oil price crash of 1986 and consequently it was not possible anymore to assume the price of coal to be around $ 50/ton (the Chernobyl accident in April 1986 effectively killed this proposal).

Our third failed attempt began in 1994 with the conduct of a feasibility study by NewJec through a loan provided by the Government of Japan. The study, which included site preparation investigations, produced positive results by 1996. The results were reviewed and its results confirmed by another consultant in 1997, but unfortunately the Asian crisis struck and because of the severe devaluation experienced in 1997-1998 the study results became invalidated. (At this time a new law on nuclear energy was enacted in 1997 which gave our organization its new name and which also established a separate nuclear regulatory agency.)

The economic and financial crisis which began in 1997-1998, and which has acquired other dimensions such as political and social, has persisted until today. The energy sector has probably saved the country from disintegration, since the country has continued to obtain foreign exchange earnings through oil, gas and coal exports, and in addition the economy is being sustained with huge inputs of energy subsidies from the government budget, in effect being funded from energy export earnings. In March 2001 the price of gasoline is 10 cents/litre and the average electricity price is less than 3 cents/kWh (calculated with an exchange rate of Rp.10400/$). The Government has found a great deal of resistance in its
attempts to raise the price of oil products and electricity tariffs. In particular, previous governments have kept the price of kerosene and diesel oil at very low levels for a number of decades (current retail price about 6-7 cents/litre).

The Government’s announced plans, however, are to remove energy subsidies gradually; this hopefully could be achieved by 2005. The energy sector is also to be restructured: regulatory agencies are to be established for the upstream and the downstream sectors of the oil and gas business, and also to oversee the power sector, with state-owned companies only dealing with narrowly defined areas for their operations. Private companies will be provided with broader and more level playing fields. (The role of the central government in most sectors would also be diminished, with more authority to be vested with provincial and local governments). For this purpose two new draft laws on oil & gas and on electricity have been submitted to the parliament this year. The success of restructuring the energy sector would of course depend greatly on the degree of success in removing energy subsidies.

The economic, financial and multi-dimensional crisis that Indonesia is experiencing today has put the country in a precarious position. This may be gleaned from the following figures on Indonesia’s debt situation. (Quoted from Global Economic Forum [Morgan Stanley Economists] February 22, 2001).

“(1) Total public debt, denominated in both foreign and local currencies, is estimated to be around US$152 billion, or 99% of GDP. Total central government debt is estimated at approximately US$134 billion, or 87% of GDP.

(2) Total private debt, denominated in both foreign and local currencies, is an estimated US$110 billion, or 72% of GDP. About 75%, or US$82 billion, represent foreign currency-denominated debt.

(3) The estimate for total medium- and long-term external debt is US$151 billion, or 98% of GDP. The public sector accounts for US$91 billion, or 60% of such debt.

(4) Total short-term external debt is estimated at around US$31 billion, or 20% of GDP. Almost all this debt is related to the private sector.

Items (1) to (4) give a fairly accurate snapshot of the overall debt profile of the economy:

(5) The country's total indebtedness is approximately US$262 billion, or 170% of GDP. The public sector accounts for US$152 billion, or 58%.

(6) Total external indebtedness of the country is approximately US$182 billion, or 118% of GDP. The public sector represents US$91 billion, or 50%.

Items (5) and (6) show clearly that the public and foreign sectors are key components of the debt problem.”

Thus, no state-owned company is in a position to undertake large investments, implying that no nuclear power plant could be built in Indonesia for the foreseeable future. There is no precedent of investors’ spending money in a nuclear power plant project in a foreign country, since developing countries have been discouraged in building nuclear power plants. The exceptions among developing countries have been those countries who have been able to secure their own financing or those who have provided state guarantees.

The Java-Bali interconnected system represents 80% of the whole Indonesian electricity Consumption. In the year 1997 the total electricity generated by PLN in Java-Bali system is about 61.9 TWh and 15 TWh generated by captive power. The Industrial sector is the largest
consumer (45%), followed by the household sector (36%), commercial sector (13%), and others (6%) as shown in Figure 1.

The electricity demand projection can be estimated based on the macro economic and population growth rates projected by the National Development Planning Agency (BAPPENAS) and also the goals on electric generation in the National Long Term Development Planning. The electricity demand and generation capacity projected by Directorate General of Electricity and Energy Development (DJLPE) is shown in Figure 2. In this study the peak load is projected, based on the latest economic figures, i.e. the economic growth –13% in the year 1998 and 1% in 1999. This projection is from the National Electricity Enterprise (PLN), based on their expansion-planning program.

![FIG. 1. Consumers of the Java-Bali System by Demand Sectors](image1)

![FIG. 2. Peak Load Forecast](image2)

Based on the IAEA guide book in the Introduction of nuclear power for the successful introduction of nuclear power in any country, the first essential requirement is a clear understanding at the decision making level of the specific aspects of nuclear power, and a thorough knowledge of the tasks and activities to be performed as well as the requirements, responsibilities, commitments, problems and the constraints involved.
The country has committed itself to the fulfilment of the requirements and has to establish clear policies to ensure continuity of the programme.

Changes in policies or in management may cause high penalties in money and time. Table 1 shows the requirement and status of NPP activities in Indonesia. With the design, constructions and operation of the Multipurpose Reactor and its supporting laboratories from 1982 to 1989, which covered fuel technology, waste technology, safety technology, QA/QC and a qualified management group, Indonesia since 1987 is technically ready to embark in NPP projects.

As a promoter of Nuclear Power Plant in Indonesia, BATAN has to face non-technical challenges such as: 1) commitment at the national level; 2) public information activities; 3) financing; 4) subsidy for electricity.

3. OPTIONS AND PLAN FOR ACTION

There are five main energy policy measures in Indonesia:

- **Diversification**: to maximize and economize the supply of energy, to curb the rate of excessive use of hydrocarbon resources, to reduce the dependence on a single type of fuel (i.e. petroleum) and later to replace it with other available fuels. In 1995 oil shares was around 60%, and in 2020 is projected to be around 40%;
- **Intensification**: to increase and expand the exploration of the available energy sources; aiming to secure sufficient supply of energy;
- **Conservation**: to economize energy production and utilization;
- **Energy Price**: to formulate energy prices based on economic values and by taking into consideration its environmental cost;
- **Clean Energy Technologies**: to support the environmental program and towards a sustainable development.

The implementation of the energy policy covers several aspects such as the issuance of regulations, standards, energy-pricing incentives and disincentives, and the application of appropriate technologies. The technologies that would be considered are identified as follows:

- Technologies to produce substitutes for oil, as oil is non-renewable and is a very limited resource.
- Technologies to support a more sustainable energy supply
- Technologies for clean and efficient energy to support environmental programs and towards sustainable development.

Indonesia has many types of energy resources, such as fossil fuels, uranium (for nuclear energy), and other renewable energy resources, etc.

A very preliminary study performed in early 2001 shows levelized generation cost which also include Externalities Cost for both coal and nuclear (table 2 and table 3).
TABLE I. ACTIVITIES REQUIRED FOR NPP PROGRAM, INDONESIA CASE

<table>
<thead>
<tr>
<th>ACTIVITIES</th>
<th>STATUS</th>
</tr>
</thead>
</table>
| Long Term Nuclear Power Programme Policy and Commitment | Study I
1969 WASP.
No commitment at national level | Study II
1982 NIRA | Study III
1991-1996 NEWJEC | Study IV
2000-2002 C.S. /IAEA
1982 Decision to build demo plant – cancelled |
| Organization for NPP Management | Since 1986 (Center for the Nuclear Energy Studies) |
| Organization for Project Implementation | Since 1984 (Nuclear Industry Project Management - Technical Task Force Unit) |
| Organization of the Regulatory Body | Since 1986
One roof with BATAN (Promoting + Regulating) | Since 1999, Separate Body Promoting vs. Regulating
BAPETEN as Regulatory Body, Since 1999 (175 persons) |
<p>| Legal Framework | Nuclear Act No. 10 of 1997 |
| Licensing and Authorization | Since 1984, Bureau of Atomic Energy Controlling (BPTA) |
| Nuclear Liability | Since 1997 by Nuclear Act No.10 |
| Manpower Development Requirements | Since 1977 by Center for Education and Training – BATAN |
| QA and QC Requirements | Referring to IAEA Safety Series No. 50-C-QA |</p>
<table>
<thead>
<tr>
<th>ACTIVITIES</th>
<th>STATUS</th>
</tr>
</thead>
<tbody>
<tr>
<td>National Participation</td>
<td>25% for NPP Unit 1&amp;2 (NEWJEC study, 1993-1995)</td>
</tr>
<tr>
<td></td>
<td>30% for first NPP of twin unit (GOI-Westinghouse study, 1996)</td>
</tr>
<tr>
<td></td>
<td>26.1% for NPP unit 1&amp;2 (BATAN-GE study, 1997)</td>
</tr>
<tr>
<td></td>
<td>Start with QA Class items and carbon steel products, incentive to bidder, establish national participation plan (BATAN-MHI study, 1998)</td>
</tr>
<tr>
<td>Financing Requirement and constraints</td>
<td>1995 and 1997, Several studies were performed</td>
</tr>
<tr>
<td>Sources of Financing</td>
<td>Not available (NA)</td>
</tr>
<tr>
<td>State System for Physical Protection</td>
<td>Since 1987, BATAN Security System (BSS)</td>
</tr>
<tr>
<td>State System of Accounting for and Control of Nuclear Materials</td>
<td>Since 1967 updated in 1987</td>
</tr>
<tr>
<td>Public Information and Public Relations</td>
<td>Available with low budget. Small group (NGO) against NPP</td>
</tr>
<tr>
<td>Electrical System, Interconnection, Pricing policy / subsidy</td>
<td>Java – Bali interconnection system since 1982 Subsidized</td>
</tr>
<tr>
<td>Disposal of Radiation Waste</td>
<td>Since 1989, Radioactive waste treatment plant (RWTP) national level</td>
</tr>
<tr>
<td>Nuclear Fuel Cycle</td>
<td>Since 1988, prototype plant</td>
</tr>
<tr>
<td>Siting for NPP</td>
<td>Since 1996, IAEA support for data site permit</td>
</tr>
<tr>
<td>Role of Nuclear Energy within a National Energy Plan</td>
<td>National Policy that nuclear as the last option after fossils</td>
</tr>
<tr>
<td>Electric System Expansion Planning</td>
<td>No decision to go nuclear</td>
</tr>
<tr>
<td>Survey of Energy Resources</td>
<td>1997 status Oil 8.898 billion barrel (1% world reserve)</td>
</tr>
<tr>
<td></td>
<td>Gas 216.8 trillion std.cubic feet (2% world reserve)</td>
</tr>
<tr>
<td></td>
<td>Coal 5 billion tons (3.1% world reserve)</td>
</tr>
<tr>
<td>Energy Demand Forecast</td>
<td>Short (1 – 5 years), medium (10 – 15 years), long term (20 – 30 years); Comprehensive study supported by IAEA (2001 – 2002)</td>
</tr>
</tbody>
</table>
TABLE II. LEVELIZED GENERATION COST

<table>
<thead>
<tr>
<th>Unit</th>
<th>Coal-fired w/o FGD</th>
<th>Coal-fired with FGD</th>
<th>Nuclear PP</th>
</tr>
</thead>
<tbody>
<tr>
<td>Capital cost</td>
<td>Mills/KWh</td>
<td>19.63</td>
<td>21.61</td>
</tr>
<tr>
<td>O&amp;M+decommissioning</td>
<td>Mills/KWh</td>
<td>6.27</td>
<td>8.50</td>
</tr>
<tr>
<td>Fuel cost</td>
<td>Mills/KWh</td>
<td>18.12</td>
<td>18.12</td>
</tr>
<tr>
<td>Generation Cost</td>
<td>Mills/KWh</td>
<td>47.98</td>
<td>49.05</td>
</tr>
</tbody>
</table>

TABLE III. SOCIetal COST

<table>
<thead>
<tr>
<th>Unit</th>
<th>Coal-fired w/o FGD</th>
<th>Coal-fired with FGD</th>
<th>Nuclear PP</th>
</tr>
</thead>
<tbody>
<tr>
<td>Generation cost</td>
<td>mills$/KWh</td>
<td>47.98</td>
<td>49.05</td>
</tr>
<tr>
<td>Externalities cost</td>
<td>mills$/KWh</td>
<td>0.9094</td>
<td>0.8156</td>
</tr>
<tr>
<td>Societal cost</td>
<td>mills$/KWh</td>
<td>48.89</td>
<td>49.86</td>
</tr>
</tbody>
</table>

The externality costs for nuclear are relatively low because of no release of CO2 and nuclear waste to the environment.

Another option which is still being considered is to have gas pipe lines from Natuna area to be distributed to Java Island, therefore, a gas scenario for Java-Bali network besides coal and nuclear.

Of course renewable energy should also be considered and carefully follow the status of technology, because when it comes to availability, cost level, efficiency, and environmental impact, every single option for production of energy brings with it a combination of pros and cons.

The option we have is Fossil, Renewable and Nuclear. Greenpeace would like to go to Renewable by excluding fossil and nuclear, but with the status of technology at the moment is premature to jettison fossil and nuclear.

4. SMALL AND MEDIUM REACTOR

As a possible breakthrough would be the introduction of SMR in the form of the fourth generation of NPP. The goal is to introduce a nuclear option, which is:

1. Low generation cost, able to compete with gas combine cycle.
2. High safety, catastrophe free.
3. Low amount of waste.
4. Proliferation proof.

Considering the introduction of SMR in Indonesia, following is a table describing the constraints, requirements as well as status of SMR Gen IV.
### TABLE IV. REQUIREMENT TO INTRODUCE NPP IN INDONESIA

<table>
<thead>
<tr>
<th>No.</th>
<th>Constraints</th>
<th>Requirements</th>
<th>SMR Gen IV</th>
</tr>
</thead>
<tbody>
<tr>
<td>2.</td>
<td>Public Acceptance (NGO’s)</td>
<td>Gen IV NPP</td>
<td>Information and Result of Pilot Plant</td>
</tr>
<tr>
<td>3.</td>
<td>Financing</td>
<td>Capital Cost Similar to CCG</td>
<td>MHTR &lt; 1000$/kw</td>
</tr>
</tbody>
</table>

The outcome of SMR pilot plant, which fulfils Generation IV criteria, will act as a strategic point to challenge constraints for the introduction of nuclear power plant in Indonesia.

It would be appreciated if today’s meeting on SMR in Cairo can provide us the hint that could fulfil the requirements for most developing countries. Many conceptual designs or even detail designs of SMR are being performed such as CAREM, SMART etc. and we will be very happy if the plant could be realized as a Pilot Plant.

If we follow the South African design and construction of PBMR this may be fitting for Indonesia, for a medium term plan, in conjunction with the Indonesian economic recovery.

Indonesia, in small scale has already participated in this endeavour. This is the agenda to be achieved.

**CONCLUSION**

Under certain conditions, prospects for a revival of the nuclear option would be possible. These conditions are:

1. Small or medium power reactors such as the HTR being developed in South Africa could be realized at the estimated costs claimed; the earliest date for this achievement is 2004.
2. Since the HTR is modular at 100 MW units, a plant with several units could be built in the Java-Bali system; single or double units could also be built outside Java, for instance in northern Sumatra or even possibly southern Sumatra.
3. Private investment in nuclear energy is enabled through amendment of existing law; this would need to be actively pursued by BATAN and other stakeholders.
4. Support in Parliament and from NGOs is essential for the above. For this purpose the Comprehensive Assessment Study currently being under-taken with IAEA assistance is an important first step to establish the idea that nuclear energy remains a plausible option for Indonesia in the future.
5. Global support for SMR development, especially in view of lack of mitigation measures to counter global warming, will be an important factor.
6. Manpower development at an early stage is also essential for successful programme implementation.
REFERENCES

SOLUTIONS LEADING TO INCREASED DEPLOYMENT OF SMRs

(Session 14)

Chairpersons

S.B.A. Hamid
Egypt

P.E. Juhn
IAEA
NUCLEAR REGULATION IN 2000

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Abstract

A key prerequisite to the development of a nuclear program to provide a country’s electricity is that governments and the public are confident that a nuclear accident “will not occur” over the whole life of the program. This can only be achieved by three elements being equally strong: an excellent design, a highly competent and safety conscious operator, and a strong competent regulator with appropriate legal powers.

This paper will outline what is necessary to achieve the third of these prerequisites. It will outline the principles by which a Regulatory Agency ensures that licensees meet their responsibility for the safety of their nuclear plant, based on both internationally accepted ideas, the Nuclear Safety Convention, and Canadian practice. The paper will cover legislative principles – what to include in legislation, what to include in regulations, and what to include in standards and guides. The problem of “prescriptive requirements” versus “general performance statements” in regulations will be addressed, and the implications of this problem on ensuring licensees retain responsibility for safety, and on licensees’ desire for a high degree of certainty in what is expected of them, both to get a licence and to keep it, in today’s economic climate.

The paper will also address compliance and the differences between the lawyers’ definition of compliance (meeting specific requirements defined by law) and the nuclear safety engineers’ view of compliance, (meeting commitments made at the time a licence was given), and how these views can be reconciled.

The paper will discuss how Canada’s new Nuclear Safety and Control Act has addressed some of these issues, and how the Canadian Nuclear Safety Commission is implementing the new Act.

The issue of transfer of regulatory programs and technology to the regulatory agencies of countries buying a nuclear plant from Canadian companies will be discussed, and examples given of how this has been accomplished to the benefit of the regulatory agencies and the benefit of safety of nuclear power plants in both the receiving country and in Canada.

1. INTRODUCTION

The economic development of the world, and the globally interrelated nature of its trade, is inevitably leading to steadily increasing levels of energy usage in all developing countries. Global warming, the Rio Declaration, the Kyoto Convention, and, probably most important, the desire of developing countries to avoid the pollution mistakes of the industrialized Western countries, all point to a steady increase in the use of nuclear energy in developing countries – and increasingly efficient use of nuclear energy in developed countries. Nuclear energy is being recognized more and more as a clean energy source – provided an accident which results in radioactive material being released to the environment does not happen.

The fear of nuclear accidents is probably the greatest stumbling block in the minds of both the public and of politicians to the use of nuclear energy as a means of producing electricity. Hence, if you ask an ordinary citizen what the primary job of a nuclear regulator is, it is likely he would say – “make sure a nuclear accident like Chernobyl doesn’t happen here”. The task of designers, operators and regulators is to turn this plain language statement into reality for the full life of the station. This is not an easy task. This paper discusses some elements that
must be in place to be able to demonstrate to the public and the politician be that it can be met.

Comparison of the serious accident rate that must be achieved by the nuclear industry with the serious accident rate that occurs in normal industrial activities shows that the nuclear industry has to be at least two orders of magnitude more safe than most other industrial undertakings if it is to meet the plain language statement quoted above. Only then can we say that the likelihood of a serious accident is so low that it, in essence, will not happen. As engineers, we can turn this safety expectation into a numerical probability of severe core damage, and another numerical probability for severe core damage together with an impairment of the containment. We can turn these requirements into specific reliability requirements for systems important to safety, into requirements for defense in depth and all the other requirements for hardware that make up a sound design. Nuclear safety is much more than sound design, however. The probability of a serious nuclear accident has to be maintained at a very low likelihood for the entire operating life of the reactor; perhaps for up to 60 years. This can only be achieved by the combination of a highly competent designer, a highly competent operator and a highly competent, independent, regulator existing throughout this period of time. All three elements are essential if this challenge – that is a keeping the probability of causing significant harm to the population at least two orders of magnitude less than is commonly achieved by well-run industrial activities – is to be met.

No country should contemplate using the atom as a source of electrical energy if it is not prepared to put the legislative process in place, to make the financial resources available to set up a competent regulatory body, to employ the necessary number of competent experienced regulatory staff to provide real independent oversight of the industry, and to regularly benchmark its regulatory agency against others around the world. This paper discusses in more detail what is needed to set up and maintain a competent regulatory body.

2. THE NUCLEAR SAFETY CONVENTION

Articles 7 to 11 of the Nuclear Safety Convention, adopted on June 17, 1994 by a Diplomatic Conference in Vienna at the IAEA, and which came into force on October 24, 1996 defines what is needed in broad terms.

Article 7 states

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

2. The legislative and regulatory framework shall provide for:
   (i) the establishment of applicable national safety requirements and regulations;
   (ii) a system of licensing with regard to nuclear installations and the prohibition of the operation of a nuclear installation without a licence;
   (iii) a system of regulatory inspection and assessment of nuclear installations to ascertain compliance with applicable regulations and the terms of licences;
   (iv) the enforcement of applicable regulations and of the terms of licences, including suspension, modification or revocation.
The most important issue here is that the regulatory body must be able to prohibit the operation of an installation, and enforce the requirements for safety by, if necessary, suspending, modifying or revoking a licence.

Ordering an operating reactor to be shut down will likely have significant impact on a country, particularly if the plant output is large compared to the total capacity of the grid. For this reason, the Regulatory Body must be separate and independent from any other body or organization concerned with promotion or utilization (Article 8). This means, in the author’s view, that a degree of independence is necessary not only from the industry, but also from that part of a country’s Government that is responsible for the economic development of the country.

Article 7 of the Convention also states that the regulatory body be given the power to make regulations, to establish national safety requirements, and to assess and inspect what is actually going on.

3. REGULATIONS

Passing regulations and setting requirements has to be done with care and due regard for a country’s cultural norms. Article 9 of the Convention places prime responsibility for the safety of a nuclear installation on the licensee. Every regulation or requirement that is passed affects, to some degree, the responsibility of the licensee, and places responsibilities on the regulator to ensure the regulations and standards are sufficient and necessary. Standards in particular represent the accumulated wisdom about what works. They also provide a degree of stability and predictability to a licensee on what is necessary to obtain, and keep, a licence. Stability and predictability of the regulatory process itself and of its requirements are essential elements if utilities are to raise capital from financial institutions to build and operate a nuclear power plant safely.

In Canada, our approach in the Regulations that have been enacted under the Nuclear Safety and Control Act is, whenever possible, to develop regulations that place a general obligation on the licensee, rather than define detailed technical requirements. For example, section 24(4) of the Nuclear Safety and Control Act itself simply states:

“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.”

This is a very general provision which is supported further in the Regulations made pursuant to the Act. The Regulations for nuclear power plants state, for example, that an application for a licence must contain information which describes the proposed measures to prevent or mitigate the effects of accidental releases to the environment. Our policy requires that these measures include, for example, two independent and fully capable shutdown systems. A
hierarchical structure is formed by this approach. The Act defines very general principles, the
Regulations give general performance requirements, and Commission policies and standards
provide the necessary detail. We believe that this hierarchical structure forms a coherent
legislative framework.

We believe there are a number of advantages to this approach. Firstly, it maintains the
licensee’s primary responsibility. Licensees cannot “hide” behind a regulation. Secondly, it
does not unduly fetter the regulator. We can, and do, develop detailed standards and guides on
what is required. An example of this is our approach to Human Factors. Article 12 of the
Convention states that appropriate steps to ensure that the capabilities and limitations of
human performance are taken into account throughout the life of a nuclear installation. The
Canadian regulation requires that information be submitted that defines the applicant’s
management structure, the proposed measures, policies and procedures for operating and
maintaining a plant, and for training workers, etc. This information is needed to ensure
“adequate provision” has been made. Given the importance of the human element, further
guidance is given through a Commission policy on human factors that states that, when
reviewing applications for a licence, the Commission will take into account human factors
that could impact on the protection of the environment, health and safety, etc. The full text is
given in P-119, Policy on Human Factors. More details still are given in a draft Regulatory
Guide, Human Factors Engineering Program Plan (C-276) and other documents.

The technique of writing regulations in a way which sets general expectations, supported by
technical standards and guides, also make it possible to utilize the safety standards and guides
developed by the IAEA. These represent today’s international consensus on what constitutes
best practice. They are not generally written in a form which is suitable for incorporation into
formal regulations and were not drafted with that in mind. Regulations which set a general
principle followed by a policy statement by the regulator on what would be acceptable
practice to meet the general statement and which calls up the appropriate IAEA standard, may
be a relatively straightforward way for developing countries to set up a scheme for licensing
most types of reactor.

4. COMPLIANCE

The sanction a Regulatory Body has in the licensing assessment phase is not to give a licence
to the applicant until it is satisfied with the license application. The submissions made by an
applicant must meet the standards expected and, until they do, a licence is not issued. This is a
very powerful financial sanction. Provided the regulator has the power to not issue a licence,
submission of inadequate information does not need to be made an offence! Once a licence
has been issued, and the plant is in operation, the regulatory regime needs to change to a
compliance mode. The challenge is now different: the risk is now real. The licensee, in
obtaining a licence, has stated in a large number of submissions, how he intends to operate,
maintain and modify the plant over its lifetime of 40 to 60 years. He has described in much
more detail than any standard or guide can do what he considers necessary to maintain safety.

The information the licensee submitted to the regulator at the time of licensing describes
everything that is needed to assure safety. Some of this information is clearly much more
important than others. Some of it can be turned into directly enforceable requirements that a
failure to meet would be an offence in law. This can be achieved in the traditional fashion by
incorporation into a regulation or by incorporation into a condition of the license.
It is not possible, however – or desirable – to make every important requirement that is needed to be met to ensure a very low probability of an accident throughout the reactor lifetime, into a directly legally enforceable statement. It would mean reams of very technical regulations, or many more conditions in a licence than is normal practice. It would also be very difficult to capture what the licensee has committed to doing to maintain safety in his plant in a regulation. It should be remembered at this point in the discussion that a high level of safety is much more than good design and well maintained hardware. Safety includes the “softer” issues – configuration control, training, good communications between the operating core of an organization and the Board of Directors, maintenance of the knowledge of the safety case – just to list a few examples.

Two techniques can be used to solve this difficulty. The first is to quote licensees’ own commitments as a licence condition. In Canada, the “Operating Policies and Procedures” defined by a licensee, are referred to in a licence condition, and become, as a result, a legally enforceable requirement. A licensee’s internal rules for the radiation protection of its workers are likewise referenced as a licence condition. This, again, maintains licensee responsibility for safety.

Another technique currently being developed in Canada, in addition to the licence condition, is to extract the major points from a licensee submission and develop “Compliance Criteria” from it. These then form part of the compliance program followed by an inspector. In essence, the licensee is being held accountable for the commitments he made when awarded a licence. Unless compliance criteria are also cited as a licence condition, they are not directly enforceable in a court of law. However, the licensee has made the commitment in his licensing submission to meet a more generally stated regulation. Hence, failure to meet the commitment or the compliance criteria developed from it, can be used as evidence that the more general requirements in the regulation have not been met, which would be an offence.

5. TRANSFER OF REGULATORY TECHNOLOGY

A Canadian company, Atomic Energy of Canada, has been successful in selling its products in the international market. The CANDU reactor which it has designed has many unique features when compared with a Pressurized Water Reactor. The regulatory requirements that have been developed over the last 40 years reflect many of the unique features of the design. An example that illustrates this point is that there is no regulatory requirement to maintain a Departure from Nucleate Boiling Ratio (DNBR) greater than a specified value as there is in a PWR. In a CANDU reactor, fuel can have a limited degree of dryout without damage. The differences in the design give rise to some differences in the regulatory requirements which need to be understood by the Regulatory Body of the purchasing country. This represents a challenge for the Canadian regulatory agency, the Canadian Nuclear Safety Commission. A sale of a CANDU reactor to another country inevitably leads to a need to transfer to the purchasing country the regulatory technology that has been developed. Training and support to Romania, Korea and China have all been given in the last few years. The CNSC’s philosophy here has been that an accident anywhere is an accident everywhere: it is in Canada’s interests to have strong independent, knowledgeable regulators of other countries that operate CANDU plants as colleagues with whom to exchange regulatory experience over the lifetimes of these plants.

Senior representatives of all the countries with CANDU power plants meet annually under the auspices of the IAEA to exchange regulatory experience. This provides an ongoing transfer of
regulatory technology. It is a mutually beneficial process, and we would like to thank the IAEA for their continued support for this activity.

CONCLUSIONS

This paper outlines in general terms how nuclear regulation is carried out in principle, with specific examples from Canadian practice. There are many other elements that must be in place in a strong regulatory body (inspectors with specific powers, for example), which have not been covered in this paper. The major theme of the paper’s argument is that care should be taken not to put detailed technical requirements into regulations. They are usually difficult to get right, they are difficult to change, and nuclear safety will not be assured simply by ensuring all the regulations are met. A more flexible approach is needed which is no less strong from a legal point of view, but takes account of the many issues – design, operational management, human factors, etc., which can have a major impact on the safety of the plant throughout its lifetime.

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[1] Nuclear Safety Convention, IAEA, Vienna
ROAD MAP FOR NUCLEAR POWER IN A DEVELOPING COUNTRY

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Abstract

The developing countries planning to go for nuclear power have to understand, that for two distinctly different types of nuclear technology applications i.e. (i) related to nuclear power applications and (ii) all other applications related to agriculture, medicine, industry, etc., the effort required differs greatly. For all applications other than nuclear power, not much effort is needed and this activity can be done to the desired scale, to meet the development goals of the country; even private institutions/hospitals can provide this service on commercial basis by importing isotopes. This in fact, is happening in most of the countries; the government comes into picture for establishing and enforcing Radiation Protection Infrastructure. Absolutely different level of effort and the commitment is required by the government of any country, planning to go for nuclear power. Siting, building, and operating NPPs safely, involves developing and maintaining considerable national infrastructure. The first and foremost requirement is to have a national R&D nuclear centre with a research reactor of at least a few megawatts, around which a number of activities are established. This centre is used as an HRD Centre and a stepping-stone to the nuclear power programme. The next step is to choose a small power reactor and in parallel establish some minimum engineering facilities depending upon the type of power reactor being considered. Some developing countries are considering the introduction of nuclear power by inviting international bids for Build, Own & Operate mode or Build, Own, Operate and Transfer mode. They feel that the vendor will recover the cost stretched over a long period, by selling electricity to the bulk consumers, like state electricity boards or private companies; in the long run this method is not likely to be successful and beneficial to the developing country even if a vendor is able to take up the project in this mode.

1. INTRODUCTION

Nuclear technology applications can be divided in two distinct categories

(i) Non-Power Applications – Applications of radioisotopes in Agriculture, Health, Industry etc.

(ii) Nuclear Power Applications

Applications which use radioisotopes in agriculture, medicine, or industry do not demand a big and elaborate infrastructure end users and for regulation. It is possible to import the required radioisotopes and convert these into the chemical form required by end users or alternately, the sealed sources/chemical compounds required for end users can be imported. This can be done by any state owned or private organisation/company. The regulation is necessary from the considerations of enforcement of radiation protection guidelines and the required radiation protection infrastructure can be established by the government. There is no need to have a research reactor or isotope production facility; this is how most of the countries are benefiting from peaceful radioisotope applications. Nuclear power generation requires a more elaborate infrastructure and a high degree of commitment from government. The paper presents the different stages and the required infrastructure needed to develop nuclear power generation in a developing country.
2. PRELIMINARY STAGES

The task begins by identification of a suitable site for the plant. Site evaluation takes into consideration the population in the vicinity of plant as well suitability from considerations of naturally occurring events like earthquakes, storms, tidal waves etc. In addition there are other requirements like availability of water, heat sink, construction power supply, and availability of sea, rail or road route, for transporting equipment. These tasks require specialists and some surveys for deciding the suitability of the site.

3. CONSTRUCTION STAGES

The major work related to plant construction can be divided in four major stages viz.

(i) Civil construction of the plant in particular foundation and containment
(ii) Fabrication and erection of various plant equipment and piping systems
(iii) Completion of instrumentation and control system
(iv) Commissioning and integration of all the plant system

It is necessary to maintain excellent quality standards during all the stages of plant construction to ensure a trouble free and reliable operation of the plant for the entire plant design life. A number of details are worked out with the vendor keeping in mind the above stages. This will also involve the involvement of local engineering personnel for all stages.

4. INFRASTRUCTURE REQUIREMENTS

Nuclear power generation requires multi-disciplinary expertise. It also requires an adequate infrastructure in many disciplines to realise its full potential. One of the alternatives to introduce nuclear power in a country is by selecting a suitable vendor and plant construction on the basis of BOOT (Build, Own, Operate and Transfer) mode. In this mode, the vendor builds the plant, owns and operates for some time and then transfers to the local body for further operation and maintenance. The vendor continues to have his presence by keeping some experienced operators. He may also accept deferred payments on the sale of electricity during the lifetime of the plant. Alternately the vendor may continue to own and operate the plant and recover all his costs by selling electricity to an electricity supply company. This may appear a more convenient way as requirement for local infrastructure may considerably be reduced. However, it may turn out to be detrimental to the country on some probable considerations:

(i) There may not be involvement of local manpower & infrastructure and thus the ‘learning curve’ may be missing.
(ii) The vendor at a later stage, may not find the arrangement economically attractive and may try to come out of contractual obligations.
(iii) The vendor may not be able to provide support due to a change in political scenario.
(iv) After the transfer of operation & maintenance to the local body, for any ‘trouble shooting’ the engineers from the vendor may have to be summoned at high cost.

Due to the above considerations, it is necessary to have a proper infrastructure for development of the nuclear power. Attempt should be made to have an infrastructure for eventual indigenisation of critical equipment. This requires a nuclear centre for carrying out
necessary research, development and testing; this type of centre is also essential for human resource development.

4.1 Research and Development Centre

Nuclear power requires a sustained technological support for satisfactory exploitation. The support is necessary to find solutions for occasional problems that may develop during the plant operation. The research and development centre can provide this support. To begin with, the centre should have a research reactor of at least a few megawatts and not a small one having a capacity in kilowatt range. A relatively large research reactor size is necessary to have a better understanding of the complexities of the nuclear power plants. Involvement in construction and operation of this research reactor by a developing country can help in gaining an insight into the operating complexities for the reactor plants. All the nuclear activities revolve around this reactor. This centre is also used as an HRD (Human Resource Development) Centre and a stepping-stone to the nuclear power programme. Manpower training can be provided not only for reactor operation and maintenance but also for tasks like:

(i) Reactor physics involving core design and fuel management. This involves development of expertise in the field of estimating reactivity balance, control rod worth, working out refuelling strategies, linking reactor physics calculations with the thermal hydraulics etc.
(ii) Design of process systems and equipment. This involves design of high-pressure high-temperature vessels, heat exchange equipment, piping system etc.
(iii) Design of the reactor control system. This involves working out reactivity control algorithms, design of control system including the requirements of electronics etc. This also involves design of reactivity control mechanisms.
(iv) Understanding the metallurgical aspects of nuclear reactors. This involves studying various construction materials for corrosion behaviour, irradiation behaviour of various materials like low alloy steels, stainless steels, zirconium alloys etc.
(v) Understanding water and steam chemistry.
(vi) Nuclear material handling, balancing & control.

5. ADDITIONAL INFRASTRUCTURE FOR POWER PROGRAMME

The next step is to choose a small power reactor before launching a big nuclear power programme. Choosing a small size reactor reduces the plant complexities and also the financial outlay. Instead of inviting international bids and choosing the cheapest offer, it is better to have bilateral discussions with probable vendors for involving the local manpower & industry with an ultimate aim to absorb technology on a step-by-step basis. Depending upon the type of the chosen power reactor, it is also necessary to establish engineering facilities to provide support for the power reactor programme. The facilities should consist of:

(i) A high-pressure high temperature facility to test plant components like pumps, valves etc.
(ii) A facility to study corrosion behaviour of materials
(iii) A laboratory for material testing including hot cells for testing irradiated materials
(iv) A laboratory for carrying out non-destructive examination of plant components. This should also involve in-service inspection of plant equipment.
In the field of reactor engineering a great emphasis is placed on analytical modelling. This requires test set-ups and personnel trained in fields like:-

(i) Safety studies: This requires instrumented engineering facility to validate analytical models

(ii) Stress analysis: This requires use of analytical tools like FEM (Finite Element Methods) and an experimental stress analysis capability

(iii) Reactor physics: This requires reactor physics codes and an experimental zero-power critical facility

The infrastructure described above does not consider the aspects related to fuel cycle viz. fuel fabrication, fuel reprocessing and waste management. These activities require an additional infrastructure in form of fuel fabrication plants, fuel reprocessing plants and plants for waste treatment. In the beginning of the establishment of the nuclear power programme, it may not be necessary to establish the infrastructure for the fuel cycle. Many countries tend to defer the decision until the programme is well under way. This is mainly due to lack of readily available technology for the fuel cycle. Many a times decisions related to waste disposal are difficult due to lack of proper waste depositories. It may be expedient to establish fuel fabrication facility and defer the decision on the waste management.

CONCLUSION

1. The road map described in the paper indicates the outline of the infrastructure necessary for a sustained development of nuclear power programme in a developing country. Even though to begin with BOOT mode of NPP (Nuclear Power Plant) installation may appear attractive, it may not offer advantage, in a long run. Hence, it is necessary to give adequate attention to building up a comprehensive infrastructure for supporting the programme. This may take a time span of about a decade but the cost is a small percentage of the total cost of the first power reactor.

2. A multidisciplinary R&D base is important in embarking on a nuclear power programme.

3. Industrial infrastructure for nuclear power cannot grow in isolation and it has to be grafted to the country’s existing industrial base.

4. Acquisition and adapting of technology to local conditions involve process of learning and development; while a high degree of indigenisation requires considerable development efforts in the initial instance, long term benefits can be significant in terms of capability of grow on one’s own efforts; spin-off benefits to other sectors of industry can be significant.
FINANCIAL ASPECTS OF DEPLOYING SMRS

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Abstract

KAERI has been developing SMART which is a 330 MWt-sized integral type pressurized water reactor for co-generation, sea-water desalination and district heating as well as electricity generation. The basic design will be finished by March of 2002 with the budget of around 25 million US dollars. The interim assessment of SMART technology has been performed by a group of Korean scientists and engineers. The technical review has shown that the technology is generally very sound, safe and licensable. Its economic evaluation of the one-unit plant with 90 MWe electricity generation along with 40,000 ton/day seawater desalination has been also performed, and the estimated unit construction cost was 2,500 US$/kW, and the electricity generation and seawater desalination costs were 40 US$/MWh and 0.5 US$/ton, respectively, which is considered very competitive against the gas-turbine plant in the world market. It is, however, strongly recommended to construct a 1/5-scale pilot plant (65 MWt) for the purpose of integral demonstration of technology. The required budget for constructing the pilot plant is estimated to be around 200 million US dollars, which will be raised by industries with the commitment of 30% Korean government funding. SMART will be very attractive in the world market by further reducing its construction cost, which could be achieved by employing modularization, 100%-factory fabrication, and repetitive duplication.

1. INTRODUCTION

Since July of 1997, the Government of Korea has been supporting KAERI (Korea Atomic Energy Research Institute) to develop the conceptual and basic designs of a small- & medium-sized reactor called SMART (System-Integrated Modular Advanced Reactor). SMART is a 330 MWt-sized integral type multi-purpose pressurized water reactor which will be used for co-generation, sea-water desalination and district heating as well as electricity generation. The conceptual design was already completed in March of 1999, and the basic design is on-going and will be finished by March of 2002. The total budget allocated to the design project (from July of 1997 through March of 2002) is around 25 million US dollars.

For a successful completion of the project, the “Steering Committee for SMART Technology Development” was formed under the Government, which is composed of 15 members in various expertise from governments, industries, research organizations and academic bodies. The Committee has been continuously monitoring the progress of its design, and providing the project team in KAERI with appropriate timely guidance. The Committee will continue to steer the project team until its completion of the development of SMART technology.

In August of 2000, the Steering Committee decided to perform a systematic interim assessment of the status of the technology development, and to judge whether the Government should continue to support the project beyond the design phase after the completion of basic design in March of 2002. The Committee appointed the KNS (Korean Nuclear Society) to perform the overall assessment. And the KNS organized a team of experts (33 engineers and scientists) to review the technical soundness of SMART technology, and subcontracted the part of its economic evaluation to KOPEC (Korea Power Engineering Company). Based upon the results of the technical review and economic evaluation, KNS has presented appropriate recommendations to the Steering Committee in April of 2001. This paper summarizes the results of this evaluation and subsequent conclusion.
2. TECHNICAL REVIEW

The main items of technical review in the SMART design proposed at present were: 1) design objectives and principles; 2) current technical safety and licensing issues; and 3) design verification tests and experiments. The questions that arose during the technical review process are as follows: Is the technology of SMART safe while keeping its cost effectiveness? Is the integral reactor manufacturable, erectable, and maintainable? Is it licensable under current licensing regulations and criteria? What should be done for technology verification in the sense of proven technology? What design improvements are needed or desirable in order to make SMART technology more attractive to customers?

TABLE I. VERIFICATION TESTS IDENTIFIED FOR SMART TECHNOLOGY DEVELOPMENT

<table>
<thead>
<tr>
<th>1.4. Area</th>
<th>1.3. Tests</th>
<th>1.1. Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nuclear</td>
<td>Critical Heat Flux</td>
<td>On-going</td>
</tr>
<tr>
<td>Mechanical</td>
<td>Fluid and Mass Balance and Effects</td>
<td></td>
</tr>
<tr>
<td>Safety Analysis</td>
<td>Heat Transfer Characteristics of Helical SG Tubes</td>
<td></td>
</tr>
<tr>
<td>Safety Injection Performance</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Safety Analysis</td>
<td>Pressurizing Performance</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Non-condensible Gas Behavior</td>
<td>On-going</td>
</tr>
<tr>
<td>Safety Analysis</td>
<td>PRHR Performance</td>
<td></td>
</tr>
<tr>
<td>Safety Analysis</td>
<td>Flow Distribution</td>
<td>Test facility under design</td>
</tr>
<tr>
<td>MMIS</td>
<td>Reactor Protection System</td>
<td></td>
</tr>
<tr>
<td>Main Equipment</td>
<td>Control Element Drive Mechanism</td>
<td></td>
</tr>
<tr>
<td>Water Chemistry</td>
<td>Material Corrosion Resistance</td>
<td>On-going</td>
</tr>
</tbody>
</table>

After going through a long and thorough technical review process, it was concluded that the technology of SMART was generally very sound, safe and licensable. However, due to the characteristics of integral reactor in itself, there could exist some unresolved licensing issues under present licensing criteria such as the lack of diverse reactivity control and in-service inspection capabilities, which requires the close scrutiny. New licensing schemes should be also investigated in a parallel manner with its technology development of SMART by the regulatory side in order to embrace the innovative features proposed and adopted in the SMART technology.
For the purpose of verifying SMART technology, various separate tests and experiments were identified, which are briefly summarized in Table 1. It has been strongly recommended to carry on the tests and experiments proposed in Table 1, and also to construct a 1/5-scale pilot plant (65 MWt) for the purpose of integral demonstration of the technology, which was considered the best approach to get public acceptance of this new concept of technology.

3. ECONOMIC ASSESSMENT

Under its economic evaluation, the unit construction cost (US$/kW) was estimated. And the unit electricity generation (US$/MWh) and seawater desalination (US$/ton) costs were computed. Table 2 summarizes the preliminary results of economic evaluation performed by KOPEC. The base case used for this economic evaluation was the one-unit plant with 90 MWe electricity generation along with 40,000 ton/day seawater desalination, which is considered as the right size of electricity and fresh water supplies to a typical Korean city of population of 100,000. For the base case, the estimated unit construction was around 2,500 US$/kW, and the electricity generation and seawater desalination costs were around 40 US$/MWh, and 0.5 US$/ton, respectively. In addition, the costs for the 2-unit plant and 4-unit plant were also estimated and presented in Table 2. As can be noticed in Table 2, the unit construction cost for a 4-unit plant is reduced to 2,000 US$/kW. Remember that these are the numbers not based upon guesstimates but based upon first-cut quotations actually received from equipment suppliers.

TABLE II. ECONOMICS OF SMART

<table>
<thead>
<tr>
<th>Categories</th>
<th>construction cost (US$/kW)</th>
<th>electricity generation (US$/MWh)</th>
<th>seawater desalination (US$/ton)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1-Unit Plant (first unit)*</td>
<td>2,510</td>
<td>38.45</td>
<td>0.537</td>
</tr>
<tr>
<td>2-Unit Plant</td>
<td>2,409</td>
<td>37.22</td>
<td>0.509</td>
</tr>
<tr>
<td>4-Unit Plant</td>
<td>2,055</td>
<td>33.03</td>
<td>0.475</td>
</tr>
</tbody>
</table>

* Base Case: 90 MWe electricity generation and 40,000 ton/day seawater desalination with 330 MWt power.

Since the electricity generation cost of gas turbine is 47 US$/MWh (1999 Data) in Korea, with this economics, SMART still gives certain economic advantages over gas turbine for peak-time electricity generation when its operation is combined with existing large-scale nuclear power plants. The world average seawater desalination cost is around 1 US$/ton, and SMART also gives an economic incentive in seawater desalination.

4. SUMMARY OF RECOMMENDATIONS FOR SMART DEVELOPMENT

There arise some other questions during the review process: Who takes or should take initiatives in developing, demonstrating and commercializing the technology? What are the selling points of SMART when it is deployed in Korea as well as overseas? Who is going to finance the construction of the pilot plant? How do we find the site for constructing the pilot plant? How do we get public and political acceptance of this new concept?
In order to successfully complete the Project of SMART development, it is recommended that the Project should progress in three separate chronological phases (see Figure 1): Phase I is the technology development; Phase II is the technology verification; and Phase III is the technology commercialization. Considering the current on-going project of developing the basic design led by KAERI as Phase I, Phase II is set for technology verification, in which the construction of a pilot plant will be specifically included in order to conduct integral and comprehensive tests for overall performance. Starting from Phase II, industries should participate in the Project and the developed technology will be systematically transferred from KAERI to industry. After Phase II is completed, industry should eventually take over the SMART technology for commercialization, which we call Phase III. In this stage, one of these industries will eventually become the vendor for 330 MWt as well as 65 MWt (a spin-off product of the pilot plant) integral-type reactors as the final products for commercialization.
Figure 1 also shows the proposed milestone schedule of SMART development. Phase II will start in August of 2001 and be completed by 2007. The budget for Phase II is estimated to be around 200 million US dollars. The recommendation is that industry should finance more than 70% of the required budget and the remaining 30% be supported by the Government using the Government Nuclear R&D Fund.

It is also recommended to form a project team which is composed of engineers and scientists from KAERI as well as industry, and 100% dedicated to this project. The Project Team will take the whole responsibility of carrying out the construction of the pilot plant and conducting the required tests for technology verification. Including the budget of 25 million US dollars which has been already allocated for Phase I, the total investment for SMART technology development through Phase II will become around 230 million US dollars all together before its actual commercialization. Figure 2 shows the work flow during Phase II operation of this Project.

RESULTS AND CONCLUSIONS

It is the general opinion that, in deploying small and medium sized reactors, the successful demonstration via constructing the lead plant is considered most important. In this aspect, it is very prudent that Korea has decided to construct the pilot plant. And the second important
thing in deploying small and medium sized reactors, is the assurance of success in the future nuclear market so that industry should get interested in investment. In this aspect, Doosan Heavy Industries, Inc. which is the one and only main equipment manufacturer in Korea, has already decided to participate in this SMART development project and has shown the willingness of investing more than 70% of the cost needed for Phase II. The Company has decided to take over the responsibility of constructing the pilot plant, and willingly to be the vendor of SMART in its commercialization phase.

It is very encouraging to learn that the Company makes this kind of aggressive decision of big investment even though the results of our review study revealed rather pessimistic economic results. The Company has expressed its strong confidence in making the economics of SMART very attractive in the world market by drastically reducing the construction cost, which could be achieved by employing modularization, 100%-factory fabrication, and repetitive duplication. Moreover, the management of the Company has sensed the big future world-wide market of small and medium sized reactors.

In conclusion, Korea believes that small and medium sized reactors are bound to be successful just like small PCs (personal computers) have been successful along with large main-frame in computer business.
CLOSING REMARKS
P.E. Juhn
IAEA

The Seminar was timely and well attended with more than 200 participants representing more than 35 countries coming from all corners of the world.

I am sure you will all agree with me that, in the last three days, we had very fruitful discussions on a variety of aspects concerning the status and prospects of small and medium sized reactors including the incentives, challenges and possible solutions for their deployment.

When we were considering the location of the Seminar, Sayed Abdel Hamid was enthusiastic about having it in Cairo. Indeed, he took on a big load as the participation in the Seminar turned out to be more than double than originally expected. On behalf of the IAEA, I thank His Excellency, the Minister of Electricity and Energy in Egypt, Ali F. El-Saeidi and the Egyptian Government for supporting the initiative wholeheartedly and making it a reality. The venue and facilities were very good and the hospitality provided to the international gathering was exceptional. Despite the tight schedule of the Seminar, the hosts managed to give us a good glimpse of the most impressive sights and sounds, of their ancient culture.

Another major factor for the successful conclusion of the Seminar is the co-sponsorship of the event by the OECD/NEA and the World Nuclear Association. The significance of the topical nature and timing of the Seminar can be gauged by the fact that the three sponsoring organizations, the OECD/NEA, the World Nuclear Association and the IAEA were all represented by the respective Directors General. I thank Mr. Louis Echavarri, Mr. John Ritch III and Mr. Mohammed ElBaradei for their full support and active participation in the Seminar. Unfortunately, we missed the Prime Minister of Egypt, His Excellency Mr. Atef Eibed as he was called away by important State duties, to go to Indonesia, at the last minute.

In just three and a half days nearly hundred technical papers and five keynote papers were presented and three panel discussions held. Most heartening was the fact that the auditorium was nearly full throughout this period. Even during the difficult post-lunch sessions as well as late evening sessions, the participation remained vigorous and at a high level. That is a real indicator of the tremendous interest in the Seminar and its consequent success in highlighting the issues that need to be considered further.

I will not take up your time by going into summaries of each session but it is worthwhile recapturing some of the main messages that need to be taken into account in formulating our respective future work programme.

- Population growth and the universal desire for a decent economic standard of living are the main driving forces of the demand for energy and electricity.
- The world population will mostly grow in developing countries, which in turn means that the world energy demand will mainly occur in developing countries.
- All the Member States and the international organizations will have to work together to identify acceptable solutions and facilitate their implementation.
• Most developing countries are not in a position to start with large nuclear power reactors for several reasons and small and medium reactors (SMRs) from a few MW(e) to about 700 MW(e) are essential for nuclear power to play a role in the developing world.

• SMRs and large reactors are by no means mutually exclusive. Like automobiles and aircraft, there will be a need for all types and sizes of reactors for different purposes. In this context use of nuclear power for non-electric applications is essential for the development of the economies of developing countries.

• Economics and public acceptance are the two most vital factors for future growth of nuclear power in the increasingly competitive environment.

• Public acceptance in turn depends largely on continued maintenance of the highest emphasis on nuclear safety; acceptable solutions to spent fuel and radioactive waste disposition and a globally accepted non-proliferation regime.

• About 40 concepts of SMRs in LWR, HWR, HTGR, LMR were presented in this Seminar and all of these could make a significant contribution to the areas of current concern to the general public and the investors. There are many new ideas in design simplification and passive safety that have floated around and were discussed by many participants.

• The time frame for the availability of SMRs is very important as most developing countries cannot wait for another two to three decades to increase their installed electricity generation capacities. If nuclear plants were not feasible, reliance on fossil fuels would continue to increase with its adverse impact on the environment.

• Lack of financial resources and infrastructure including qualified manpower would be the main constraints to the introduction of nuclear power in developing countries. Some concepts have been suggested for improving these areas through international/regional cooperation.

Those are some of the important messages that came through loud and clear in the Seminar. Some developing countries would like to gradually increase the level of national participation in the nuclear power programme implementation. I would like to state that this aspect is already being taken into account in the INPRO initiative of the IAEA, which has emphasized to develop End-User Requirements and Nuclear Energy Development Criteria which is aimed to be implemented with other initiatives such as GEN IV International Forum in a complementary manner that I described on the first day. We are treating the infrastructure requirements not as a static factor but as one that changes with time and the level of economic development. We will take back with us all the messages and suggestions from this Seminar and adjust our current programme activities of the IAEA to the extent feasible and more so in the future activities, to respond to these needs of Member States. I would like to request you all to treat this event just as a beginning of our association and continue to network with each other to collectively bring about the changes that we all desire to see in the world with respect to increasing nuclear power utilization.

Last but not least, I would like to thank the Secretariat of the Seminar consisting of D. Majumdar, M. Rao, M. Methnani, K. Morrison, T. Niedermayr, S. Abdel Hamid and M. Megahed for the intensive efforts they put in to make this Seminar a success. Thanks are also due to Messrs. Kupitz, Gasparini and Choi, my colleagues from the IAEA as well as to the Seminar Organizing Committee in Egypt for the excellent support they provided in a variety of ways. A special mention should be made of the quietly efficient role played by L. Wedekind of the IAEA in disseminating news about this seminar to the rest of the world.
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