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Performance of operating and advanced light water reactor designs

Proceedings of a Technical Committee meeting held in Munich, Germany, 23–25 October 2000

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FOREWORD

Nuclear power can provide security of energy supply, stable energy costs, and can contribute to greenhouse gas reduction. To fully realize these benefits, a continued and strong focus must be maintained on means for assuring the economic competitiveness of nuclear power relative to alternatives. The IAEA's nuclear power programme includes information exchange activities to achieve improved reliability and cost effectiveness of nuclear power plants by promoting advanced engineering and technology.

Over the past several years, considerable improvements have been achieved in nuclear plant performance. Worldwide, the average energy availability factor has increased from 66 per cent in 1980 to 81 per cent in 1999, with some utilities achieving significantly higher values. This is being achieved through integrated programmes including personnel training and quality assurance, improvements in plant system and component design and plant operation, by various means to reduce outage duration for maintenance and refuelling and other scheduled shutdowns, and by reducing the number of forced outages.

Application of technical means for achieving high performance of nuclear power plants is an important element for assuring their economic competitiveness. For the current plants, proper management includes development and application of better technologies for inspection, maintenance and repair. For future plants, the opportunity exists during the design phase to incorporate design features and technologies for achieving high performance.

This IAEA Technical Committee meeting (TCM) was hosted by E.ON Energie, Munich, Germany from 23–25 October 2000. The TCM provided a forum for information exchange on design features and technologies incorporated into LWR plants commissioned within the last 15–20 years, and into evolutionary LWR designs still under development, for achieving performance improvements with due regard to stringent safety requirements and objectives. It also addressed on-going technology development expected to achieve further improvements and/or significant cost reductions. The TCM was organized by the IAEA's Division of Nuclear Power. The IAEA officers responsible for this publication were J. Cleveland and T. Mazour.

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SUMMARY

The Technical Committee Meeting (TCM) on Performance of Operating and Advanced Light Water Reactor Designs was convened within the framework of the IAEA's International Working Group on Advanced Technologies for Light Water Reactors. Topics addressed within the frame of this International Working Group focus on technology developments for improving economic competitiveness of LWRs while meeting safety objectives.

The TCM was attended by 32 participants from 14 Member States: Argentina (1), Bulgaria (1), Czech Rep.(2), Finland (1), France (3), Germany (9), Hungary (2), Japan (1), Republic of Korea (2), Mexico (1), Russian Federation (1), Slovakia (1), Spain (1), Ukraine (2)], the European Commission (2) and the IAEA (2).

The meeting was chaired by D. Brosche, Director of E.ON Energie AG, and Manager of the ISAR-1 and ISAR-2 Nuclear Power Plants.

A total of 19 technical papers were presented in the following areas:

- Design features and technical means for improving current LWRs;
- Development of new LWR designs and technologies with a focus on performance and economic viability.

Key results concerning performance of current plants include the following:

The European Commission (EC) is funding an extensive project in plant life management at a funding level of EURO 9.2 million within the 5th framework programme (1999–2002). This work is being co-ordinated with the IAEA through EC participation in the IAEA's International Working Group on Plant Life Management. Technical areas addressed include the following:

- Integrity of equipment and structures
  - Irradiation embrittlement;
  - Optimization of operational conditions focusing on corrosion issues;
  - Prediction of structural safety margins;
  - Optimization of operational conditions focusing on dynamic fluid-structure interactions;
- On-line monitoring, inspection and maintenance;
- Organization and management of safety.

Extensive modernization programmes have been carried out at older plants (e.g. ISAR-1 and Olkiluoto-1 and -2) that have resulted in achieving both a high level of safety and competitive economic performance. ISAR-1 has undergone extensive improvements since it was connected to the grid in 1977. In 1999 ISAR-1 achieved the best performance record since commissioning with an availability factor of 99.7% and a capacity factor of 98.7%. The Olkiluoto units were commissioned in 1979 and 1982 respectively. Advanced features were incorporated in the original design, and numerous improvements (e.g. exchange of power conversion equipment to achieve power up-rating) have been made. The units have operated with combined capacity factors of 90% and above since 1985. The units typically achieve annual outage lengths of between 10 and 20 days.

Many design improvements have been and are being carried out at Kozloduy Units 1,2,3 and 4 to meet modern safety requirements. The main areas include:

- Primary circuit integrity;
- Severe accident behavior;
- Leak before break application;
- Confinement leak tightness; and
- Seismic design improvement.

Design improvements have been made in Ukrainian WWERs to meet safety requirements. Examples of design changes involve the emergency core cooling system, and improvement of the reactor control and protection system. Improved fuel management has resulted in reductions of fuel cycle costs in the range of 10 per cent.

The Paks Nuclear Power Plant (NPP) experience with respect to development of symptom-based emergency operating procedures (EOPs) was reported. These symptom based EOPs improve the performance of the plant in the event of an incident or accident because they provide a framework within which all critical safety functions can be monitored and appropriate actions taken. They provide a complement to event-based procedures because it isn't possible to anticipate all plant events, particularly combinations of individual events. The Paks NPP staff have integrated the existing event-based procedures with symptom-based EOPs to provide a comprehensive framework to appropriately respond to all abnormal and emergency conditions.

Optimized preventive maintenance programmes can contribute to ensuring both safe and competitive NPPs. Reliability centered maintenance provides an optimization of preventive maintenance. The principles underlying reliability centred maintenance include preventing failures where the repercussions for the plant could be serious in terms of safety or economics. The methodology involves evaluation of the functional consequences of failure, analyses of experience feedback, and optimization of preventive maintenance tasks. Electricité de France (EdF) has implemented preventive maintenance programmes based on a reliability centred maintenance methodology at EdF nuclear plants, and results have shown benefits in safety, performance and cost.

The nuclear plant service industry is implementing improved service approaches for maintenance. An example is the integrated valve maintenance approach developed and offered by Siemens. The goal is to optimize the overall sequence including advanced planning and conduct of the maintenance activities. The approach has been carried out at several plants in Europe.

Operational experience of current plants provides good guidance for developers of future plants. As an example, experience of the Laguna Verde BWR in introducing flexibilities in core flow operation at full power to achieve improved fuel utilitization provide useful guidance in regard to design features and operating procedures to address turbulence and flow induced vibrations. Power up-rating at this plant through the use of improved analytical methods and calculational tools has been accomplished without decreasing any safety margin.

In summary, the papers presented concerning improvements in the performance and design features of current plants reinforced the valuable contribution that operating experience provides, both for current plants and in the design of new plants, particularly evolutionary plants. Operational experience has demonstrated that there is a potential to increase the competitiveness of the existing nuclear power plants including relatively old ones. Further improvement of nuclear power competitiveness is possible by the extension of NPP lifetimes. This work is now being performed in some countries; for example, the possibility to increase Novovoronezh-3&4 and then Kola-1&2 lifetime by 10–15 years is now being studied in Russia. Another effective way to reduce the nuclear electricity cost is to increase the rated power of the plant. ISAR-2 in Germany and the Olkiluoto and Loviisa plants in Finland provide good examples of such activities.

Key results concerning new plant designs include the following:

Evolutionary designs for new plants have lower commercial risk than innovative designs because they build directly on performance experience of existing plants. However, in some countries, these designs are having difficulty in achieving competitiveness with alternative electricity generating sources (e.g. combined cycle gas turbines). Innovative designs can potentially be less expensive while also achieving very high safety levels. However, the possible need to have a prototype as part of the development programme will be a large financial impediment. Each generation of nuclear power plants has sought higher levels of nuclear safety, with an associated cost. In the future, in order to remain competitive, it will be necessary to first identify clear safety goals and then to find cost effective ways to achieve these goals.

With regard to future plants, some evolutionary water-cooled reactors are starting operation or are under construction, some designs have been certified by regulatory authorities, and some are under development. Examples of evolutionary designs include the 1360 MW(e) Advanced Boiling Water Reactor (ABWR) which is currently operating in Japan, the 1545 MW(e) European Pressurized Water Reactor (EPR) which is under development by Nuclear Power International, the boiling water reactor SWR-1000 of Siemens, the 1450 MW(e) Korean Next Generation Reactor (KNGR) which is being developed by the Korea Electric Power Corporation and the Korean nuclear industry on the basis of operating experience with the Korean Standard Nuclear Plant (KSNP), the WWER-1000, V-392 design under development in the Russian Federation, and the AP-600 which has been developed in the United States by Westinghouse and certified by the US Nuclear Regulatory Commission.

The present situation encourages a strong focus on technologies for improving economics. Evolutionary designs have incorporated the large experience base of operating plants plus many technological developments to enhance safety — but these design features have often had increased capital costs. Designers of evolutionary plants have of course had reductions of both capital and operating costs as goals, but in some cases the target costs used in the design activities have been considerably above costs that would be competitive in current privatized, de-regulated markets. Further, design, development and certification of some evolutionary designs has taken 12 years and more — maybe too long in changing market conditions.

Therefore, the nuclear industry is challenged to develop advanced reactors with

- considerably lower capital costs and shorter construction times;
- sizes (including small and medium sizes with load following capability) appropriate to grid capacity and owner investment capability;
- high levels of standardization and modularisation.

Designers of innovative plants may be able to take even more advantage of cost reduction approaches. Innovative designs are being developed in several countries for all reactor lines (gas, water and liquid metal cooled). Goals include low capital cost, short construction time, enhanced safety and proliferation resistant features. Several of the innovative designs are small and medium size reactors (SMRs), which are a better fit to modest demand growth and smaller electricity grids (e.g. in developing countries), are easier to finance, can be simpler and often employ passive safety systems, and are a good fit for several non-electric applications.

Innovative water-cooled reactors include integral designs, such as the CAREM design which has been under development in Argentina since 1984, and high performance designs operating thermodynamically in the supercritical regime (above 22 MPa and 374 C) to achieve high thermal efficiency for reduced capital cost. In addition to the CAREM development activities, development of integral designs is underway in the Republic of Korea for the SMART design, in Russia for the VPBER-600 design, and in the USA. for the IRIS Generation IV design. High performance designs operating in the supercritical regime are being examined co-operatively by design and research organizations in Europe and Japan. Furthermore, Atomic Energy of Canada, Ltd (AECL) has selected such systems as their innovative approach for heavy water moderated reactors.

In the USA, the US Department of Energy initiated the Nuclear Energy Research Initiative (NERI) in 1999 within its Office of Nuclear Energy, Science and Technology, as a result of recommendations of President's Committee of Advisors on Science and Technology (PCAST) that the following should be addressed:

- Technologies for proliferation resistant reactors and fuel cycles;
- New designs with high efficiency, reduced cost and enhanced safety;
- Designs with lower output; and
- New technologies for
    - On-site and surface storage of waste; and
    - Permanent disposal of waste.

PCAST also suggested that the task areas should be recommended by scientific investigators as a less prescriptive means of determining the tasks to be conducted. Awards are granted following solicitation, merit review and a peer review and selection process. Collaboration is encouraged among national laboratories, universities, industry and international R&D organizations.

Funding for 1999 and 2000 for NERI was US$ 19 million and 24 million respectively, with an increase anticipated for 2001. Forty-six projects were initiated in 1999 and ten more were started in 2000. Some examples of ongoing activities within NERI that should contribute to improved economics of future LWRs include:

- A risk informed assessment of regulatory and design requirements;
- A study of "smart" equipment and systems to improve reliability and safety;
- A study of a larger version of the AP-600 design to benefit from economies-of-scale while retaining passive system technology;
- A study to develop means for forewarning of failure in critical equipment.

NERI serves as "birthing place" for future programmes, including the Generation IV programme. This new programme would focus on advanced reactor designs beyond the advanced LWRs that were developed in the USA during the last half of the 1980s and 1990s. Goals of Generation IV reactors would include the characteristics:

- Highly economical;
- Enhanced safety;
- Minimized waste; and
- Proliferation resistant.

The following summarizes specific advances for future reactors that were described in detail at the TCM.

The CAREM reactor under development in Argentina is a 100 MW(th) (about 27 MW(e)) design based on natural circulation. It has an integrated primary circuit comprising the core, steam generators, control rods with their drive mechanisms and the entire primary coolant. Several experiments examining core neutronics and thermal hydraulics have been conducted in test facilities. The construction of a prototype is planned.

In Japan, the first two ABWRs in the world, Kashiwazaki-Kariwa Units 6 and 7, have been in operation since 1996 and 1997 respectively. ABWR plants are under construction at Hamaoka Unit No. 5 and Shika Unit No. 2, and under licensing at Ohma Unit 1. Another eight ABWR plants are in the planning stage in Japan, and construction has started on two ABWR plants in Taiwan, China. Expectations are that future ABWRs will achieve a 30% cost reduction relative to the costs of Kashiwazaki-Kariwa Units 6 and 7. The means for achieving this cost reduction include standardization, design changes and improvement of project management, with all areas building on the experience of the ABWRs currently in operation.

The next version of the ABWR, the ABWR-II, is under development in Japan with the objective of reduced cost. The electric power is increased to 1700 MW(e), with commissioning of the first ABWR-II foreseen in the late 2010s. Design changes include larger fuel assemblies, passive systems, and intensified precautions against severe accidents.

An advanced version of the WWER-1000 reactor, the V-392 design, is under development in the Russian Federation. The V-392 design is based on the presently operating V-320 plants. According to the programme of the Government of the Russian Federation from 1998, the WWER-1000, V-392 design has been selected as the lead design for new LWRs. A pilot plant of the V-392 design is to be built at Novovoronezh. Modifications relative to the V-320 design include incorporation of passive features into the safety systems. A preliminary assessment of the V-392 design against the European Utility Requirements (EUR) has been conducted by EdF indicating good conformance with the EUR. The electricity production in Russia is estimated to grow in the coming years (perhaps 30 per cent by 2010), resulting in good chances for additional construction of new nuclear capacity.

In the Republic of Korea, the project to develop the KNGR was started in 1992. The design builds on the experience of the KSNPs that are operating and under construction. The conceptual design of the KNGR was completed in 1994, and the basic design was completed in 1999. Design certification by the regulatory authority is expected by the end of 2001. Recent development has focused on improvement of availability and reducing costs. Performance and economic goals include 90 per cent availability and a 20 per cent cost advantage over coal fired electricity generation plants. The nuclear steam supply system is designed to operate at 4000 MWth to produce 1450 MW(e). A recent optimization study addressed several items including safety system optimization, fuel and core design for optimization of thermal margins, and optimization of the containment design and measures against severe accidents. Resulting design improvements included the higher core power mentioned above, increased fuel enrichment, adoption of a single containment with the approach of in-vessel retention with an active system for cavity flooding for the hypothetical core melt accident, rather than a double containment, and elimination of the passive secondary condensing system.

The basic design of the EPR was completed in 1998, followed by a design optimization phase. An increase in power was considered, but a decision was made by EdF to keep the original 1545 MW(e) to avoid risks related to public acceptance. Safety improvements have been introduced to improve both severe accident prevention and mitigation. Design features for improved mitigation include a containment heat removal system for severe accidents with a dedicated cooling system, and a dedicated primary circuit depressurization system. To limit the cost impact, the design of these systems was based on best estimate methods for functional requirements and equipment qualification. Examinations in several areas led to a ten per cent reduction of cost during the design optimization phase. The improvements resulted from, for example:

- Updating of licensing rules dealing with over-pressure protection; and
- Achieving mechanical classification of piping systems that depends on the contamination level of the contained fluid and on the isolation devices designed into the system.

For the EPR, reconsideration of the requirements placed on suppliers of nuclear equipment to bring in recent technological advances is currently underway. First results show that optimization of the specifications will result in further savings. Further improvements in performance and safety have been accomplished by specific steps to improve un-availability based on operating experience of current plants.

While a site has been identified, there is no fixed timetable for construction of the first EPR.

Following the presentation of the technical papers, participants were provided with a tour of the ISAR-1 and ISAR-2 nuclear plants, and further technical discussions of the improvements that have been made at ISAR-1.

DESIGN FEATURES AND TECHNICAL MEANS FOR IMPROVING PERFORMANCE, ECONOMICS AND SAFETY

(Session I-a)

Chairperson

M. MARTIN-ONRAET
France

FP-4 AND FP-5 EURATOM RESEARCH ACTIVITIES IN THE FIELD OF PLANT LIFE MANAGEMENT

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Abstract

In this paper an overview is given of the European Union (EU) Euratom research conducted through shared cost and concerted actions in the field of plant life management. After a general introduction on the organisation of the research framework programmes the achievements of the 4th framework programme (FP-4/1994-1998) and the activities under the 5th framework programme (FP-5/1999-2002) in the field of plant life management are presented and discussed in detail. Besides technological safety requirements, socio-economic aspects are becoming increasingly important due to the level of public and political acceptance and to the economic pressure of deregulated electricity markets. It is shown that research conducted in the Euratom framework may contribute to meet these requirements, thereby maintaining nuclear power as a competitive and sustainable option for the energy policy of the European Union.

1. GENERAL INTRODUCTION

The European Commission's activities in the field of nuclear energy are governed primarily by the Treaty establishing the European Atomic Energy Community (Euratom) and by a body of laws and rules based on the same Treaty. They are extremely wide-ranging, with operational responsibility shared between several Directorates-General (DG), depending on the nature, type and scale of the tasks. Chapter 1 of the Euratom Treaty contains the articles on promotion of nuclear research which provide the legal basis for the nuclear research activities proposed and implemented by the European Commission's Directorate-General Research. DG Research's nuclear research and development tasks for the Community are geared primarily to three objectives:

- to develop the EU's research and technological development strategies;
- to implement these strategies by means of the four-year Euratom Framework Programme allowing scientific cooperation between different European organisations;
- to promote and disseminate the knowledge acquired by stimulating scientific debate at European level.

Consequently, DG Research is responsible for attaining these objectives, with the support of the other Directorates-General responsible for various areas of nuclear research, such as DGs Environment, Transport & Energy and the Joint Research Centre (JRC). Since the ratification of the Treaty on European Union, all the Community's research, technological development and demonstration activities have been brought under the twin four-year Framework Programmes: the Euratom Framework Programme (Euratom FP) on nuclear issues and the European Community Framework Programme for all other branches of science.

The 4th Euratom FP consisted basically of two specific programmes, one on controlled thermonuclear fusion, the other on nuclear fission safety. The EURO 170.5 million for the indirect actions under the nuclear fission safety programme was spent mainly to finance research projects, but also on various accompanying measures and training schemes. Under

FP-4 a total of 229 shared-cost and concerted action projects were supported in total by the European Commission between the following five research areas:

- Exploring innovative approaches
- Reactor safety
- Radioactive waste management and disposal and decommissioning
- Radiological impact on man and the environment
- Mastering events of the past (related to the Chernobyl accident).

The activities related to plant life management were conducted in the framework of the research area of reactor safety and will be treated more in detail further.

The 5th Euratom FP (1999 to 2002) was approved at the end of 1998 and is now in its second year. It has a total budget of EURO 1260 million. The principal strategic goal of the 5th Euratom FP is to help exploit the full potential of nuclear energy, both fusion and fission, by making current technologies even safer and more economical and by exploring promising new concepts. To achieve this objective, it has been given the following structure: 2 key actions on controlled nuclear fusion and nuclear fission, generic R&D activities on radiological sciences, support for scientific structure and the direct actions executed by the Joint Research Centre. The 5th Euratom FP is implemented by means of indirect research and training activities, including shared-cost actions (the main mechanism for implementing the programme), training fellowships, support for training networks, concerted actions and accompanying measures.

The main objectives of the key action on nuclear fission (total budget of EURO 191 million) are to enhance the safety of Europe's nuclear installations and improve the competitiveness of Europe's industry. Within these broader objectives, the more detailed aims are: to protect workers and the public from radiation and ensure safe and effective management and final disposal of radioactive waste, to explore more innovative concepts that are sustainable and have potential longer term economic, safety, health and environmental benefits and to contribute towards maintaining a high level of expertise and competence on nuclear technology and safety. It covers four principal areas of research:

- Operational safety of existing installations
- Safety of the fuel cycle
- Safety and efficiency of future systems
- Radiation protection.

The research related to plant life extension and management is conducted within the framework of the research area "Operational safety of existing installations" and will be dealt with in more detail further in this article.

The research under the key action of nuclear fission is carried out principally by laboratories and research bodies, electrical or engineering companies, the nuclear industry, nuclear regulatory authorities, small and medium-sized businesses and other public or private undertakings which can contribute their experience, possibly even in fields which are not exclusively nuclear, to attaining the objectives of the programme. On the other side, the principal end-users of the results are nuclear power stations, national regulatory authorities and industry in general.

2. BACKGROUND TO EU SPONSORED RESEARCH IN THE FIELD OF AGEING AND PLANT LIFE MANAGEMENT

Looking to the energy market in The EU one can expect a relatively stable number of power plants operated in the foreseeable future. The industry in most of the countries has changed already from a construction-oriented profile towards a profile, which serves the needs of utilities for maintenance and technical support. Most of the nuclear power plant units are approaching or are already in their second half of design lifetime. The main goals for the next decade will hence be [3]:

- To maintain excellent plant reliability and safety in a changing electricity market
- To manage uncertainties with limited resources
- To gain public confidence

All end-users believe that research is still needed to further increase the safety and performances of these power plants, in line with the steadily growing pressure of market and regulatory forces as well as of the public opinion. Looking at the future of these plants, there is a strong drive to extend their lifetime where this can be achieved safely, bearing in mind that the lifetime of a nuclear plant is definitely limited by the ageing of non-replaceable components like the nuclear reactor pressure vessel (RPV). As a consequence, the understanding of ageing is becoming of increasing importance (e.g. irradiation embrittlement, other changes in mechanical and microstructural properties, etc.) to the nuclear industry. More importantly, the optimisation of the operational conditions of aged reactors and the development of appropriate prediction tools for evaluating the safety margins are therefore becoming key issues for those in charge of safety and performance. However, as with all engineering constructions, having an inherent capacity to cause hazards to people and environment, the safety of nuclear installations does not rely only on technical precautions (T) but also on their interaction with the attitude (M) of men, and organisational measures (O). The so-called MTO factor has become world-wide the basis of the reactor safety culture.

In the "Operational Safety" research activities of the Community, up to now and especially under FP-4 the emphasis was put on the T component, i.e. the identification and solution of technological problems. Traditionally the technological problems of nuclear reactor safety are related to the 3 basic safety functions, namely: controlling the power, cooling the fuel and confining the radioactive material. Their solution lies in the standard 3-levels defence-in-depth approach against accidental radioactivity releases, i.e.:

- prevention of abnormal operation and failures
- control of abnormal operation and detection of failures
- control of accidents within the design basis

Traditionally associated with this approach are the following concepts:

- the multiple barrier design for the confinement of radioactive material;
- the protection and safeguard systems to ensure the integrity of the barriers;
- the regulatory procedures (for example, the safety analysis reports) to ensure the health and safety of the plant workers and of the population.

In FP-5 more attention is devoted to the M and O components, including the identification and solution of practical aspects of operational safety (in particular, man-machine interface and organisational factors) in close relation with the "changing world". This implies also that issues like cost/benefit optimisation, plant simplification whilst maintaining the same level of safety and public acceptance in general need to be considered.

3. OBJECTIVES OF FP-4 AND FP-5 SPONSORED EU RESEARCH IN THE FIELD OF AGEING AND PLANT LIFE MANAGEMENT

The principal objectives under FP-4 of the research conducted in the field of reactor safety were to acquire a knowledge of the main phenomena occurring in the event of severe accidents, to reach a consensus on how to deal with them in accident analyses, to establish means of mitigating and managing accidents and to generate experimental results for designing and verifying computer programs. The fields of research are subdivided into the following sub-areas: in-vessel core degradation and coolability, ex-vessel corium behaviour and coolability, source term, containment performance and energetic containment threats and accident management measures and ageing phenomena. [1].

An important objective of the research activities conducted for ageing under FP-4 was to assess the safety margins in connection to plant ageing management. The emphasis was on knowledge of ageing related damage mechanisms, methods for detection and on-line monitoring of degradation and models allowing to predict the behaviour of components and structures. Another important objective was to propose qualified mitigation methods.

The objectives in FP-5 for what concerns operational safety are to provide improved and innovative tools and methods for maintaining and enhancing the safety of existing installations, for achieving evolutionary improvements in their design and operation and for improving the competitiveness of Europe's nuclear industry. Three top research priorities have been identified:

- Plant life extension and management
- Severe accident management
- Evolutionary concepts.

The objectives of FP-5 for what concerns plant life extension and management, which is much more end-user driven than FP-4, are:

- to develop a common basis for the continued safe operation and prolonging the safe operational life-spans of existing nuclear installations;
- to develop better methods for their inspection, maintenance and management (both in terms of performance and occupational exposure).

The following three sections have been proposed under this heading of the work programme:

- integrity of equipment and structures,
- on-line monitoring, inspection and maintenance,
- organisation and management of safety.

Besides the traditional technological challenges, socio-economic concerns are also taken on board, such as public acceptance and cost of the nuclear option as well as plant simplification and man-technology-organisation interaction.

An additional challenge for the EU consists of its enlargement towards Central and Eastern European countries in the coming years. Therefore FP5 pays also attention to plant safety assessments of Russian design reactors and to the spreading of the new safety culture in these candidate countries in co-operation with similar activities run at the Commission, especially under the programmes of Tacis/Phare and of the JRC (Joint Research Centre). For example, the JRC run networks in the field of structural integrity, the Network for the Evaluation of Steel Components (NESC), the Aged Materials Evaluation Studies network (AMES) and the European Network for Inspection Qualification (ENIQ) could provide a scientific support in clustering relevant activities.

4. OVERVIEW OF PROJECTS FUNDED UNDER FP-4 AND PROJECTS TO BE FUNDED UNDER FP-5 IN THE FIELD OF PLANT LIFE MANAGEMENT

Under FP-4 a total of 11 projects was funded in the field of ageing for a total value of EURO 3.5 million, out of which EURO 2.1 million was financed from the EU budget (i.e. 60%). All these projects were grouped in a cluster called AGE. Note that some FP-4 projects in the Accident Management Measures (AMM) cluster were also devoted to human problems as well as to maintenance and inspection problems, addressing hence M and O components of the TMO factor as discussed above. Under FP-5 so far a total of 16 projects with a EU budget of about EURO 9.2 million was selected (total project value estimated at about EURO 18.5 million).

In what follows an overview is given of the results obtained in FP-4 and the research which is conducted in FP-5 using thereby the structure of FP-5. Figure 1 shows a schematic representation of all FP-4 and FP-5 projects in the field of plant life extension and management. In reference [2] one will find a detailed overview of the EU sponsored research in the AGE cluster and more generally in the area of reactor safety under the FP-4.

4.1. Integrity of equipment and structure

4.1.1. Embrittlement

Irradiation embrittlement was the subject of a number of different projects funded under FP-4 and focusing on reactor pressure vessel (RPV) ageing aspects, such as:

- how to re-use already tested surveillance specimens in order to obtain more fracture toughness data;
- the way the neutron doses received by the material are measured in order to be able to compare results obtained from different reactors;
- the loading modes used to measure the fracture toughness.

A European code of practice for reconstitution of irradiated Charpy specimens (used to monitor the embrittlement behaviour of the RPV) was developed within FP-4 project RESQUE. This should allow to re-use surveillance specimens, which were tested destructively in order to determine the fracture toughness.

A guideline for a reference standard dosimeter was developed within FP-4 project AMES-DOSIMETRY. As a direct result of this project, this type of dosimeter is already applied by industry. A pilot database, containing important dosimeter parameters used in different countries, was developed within MADAM. These projects contributed to a harmonisation of the way neutron doses are measured in different countries which should lead to an easier comparison of data obtained under different circumstances in different countries and reactors.

Furthermore, validation work of different fracture toughness indexes was performed within the FP-4 project REFEREE. In this project the influence of dynamic loading (used for the surveillance specimens) on the shift of fracture toughness was studied with respect to the shift obtained under quasi-static loading (as is the case under real conditions in the RPV).

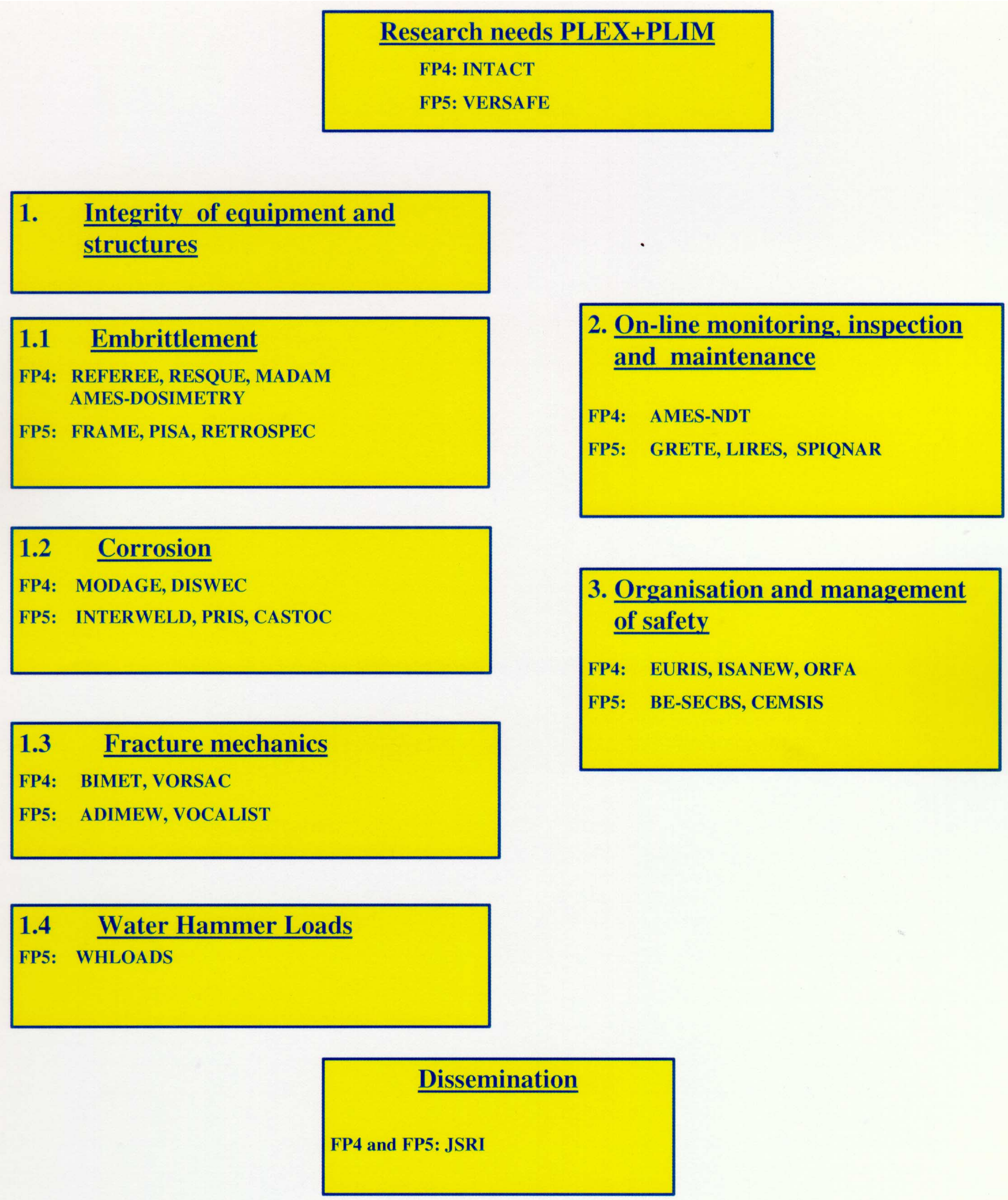

FIG. 1. Schematic representation of all FP-4 and FP-5 projects in the field of plant life extension and management, using the structure of FP-5.

The projects which have been selected for FP-5 in the area of embrittlement treat the following problems: improvement of fracture toughness measurements, dosimetry and role of impurities such as phosphorus in the embrittlement process.

Within FP-5 project FRAME work will be done to improve the assessment of the most important parameter used to measure the embrittlement conditions of the RPV. Currently this is done through indirect measurements in a rather conservative way (the so-called reference temperature methodology, which makes use of Charpy-V notch impact testing). It is difficult to estimate in a quantitative way the conservatism of this methodology. Therefore the work proposed will focus on the development of a method which allows to measure directly the fracture toughness. This should result in a better and more accurate estimation of the embrittlement conditions of the RPV material.

Within FP-5 project RETROSPEC work will be done to improve the evaluation of the neutron doses induced in reactor structural materials in those cases where no or unreliable data from surveillance specimens are available (for example the older generation of WWER-440 type reactors). The objectives of this project are to develop procedures and guidelines for retrospective dosimetry which are independent of the type of steel and of the time elapsed after removing the material from the reactor.

The objective of FP-5 project PISA is to better understand the role of phosphorus in the embrittlement process of RPV steels and improve the predictability of the impact it can have on embrittlement. Further work is still needed to understand, in particular, the segregation mechanisms of phosphorus to internal grain boundaries in RPV steels, as a result of exposure to irradiation or elevated temperatures, and the subsequent brittle intergranular failure of the material.

4.1.2. Optimisation of operational conditions focusing on corrosion issues

A review of plant concerns has revealed that besides embrittlement of the RPV also corrosion (i.e. damage induced by chemical reaction with the environment) is one of the more important ageing-related degradation mechanisms.

For example, ageing of reactor internals is in almost all cases associated with irradiation assisted stress corrosion cracking. For Boiling Water Reactors (BWR) the 2 major corrosion concerns at present are indeed cracking of the core shroud/plate and possible cracking in the lower plenum region.

Under FP-4, two projects were dealing with corrosion. Code validation work was done in the FP-4 project MODAGE on various corrosion models. Within the framework of FP-4 project DISWEC, guidelines were produced on the best use of methods to test dissimilar metal welds for their susceptibility to environmentally assisted cracking (EAC) and to generate quantitative data on crack growth rates in the regions of the dissimilar metal weld most sensitive to EAC.

In FP-5, two projects deal with irradiation assisted cracking of austenitic steels, the material used for reactor internals. Within the framework of FP-5 project INTERWELD the radiation induced damages that promote cracking in the heat affected zones of PWR and BWR core internal components will be studied looking at parameters such as neutron fluence/irradiation conditions, residual stresses, microstructural and microchemical conditions. Further work is also needed to produce materials data for irradiated stainless steels of LWR internals as a function of fluence up to 70 dpa. This will be done within FP-5 project PRIS.

Within FP-5 project CASTOC environmentally assisted corrosion of low alloy steels under static and cyclic conditions will be studied with the aim to improve service operation and code implementation.

4.1.3. Prediction of structural safety margins: behaviour of full-scale components (fracture mechanics)

The integrity of the bimetallic welds without and with hypothetical cracks has to be justified in all conditions for the whole life of the plant. Recent international surveys on BMW behaviour have shown that there are several types of outer surface cracking which may occur and which fall into the following two categories : those related to fabrication and those caused by surface corrosion. Classical fracture mechanics methods are difficult to apply to this specific case due to a number of complicating factors such as the prevailing mixed-mode loading conditions, the variation in material constitutive equations across the weld zone and the presence of a large residual field. In addition, test techniques in this area are not well developed or standardised.

A better understanding of the safety margins for bimetallic welds was achieved in the FP-4 project BIMET. Furthermore a series of prediction tools for the safety margins were developed within BIMET. The work done in BIMET allowed hence to make recommendations on the validity of existing structural integrity methodologies and to identify those areas where more innovative methodologies should be used. Further work is required to confirm these findings on real, scale 1 to 1, components, under prototypical loading conditions (e.g. temperatures in the order of 300 °C and bending facility). This is exactly the objective of the FP-5 project ADIMEW.

The generation of residual stresses during welding, their variation during service life under thermal and mechanical loading, and the effect of stress relaxation on creep damage in stainless steel have been examined in the FP-4 project VORSAC. Creep damage (at temperatures up to 1200°C) has been evaluated by modeling, scanning electron microscopy (SEM) and small angle neutron scattering. The project provides increased knowledge of the effectiveness of remedial methods including post-weld heat treatment and last-pass heat sink welding, and on the effect of service temperatures, load transients and constraint on stress variation during manufacture, heat treatment and service life.

Defect assessment techniques should be further improved to better predict safety margins, in particular with respect to the constraint effect (i.e. the pattern of crack-tip stresses and strains causing plastic flow and fracture), which gives rise to an effective toughness for components higher than that measured on test specimens. This is done in the FP-5 project VOCALIST.

4.1.4. Optimisation of operational conditions: dynamic fluid-structure interactions (water hammer loads)

It is worth recalling that decisions about repair/replacement/back-fitting and safe operational performances do not depend only upon the loss of integrity of the material (embrittlement, corrosion) but also upon dynamic loadings during operation (generated, for example, by condensation induced water-hammers in pipes and open networks). This issue is treated in the FP-5 project WHLOADS.

4.2. On-line monitoring, inspection and maintenance

Under FP-4 only 1 project, AMES-NDT, was dealing with this issue. The objective of the concerted action AMES-NDT was to verify to which extent non-destructive testing (NDT) techniques can be used in order to assess material damage. It resulted from the work done that some of the NDT techniques applied such as for example thermal power measurements showed promising results. Further validation work is required in order to verify to which extent they can be applied in-service on real components. For other NDT techniques (e.g. ultrasonic back scattering and absorption, positron annihilation, etc.) more development and validation work is required to evaluate their potential to support the decision-making process for failure prevention (i.e. change of operational parameters, increased inspections or replacement of components), focusing, for example, on irradiation damage in RPVs and thermal fatigue in pipings. This work will be continued under FP-5, as a shared cost action within GRETE.

Within FP-5 attention will also be devoted to improve the inspection performance of ultrasonic inspection aimed at detecting and sizing of possibly present cracks in structural components (FP-5 project SPIQNAR). Specific issues addressed in this project are:

- Development of a reliable methodology to produce synthetic defects (or "virtual defect" signals), which mimic the response of certain important service defects and this with the aim to improve inspection qualification methodologies;
- Development of signal processing techniques for ultrasonic inspection techniques for the detection and sizing of cracks in relevant austenitic components.

As already mentioned before some of the major corrosion problems for Light Water Reactors (LWR) are IASCC damage to core internals and IGSCC damage to steam generator tubing. The remedial actions (e.g. the best chemistry conditions) are still unclear. An important parameter for IASCC, like any other aqueous corrosion process, is the corrosion potential, which should be measured as an on-line monitoring technique inside the reactor. For that purpose use is made of a robust reference electrode. The development of such electrodes which can work in the harsh operational conditions of LWRs is the objective of LIRES.

4.3. Organisation and management of safety

Under FP-4, several projects devoted to accident management measures in the AMM cluster and 1 project in the AGE cluster were dealing with issues on organisational matters.

Under the AMM cluster a thematic network (FP-4 project ISANEW) was set up in order to develop and compare different methods in order to study the interaction between human and technological systems. Attention was focused on methodologies based upon the integrated sequence analysis (ISA) of possible events, considering both the human system and the physical process. Sequences of events are simulated through modeling of the plant system and the actions of the operators. Input data used for the simulation come from different disciplines such as probabilistic safety assessments (PSA) and human reliability analysis studies. The work within this project has shown that the essential elements for an integrated sequence analysis are already available.

The concerted action ORFA looked at organisational factors and how they influence nuclear safety. In many studies it is recognised that organisational factors are often the root cause of incidents and accidents. However, there is unfortunately no agreed and validated method for their assessment. Important issues for short-term research are related to the identification and description of those factors which define good practice, the development of organisational self-assessment tools, the inclusion of organisational factors in incident analysis, definition of methods of how to maintain the corporate knowledge, etc. Long-term research needs are related to the development of pro-active methods of organisational design and methods of integration of organisational factors in PSA models and theoretical models of the interaction between organisational factors and performance of crucial components.

Within FP-5 the following issues are addressed in already approved projects:

- computer-based systems embedded in a nuclear installation to support instrumentation and control (I&C) functions important to safety within FP-5 project BE-SECBS;
- the development of a safety justification framework for the refurbishment of systems important to safety (SIS) that is acceptable to different stakeholders (especially licensing bodies and utilities) within FP-5 project CEMSIS.

Guidelines for a European framework for risk informed inspection were developed within a thematic network, which was driven by the utilities (FP-4 project EURIS). These guidelines should be able to identify safety-significant categories for power plant components and to optimise the targeting of costly inspections. It includes feedback from plant operation and must indicate the specific components and the locations to be inspected, the defects to be detected and the performance in detection and sizing to be achieved. The methodology integrates actions or mitigation methods other than inspection, in order to manage the risk. As a consequence risk informed in-service inspection (ISI) should, by this optimisation of the ISI programme, reduce the cost and efforts whilst maintaining safety at its currently high level or above. These guidelines will be used as the basis of a further discussion first between utilities and then between utilities and regulators in order to verify how they can be implemented in practice.

4.4. Networking

As far as Western reactors are concerned, under FP-4, some thinking about a strategy for PLEM was started under the concerted action **(**INTACT**),** devoted to the integrity assessment of ageing components in general, considering both metallic and non-metallic components. The purpose of this concerted action was to provide recommendations on future research needs.

With respect to enlargement it is important to mention that in the area of nuclear safety the applicant countries “should co-operate fully in efforts to bring their levels of nuclear safety up to international standards” in accordance with the approach of the G7 since 1992 and that “the Union should co-operate closely with the safety authorities of the countries concerned to create a climate favorable to nuclear safety”. The challenges in the area of nuclear safety are actually of both an organisational and a technological nature. They can be solved in part by

Community research involving both West and East European partners: particular aspects such as plant life management and accident prevention/mitigation strategies will be dealt with in a new concerted action VERSAFE, which is funded under FP-5.

In FP-5 a special effort will be done for clustering of the activities and also establishing interfaces/contacts with other related Community activities. This issue is specifically addressed in the next call for proposals which will be published mid October 2000 (see also next section).

5. CONCLUSIONS AND FUTURE PROSPECTS

5.1. Conclusions of FP-4

The safety issue considered in the AGE cluster under FP-4 was that of the continued safe operation of ageing nuclear power plants. The cluster was composed of 7 cost-shared and 4 concerted actions, grouping all together 35 institutions from the EU and Associated Countries (including Switzerland). The 4th framework programme was the first one, in which the issues of ageing and plant life management were addressed in a systematic way. The results generated within the projects allow to validate assessment tools for ageing behaviour (considering corrosion, fracture mechanics, non-destructive testing and irradiation embrittlement). This validation work will lead to a reduction in uncertainties, increasing hence the safety – and the performances - of nuclear power plants.

An important objective of this research was to assess the safety margins, using different assessment tools, and so to improve plant ageing management. The emphasis was on knowledge of ageing related damage mechanisms, methods for detection and on-line monitoring of degradation and models allowing to predict the behaviour of components and structures. Another important objective was to propose qualified mitigation methods.

Worth mentioning is also the wider trans-European collaboration which exists in the area of reactor safety research through many bilateral and Community programmes which are described in the Joint Safety Research Index (FP-4 concerted action JSRI). These RTD programmes are also described in the homepage http://www-is.ike.uni-stuttgart.de/sinter developed by the R&D Network on Safety-Related Innovative Nuclear Reactor Technology (FP-4 project SINTER).

5.2. Prospects under FP-5

The activities under the 5th framework programme in the field of ageing build further on the activities conducted under the 4th framework programme, considering aspects of both safety and cost. The objectives are to develop a common basis for the continued safe operation and prolonging the safe operational life-spans of existing nuclear installations and to develop better methods for their inspection, maintenance and management.

The following specific issues are addressed in the 16 projects, which started just after the summer 2000:

- Integrity of equipment and structures: embrittlement, corrosion, behaviour of components as a whole (fracture mechanics) and water hammer loads;

- On-line monitoring, inspection and maintenance: development of innovative techniques for monitoring of corrosion conditions and material damage and of improved ultrasonic inspection techniques;
- Organisation and management of safety: cost-effective modernisation of systems important to safety and safety evaluation of computer based systems.

The next dead-line for submission of research proposals on Nuclear Fission, Radiation Protection and Support for Research Infrastructure is 22 January 2001. In view of this deadline the Commission has on the 25 July 2000 adopted a revised Work Programme. It describes the overall objectives of the programme and gives the priorities and indicative total budget for proposals to be evaluated after the deadline of 22 January 2001. The revised Work Programme [4] is available on the Cordis web-site: www.cordis.lu/fp5-euratom/docs/
For what concerns plant life management the following issues are mentioned:

- embrittlement/re-embrittlement of RPV material,
- thermal fatigue/corrosion,
- ageing of concrete: development of innovative on-line monitoring and inspection techniques,
- risk informed methodologies,
- organisation and management of safety.

Note that also WWER operational safety issues such as embrittlement and accident assessment and management are mentioned.

In FP-5, a special effort will be done for clustering of the activities and also establishing interfaces/contacts with other related Community activities, thereby contributing to a larger dissemination and transfer of knowledge gained through national and Community programmes. This issue is specifically addressed in the updated work programme.

More generally, a new impetus will be given to the integration of all European research efforts, especially in the light of the current discussions around the Commission Communication “Towards a European research area” [5].

5.3. Dissemination of community research

More information about the above research themes, which are implemented primarily by DG Research as “indirect” shared-cost actions can be found in homepage http://www.europa.eu.int/comm/dg12/rtdinfo.html). The Joint Research Centre (JRC), with its seven institutes, is the Community’s own research Centre which contributes to the implementation of the framework programmes by carrying out “direct” actions (see homepage http://www.jrc.org/jrc/index.asp). Further information about Community research policy and rules can be found on the WWW site of the Community R&D Information Service (http://www.cordis.lu).

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DESIGN SAFETY IMPROVEMENTS OF KOZLODUY NPP TO MEET THE MODERN SAFETY REGUIREMENTS TOWARDS THE OLD GENERATION PWR

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Abstract

Activities related to safety improvement of Kozloduy NPP units, started at the end of 1970s included seismic resistance upgrading, fire safety improvement, reliable heat final absorber etc. During the last 10 years the approach was systematized and improved. Units 1 to 4 are of great interest; therefore here we will discuss these units only. As a result of studies and analyses performed at the end of the 1980s and the beginning of the 1990s, problems related to the safety were identified and complex of technical measures was developed and planned. A considerable part of these measures has already been implemented, and the rest will be performed during the next years. Activities were performed by stages, and at the moment the last stage is under way. It shall be finished by the year 2003. The number of the measures is quite large to describe them here in full scope — during the first stage of the safety program (1991-1993) were developed and analyzed more than 4200 documents and more than 160 measures were executed. During the second and third stages more than 300 important improvements were realized. In the frame of the program, financed by EBRD, 10 new systems with great importance were implemented and 8 systems were significantly modified. The main measures are described below.

1. PRIMARY CIRCUIT INTEGRITY

The main problem is reliability of primary circuit — RPV, main circuit piping and steam generators. Only after proving the possibility for safe operation of this equipment, we can talk for acceptable safety level.

1.1. RPV integrity

RPV integrity is of great importance. Problems related to the low temperature embrittlement of RPV metal and welds are well known.

Analyses were performed by the support of OKB "Gidropress", Siemens and Westinghouse, including analyses of RPV metal samples for defining the chemical composition of welds metal, neutron fluence and critical temperatures. Annealing of the metal was performed.

Results definitely show, that all the RPVs have the necessary lifetime to be operated by the end of their design lifetime; moreover they can be in operated after expiring of the design lifetime.

Additionally preventive measures were taken: screen-cassettes were installed and low-leakage refueling systems are adopted in order to reduce the fluence on the core periphery; boric water temperature in the emergency storage tank is kept at 55 C; low temperature overpressure protection is implemented; yearly an extended scope ultrasonic metal control is performed according to the program agreed with BNSA. Modifications in the reactor emergency protection system are executed.

1.2. Primary circuit piping integrity

One of the main purposes was to prove applicability of LBB conception. Destructive testing of metal samples was done. Also stress and fatigue analyses were performed, considering the actual data for seismic impact in the KNPP site. These analyses show the expediency of additional seismic reinforcement of equipment and piping, and reinforcement was executed later on.

An extended scope ultrasonic metal control program is agreed with BNSA and adopted.

Operation manuals (instructions) were analysed and number of improvement was made.

At the moment, the probability for primary circuit piping break is evaluated on less than 1.x 10-5 y-1. The result shows, that SGs are in good condition.

A large scope of activities related with the SG RLT evaluation was performed. Criteria for SG pipes plugging was defined. Special program for SG pipes status control is adopted and performed.

Stainless pipes replaced SG blow-down pipes.

1.3. Overpressure protection

Cold overpressure protection was implemented within the program, financed jointly with EBRD. It includes complex of valves and control system.

The complex of relief - isolation valves includes one relief valve and isolation valves. The valves are qualified for all expected media and for the environmental conditions expected in case of accidents. The control system follows the primary circuit parameters, compares their values with the preliminary set initial values and in case of necessity produces a signal for valves opening and closing correspondingly.

The system is multifunctional. The actuation levels of the pressurizer's main safety valves and the complex of relief - isolation valves are set in such manner, so the last one can serve as a relief control valve. Also feed and bleed procedure can be realized through this complex of valves, if necessary.

2. LBB APPLICATION STATUS

Primary circuit large diameter piping break is not considered in the design of the units. Therefore, it was of great importance to prove the applicability of LBB conception.

For that purpose, a number of studies and analyses were performed including metal samples tests, and they proved the applicability of the LBB conception. As a result, acoustic leakage control system ALUS was installed, covering also surge line DN 200 pipes, and the system shows good results. Additionally for the same purpose, two new systems of that type are now in process of qualification.

Another information system was installed on the surge line and database was created for real time monitoring of pipe metal temperature to evaluate the possible effect of the fluid stratification. It was proven that the stratification phenomenon does not create unacceptable stresses on the pipe metal.

The combination of stress analyses results, additional reinforcement and extended metal control, guarantees the reliability of piping system.

3. DBA AND TRANSIENTS, SEVERE ACCIDENT ANALYSES

It is well known, that the original design of the units considers DN 32 mm piping break as a DBA.

To evaluate the consequences of breaks of another diameters primary circuit pipes, a large scope of activities was performed, by the help of IAEA (Regional Project RER 09/002)~ WANO (6-months program for KNPP), and Westinghouse (transfer of know-how).

Events were analyzed, such as:

- main steam line piping break;
- feed water line piping break;
- SG pipes break;
- Boron solution dilution;
- And number of others measures.

The results of analyses, performed during the first stage of work; showed that the units have large reserves and in some defined conditions also leakages, larger that the considered by DBA, can be controlled. At the same time, the necessity for realization of some technical measures was defined. All necessary measures were implemented and now the units are in a status, which allows them to overcome all the postulated initial events (design deviations, abnormal conditions and accidents).

LOCA analyses were made in two stages. During the first stage, DBA was extended up to LOCA DN 100 and several modifications were implemented to increase the ECCS capacity and reliability.

The second stage started at the end of 1999, in order to prove that LOCA DN 200 mm can be accepted as DBA, considering the current capacity of ECCS and Localization system. This DBA provides significant overlapping with the pipe breaks, prevented by LBB.

The analyses were finished in the middle of year 2000, and show that the acceptance criteria for DBA are met. These criteria are related to the fuel cooling and radiological consequences. Now the results of analyses are in process of licensing by the regulatory body (BNSA).

Simultaneously with the design events analyses, considerable work was done to study the severe accidents consequences. Models were created and computer codes were verified. Analyses of different events were performed.

Based on the updated results, which were obtained, new accident instructions and new emergency response plan were developed.

4. IMPROVEMENTS OF SAFETY AND SUPPORT SYSTEMS

During the first stage of the work number of improvements were performed for the systems, important to safety. These improvements were directed to eliminate the most significant insufficiencies and they did not completely solve the problems. Only during the second and third stages of the work a number of new systems were installed, which necessity and effectiveness was proved by the performed analyses, described above. Generally, we can affirm that the implemented measures allow reducing of the probability for core damage accidents about ten times, which was shown by the PSA-level 1. Besides that there is some not completely eliminated problems, the positive effect of the implemented measures is obvious.

At the first stage, number of modifications of schemes was made, with the purpose to prevent boron dilution in primary circuit, to reduce the single failure and common mode failure probability. The effectiveness of primary circuit emergency supply systems and spring system was increased. The reliability and effectiveness of the SG emergency feed water systems are increased. Reliability of the emergency power supply systems and control systems is increased.

The main part of the new systems were designed and implemented in the frame of the program, financed by EBRD:

B.1 - Installation of Fast Closing Main Steam Line Valves;

B.2 - Replacement of Steam Generators Safety Valves;

B.3 - Installation of Pressurizer Valves for Feed and Bleed;

B.4 - Installation of Steam Generators Leak Survey System-,

B.5 - Construction of Complementary Emergency Feed-water System;

B.6 - Implementation of Safety Parameters Display System;

B.8 - Installation of Generator Breakers;

B.9.1 - Second Fire Protection Pumping Station.

It should be emphasized, that all the measures, which were undertaken, are not separate. They are mutually related and completed technical decisions, in frame of one general conception.

For example: Installation of Fast Closing Main Steam Line Valves ensures protection of RPV against thermal shock; but in combination with measures for whip effect protection, it provides SG isolation in case of common mode failure in turbine hall; replacement of SG Safety valves protects primary circuit against overcooling, but at the same time ensures its cooling through steam relief to the atmosphere with the simultaneous feeding to the SG from CEFWS (primary circuit feed & bleed procedure) down to a temperature, which allows to start water-water cooling by CEFWS. Installation of Pressurizer Feed & Bleed Valves (relief-isolation complex of valves) provides primary circuit protection against cold overpressure, but also allows implementation of primary circuit feed & bleeds procedure, if necessary; and moreover, serving as a relief control valve it reduces the risk for opening and no closing of Pressurizer main safety valve. SG leakage control system provides minimization of radio-nuclide deposition to the environment, from one side, and from the other side, reduces the risk of SG header break.

Probably, it will be more interesting to discuss the CEFWS. However, functioning of this system should be examined taking into account the measures, performed on items B.1, B.2 and B.9.1 of the EBRD Project.

CEFWS is situated in two separate buildings. In each of the buildings two separate and equal loops of the system are installed and each loop has 100% capacity for cooling of one unit.

Pumps, heat exchangers, auxiliary systems and diesel-generators with the respective electrical equipment, are located in the buildings. Unsalted water storage tanks are designed for each building. Heat exchangers' cooling is made by the new fire protection pump station (item B.9.1), and also possibility for cooling water feeding from the artesian pump stations is provided. The same cooling water is fed to the spent fuel tanks heat exchangers of the units, and cooling of the units' diesels generators can be provided if necessary.

The pumps, located in the buildings, are connected to each SG via independent piping system (two systems are available working in parallel). These piping systems are designed and installed in a manner to provide full reliability of feed water supply to the SGs. They are completely separated from the units' feed water systems, originally designed.

In case of failure of all the systems in turbine hall, steam line are isolated by fast closing main steam line valves, (item B.1), pumps in the CEFWS buildings are working to feed water from the emergency water storage tanks to the SGs, and the generated steam is relieved through the SG SV (item B.2). This way cooling of the reactor installation is executed down to temperature, at which water-water cooling is started. The scheme is switched and SG cooling is started through the CEWFS heat exchangers, reaching the reactor installation cold condition, at which the installation can be kept during unlimited time. Emergency water storage tanks can be fed with water from the fire protection systems or by service water, if necessary.

Of course, execution of the described technical measures does not solve all the problems and the work is gone. In 1997 Program for reconstruction (modernization) of the units (PRG'97) was developed, and the bigger part of the program has already been implemented. The program was upgraded in the year 2000, (PRG'97/A) and is planned to be finalized during the next two years. Some of the most important measures are:

- instalation of new localization system of units 3 and 4 — through installation of jet vortex condenser, low mass flow rate filter, in combination with filter venting system and hydrogen detection and recombination (or burning) system;
- providing reliable core cooling in case of break of main primary circuit pipe DN 500 mm;
- elaboration of full scope SAR for units 1 to 4 according to the TOR, approved by NEK in 1998;
- performance of PSA-level 2 analyses for units 1 to 4;
- strength analysis of RPV internals in conditions of the new DBA LOCA DN 200, according to the contemporary norms;
- determination of probability for RPV destruction;
- development of a program for control and management of RLT of equipment and structures of units 3 and 4.

It can be affirmed, that the performed technical measures and those, which execution is planned in the upgraded program PRG'97/A, will provide full conformance with the recent norms for acceptable safety level.

Together with the described modernizations, a number of studies and improvements were performed in the control & management systems and electrical power supply systems. Some of them were mentioned here. Because of the limited scope of this report (presentation), they will not be discussed in details.

5. CONFINEMENT LEAK TIGHTNESS AND STRENGTH

It is necessary to discuss especially the problem, related to the confinement tightness and effectiveness of localization systems.

The tests, performed in 1990-1991 revealed leakages from the confinement, more than 2000 volume%/day.

Study was performed according to a specially developed methodology, critical elements were defined and measures for improvement of the tightness were established. As a result of the implemented measures, the tests, repeated in 1995-1996, showed un-tightness about 350 volume%/day.

At the moment, new program is developing for new tests and studies, and this program shall ensure better confinement tightness.

Simultaneously with the tightness improvement, structure integrity of civil constructions was analyzed in case of extreme impacts, particularly under the maximum acceptable pressure in the compartments (1996-1997). Civil constructions model was developed, and it was studied using a modem methodology and advanced computer codes, and considering the impacts of the results of the performed severe accidents analyses.

In the frame of the studies, related to the new DBA LOCA DN 200 mm, once again static and dynamic structural analyses were performed, and they confirmed the existing civil construction stability.

The central project in the modernization program is the new localization system of the units, based on the jet vortex condenser. In parallel with this, series of new studies are planned for defining the civil construction behaviour after installation of the new system's components. The work is expected to be finished at the beginning of the year 2001 for unit 3), and in 2002 for unit 4.

6. SEISMIC SAFETY IMPROVEMENTS

After the earthquake in Vrancha in 1987, reassessment of the seismic impact was done, and number of measures for improvement the seismic stability of civil constructions and equipment was performed. After the year 1989, IAEA Project (BUL/9/012) was initiated. Within this project additional investigations of the seismic impact on the site were executed, considering also the effect of the local earthquakes. New seismic characteristics of the site were defined SL — 1 = 0.1 g (10^{-2} y^{-1}) and SL — 2 = 0.2 g (10^{-4} y^{-1}). Using the new characteristics, new response spectra were developed (defined) and behaviour of all the components, important to safety, was analyzed under this impact.

Critical elements were defined and designs were elaborated for their anti-seismic reinforcement.

Now, the implementation of these projects is almost finished.

7. FUTURE PROJECTS

Based on the significant design modifications, implemented during the last decade, KNPP has recently adopted and declared a project for upgrading of the design of units 3) and 4 to a new model (WWER-440/B-209M), which satisfy the current international safety standards and completely meets the safety standards in Bulgaria.

The original special safety features of these units (safety systems similar to the B-213 model - 3 -channel structure of the safety systems, system physical separation and redundancies of ECCS, Emergency Control Room etc.) have an important role for achieving of this target.

This project is next step development of the Complex program for modernization and it will be implemented by the year 2003. It is planning to apply for plant re-qualification procedure in Bulgarian Safety Authority in 2002.

UPGRADING UKRAINE'S NUCLEAR POWER PLANTS

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Abstract

In the article materials the description of the historical process of Ukrainian NPP modernization is presented. The existed procedure of modernization implementation is described in details. The main features of this procedure are illustrated on several examples like of ECCS sumps, improvements of hydrogen recombination system and development of RPS control rod revision program.

1. INTRODUCTION

Necessity to keep acceptable level of NPP safety requires constant search of perfection while undertaking of the activities connected with its operation. One of the directions concerning the mentioned activity is development and implementation of the measures aimed at NPP upgrading and its safety improvement. Among the conditions stipulating implementation of such upgrading the important place is attributed to the new understanding of the peculiarities as regards functions of systems and activity undertaken by staff appeared as the result of experimental and research work along with assessment or reassessment of safety and analysis of operational experience.

This work presents the results of addressing of various aspects of upgrading performed at NPP in Ukraine. The results of operator and regulator are presented at the certain examples as to adequacy of the decisions made to the real conditions of units operation along with the requirements established concerning provision for operational safety.

2. HISTORY OF NPP IN UKRAINE UPGRADING

Designs of units with WWER under operation in Ukraine were developed in 70-s. The basic safety criterion laid into these designs was to provide for safety under all the design basis accidents/at that time, exclusively deterministic approach to safety level assessment was adopted as the basis.

Working designs were prepared in compliance with the regulations in force by the moment. In part, some measures were implemented during working design development aimed at provision for requirements of OPB-82 meeting.

During operation of units with WWER, also, as the new norms and rules were developed and enacted, the attempts were made as to put them into compliance with these regulatory requirements. However, these measures were of local nature and in many cases unfeasible from both technical and economical reasons.

Since the middle of 80-s USSR started work on improvement of all the NPP under operation safety. The "Integral Measures to Improve Safety of Power Units under Operation (SM-88, SM-90)" were developed based upon analysis of units assembling, start up and operation.

Their implementation was also started at NPP in Ukraine. However, substantiation of the units safety performed before did not completely take into account necessity of the efficient analysis of operational experience, assessment of common cause failures (of natural and technogeneous origin) impact, analysis of beyond design basis accidents and development of the measures to manage them as well as a number of other provisions.

One of the first decisions made by the Ukrainian National Regulatory Authority — the State Committee of Ukraine on Nuclear and Radiation Safety (UkrSCNRS) — was devoted to necessity of detailed safety assessment for all the NPP under operation and development of "Safety Analysis Reports" based upon this assessment results. These documents included the working programs on the further improvement of each unit under operation safety based upon the approved by the Regulatory Body (RB) "Program on Safety Improvement of NPP with Rector Facilities WWER-1000, WWER-440". Besides, UkrSCNRS had approved the "Program on Immediate Work on Improvement of NPP with WWER-1000, 440 Safety" identified two levels of the working priorities directed to improve safety and reliability of NPP operation.

Under development of the "Program on Safety Improvement of NPP with Rector Facilities WWER-1000, WWER-440" as the methodological basis the IAEA classification was adopted as to extent of impact from considered deviations from upon the in-depth-protection (IAEA-EBP-WWER-05). Subdivision of deviations into four categories according to their safety-significance allowed to identify the main directions of work and list of the immediate measures of significant impact upon safety which could be undertaken in the visible terms, namely, by 1998.

As scientific-technical and normative base of nuclear power in Ukraine was developed, the criteria for safety level assessment had substantially changed. This resulted in that, by beginning of 1997, the new requirements to contents of SAR were formulated concerning units of NPP under operation in Ukraine. Besides, the necessity had grown to develop the long-term programs on upgrading of units.

According to the RA order and under participation of IAEA the long-term programs were developed in Ukraine as to improvement of safe and reliable operation of each unit. These programs, though being provisional ones, nevertheless, take into account a number of international recommendations along with the previous experience gained as to similar activity undertaking. The results of SAR development shall be used to improve the named programs with the priorities of particular measures undertaking intended to improve safety.

Below some aspects of units at NPP in Ukraine upgrading are described at particular examples.

3. PROCESS OF TECHNICAL DECISIONS IMPLEMENTATION IN CONNECTION WITH UPGRADING OF UNITS AT NPP

3.1. System of hydrogen removal from SLA under emergency modes

The Ukrainian RA had developed the procedure on licensing of the technical decisions connected with upgrading of NPP units. As an example of this procedure implementation the measure on development of the system to remove hydrogen from SLA under emergency modes can be named under development at Rivne NPP in compliance with the "Program on Immediate Work on Improvement of NPP with WWER-1000, 440 Safety".

Problem of danger caused by hydrogen is one of the most significant among NPP safety aspects due to possibility of hydrogen burning and explosion inside leaktight premises during normal operation of NPP and under emergency modes.

According to the program on improvement of units 1 and 2 (RF WWER-440/V-213) safety at Rivne NPP, the decision was made to develop the measures on removal of hydrogen from leaktight premises.

Implementation of this project started on-site in 1997 and goes on in compliance with the procedure developed by the RA as to licensing of technical decisions connected with upgrading of units at NPP. This procedure envisaged four stages of the licensing process.

Stage 1 - Reviewing by RA of the Technical Decision concerning the concept of upgrading.

SSTC NRS performed technical evaluation ordered by the RA concerning substantiation of necessity to install modules of passive catalytic after-burners of hydrogen (PKDV). In the frames of the assigned task the following aspects presented in the Technical Decision were analyzed:

- Classification of on-site elements according to the “Overall Provisions on Nuclear Plants Safety Assurance";
- List of norms and rules which requirements they shall meet;
- Potential suppliers of the system;
- Expected result;
- Provisional assessment of impact caused by reconstruction upon safety.

Also, under technical evaluation the following items were addressed:

- Initiating events during which hydrogen is escaped into leaktight premises;
- Qualitative analysis of possible concentration and places of hydrogen concentration inside leaktight premises;
- Technical options proposed for implementation of the system for removal of hydrogen or prevention from its accumulation in dangerous concentrations.

SSTC NRS had proposed to the RA the results based upon the completed analysis which indicate the safety deficits along with deviations from the requirements posed by SSTC NRS to be eliminated by installation of PKDV modules. Besides, the recommendations were proposed to the RA as to well grounded bases and expediency of this Technical Decision implementation.

Stage 2 — Reviewing by the RA of the Technical Conditions on supply of the system equipment and elements along with certification of the delivered elements. By the moment, this is just the stage of this Technical Decision licensing process. It shall be noted that since this project is implemented as part of the TACIS Project, then before identification of the bid competition concerning deliveries, only provisional Technical Specifications were submitted to the RA.

SSTC NRS analyzed the submitted documents from the point of view on requirements to be met by this article along with comparison as to requirements of norms and rules in force in Ukraine.

In the nearest future, after nomination of the certain supplier, the final package of the documentation concerning this supply will be submitted to the RA. If necessary, possibility will be envisaged concerning certifying and qualifying tests.

Stage 3 — This stage is aimed at assessment of how this project provides for system of the assigned functions performing under all the most unfavorable combinations of conditions and loads described in the normative-technical documentation (NTD) in force. Under technical evaluation of the SAR for this system submitted to the RA SSTC NRS will estimate results of the quantitative analysis as to emission of hydrogen, possibility of its explosion dangerous concentration formation and place of its possible accumulation under the most unfavorable initiating events among those addressed at the 1st stage.

Stage 4 — Agreement with the RA upon of the Technical Decision on the system commissioning together with the act on completion of assembling and start-tuning work, program and results of its testing at unit, necessary amendments of existing operational documentation.

The proposed procedure on licensing of the Technical Decision on assembling of hydrogen after-burners illustrates the overall requirements imposed by the RA concerning modernization of units at NPP. The below presented examples show some factors which require adjustment of these requirements and separate decisions to be mage by the RA as regards issues of upgrading.

3.2. Screen-type structure of ECCS sumps

ECCS includes emergency sumps where water is collected from spraying system pumps and from other safety system pumps connected to collector of water. By the norms and rules in force in Ukraine, the following is required: "Structure of water collector shall include protection against dirty, for example, filtrating elements (multi-row labyrinth screens, grates) and exclude loss of water under any mode of NPP operation". At that time, structure of filtrating elements and intake devices shall provide for simultaneous operation of all the pumps connected to them without failure of supply taking into account of delayed water supply into water collector from the premises of accident localizing. To provide for the above mentioned requirements the filtrating elements (screen-type structures) designed to prevent from debris or fragments of destroyed thermal isolation (under maximum design basis accident) ingress into emergency sumps shall be installed there.

To meet the above mentioned requirements of norms and rules in force in Ukraine at units 1 and 2 of Rivne NPP in 1998 the Technical Decision “On assembling of screen-type structures in emergency core cooling system” was developed and submitted to the RA for reviewing. The 1st stage of the licensing process was successfully completed. Implementation of this Technical Decision was one among the special prerequisites imposed by the RA for issuing of permission for unit start up and operation after PPR-98.

However, under performing of the 2nd stage of licensing it revealed impossible to purchase the design of screen-type structures installed in Hungary at NPP Packs and passed natural tests in Finland, recommended by the RA to implementation at Rivne NPP. To cope with this problem Rivne NPP signed the contract with the Russian “Atomenergoproject” (St. Petersburg) which envisages to link design of these structures to that implemented at Kola NPP in Russia.

Under technical evaluation of this design SSTC NRS found the concept of this design decision inefficient proceeding from the results of the similar screen — type structures testing at unit 3 of South Ukraine NPP and unit 5 of Zaporizhya NPP. The results obtained under testing did not confirm that these structures under maximum design basis accident would not be completely clogged by thermal isolation, thus this construct would provide for performing by ECCS its designed safety functions. Therefore, Rivne NPP submitted application for TACIS-97 Program for theme: "System of screen-type structures of ECCS. Units 1,2".

Taking into account that since acceptance of such TACIS application through completion of the Project not less than 5 years passes, the RA issued permission to assemble temporary screen-type structures designed by the Russian "Atomenergoproject" (Saint-Petersburg). Besides, the RA established additional constraints such as: obliged Rivne NPP to develop the "Program on checking of LP ECCS operability under 50% obstruction of filtrating surface of screen-type structures installed in box SG of ECCS sump".

In 1998 the Technical Decision "On assembling of temporary screen-type structures in emergency core cooling systems" was implemented. The "Program on checking of LP ECCS operability under 50% obstruction of filtrating surface of screen-type structures installed in box SG of ECCS sump" was submitted to the RA followed by the information on results of assembling and start-tuning work.

Presently, Rivne NPP develops the projects "System of screen-type structures of ECCS. Units 1,2" and "System of hydrogen after-burning inside leaktight premises" as part of the TACIS Project. These projects in parallel pass all the licensing stages, namely: as it was already mentioned, the "System of hydrogen after-burning inside leaktight premises" is now at the stage of supply tender identification.

3.3. Reactor control and protection system

Designs for units with WWER under operation in Ukraine were developed in 70-s. structural requirements to FA (Fuel assembly), CPS control rods and geometry of IRD provided for normal operation of reactor during 2-years fueling campaign. However, in 1992–1993, due to transfer to 3-years campaign occurrences of CPS control rods operational malfunctions were registered at NPP with WWER -1000 which were exceeding of the designed duration of their insertion into core. To arrange and coordinate the work on clarification of CPS control rods operational malfunction causes Heads of operating organizations in Ukraine and Russia established the inter-agencial commission.

In the frames of the "Program and Methodology of Testing to Check Passability of CPS Absorbers" developed by OKB "Hydropress" and adopted in 1993, at units of NPP in Ukraine the studies were undertaken which showed that the basic cause of the regulated value exceeding as to time of CPS control rod dropping into core is bending of FA which causes appearing of additional forces of friction between linear absorber and wall of FA.

The technical decisions directed to upgrading of CPS control rods and adjustment of IRD (Inside reactor devices) were developed based upon the obtained results. They were reviewed by the RA and agreed upon under the established order.

Since 1995, to reduce bending of FA and provide for safe operation of units at NPP with reactors of WWER-1000 type in Ukraine the following measures undertaking started during planned outages:

- Adjustment of reactor UPT (The unit of protective tubes) position to reduce compression of spring blocks in FA heads;
- Perforation of CPS drive stems to reduce force of hydraulic resistance under insertion of control rods into core;
- Increasing of CPS drive stems weight or use of absorbers with increased weight;
- Pulling and reciprocation of CPS absorbers along guiding channels of FA;
- Replacement of damping tubes.

These measures are still implemented presently.

During the measures development and implementation directed to exclude not designed time of CPS control rods dropping, according to the RA Resolution of 08.06.93, the periodicity of CPS control rods testing was changed. Later on, in 1997, the necessity was confirmed as to testing with this periodicity, and the requirement established to perform one additional testing in the middle of campaign at the level of 50% N_{rated}. The technical measures undertaken at units of NPP in Ukraine along with the work performed during each PPR on completing of control rods resulted in considerable improvement of temporal characteristics as to dropping of CPS control rods with practical exclusion of some control rods dropping duration (Fig. 1 and Table I).

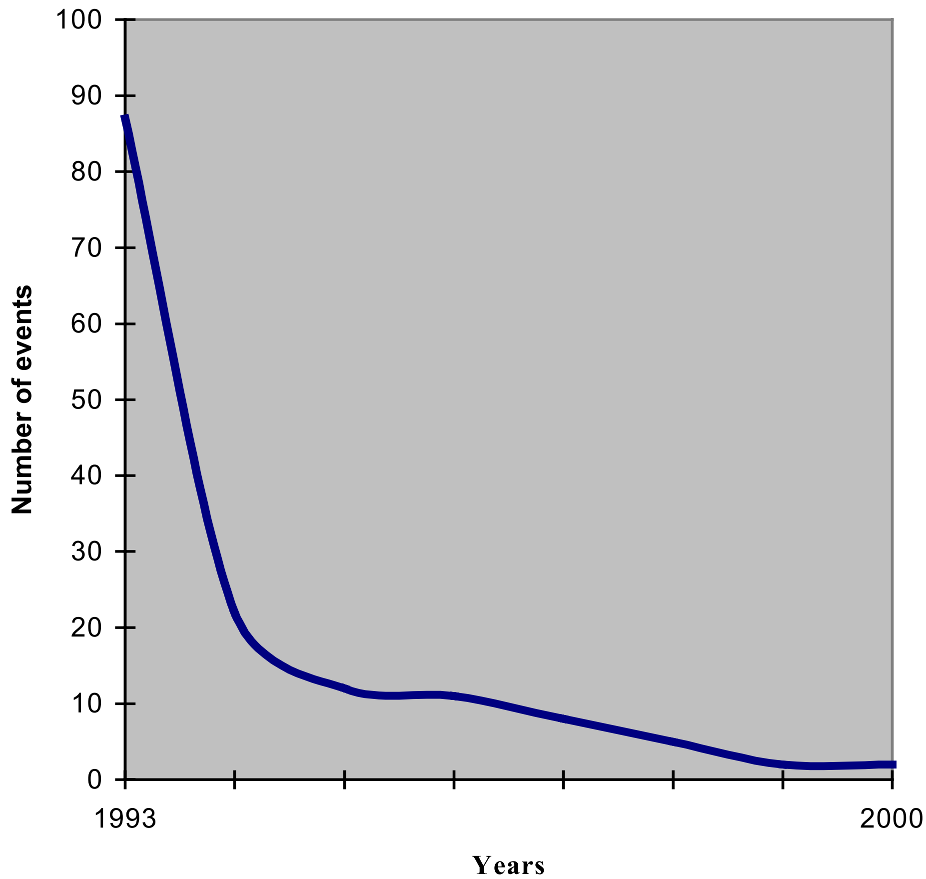

FIG. 1. Dynamics of events number connected with CPS control rods sticking and exceeding of duration as to time of dropping at units of NPP in Ukraine.

TABLE I. AVERAGED DATA OF RESULTS ON TESTING OF CPS CONTROL RODS

Unit	Before start of the measures implementation in 1993-1995			During the measures implementation in 1995-1997			During fueling campaign in 1997-1998			During fueling campaign in 1998-1999		
	$t_{average}$	CPS control rod with t>4 s	Number of stuck CPS control rods.	$t_{average}$	CPS control rod with t>4 s	Number of stuck CPS control rods.	$t_{average}$	CPS control rod with t>4 s	Number of stuck CPS control rods.	$t_{average}$	CPS contro l rod with t>4 s	Number of stuck CPS control rods.
ZNPP-1	3.22	34	8	2.33	0	0	2.33	0	0	2.37	0	0
ZNPP -2	3.18	30	0	2.19	0	0	2.23	0	0	2.47	0	0
ZNPP -3	3.17	29	6	2.56	0	0	2.71	0	0	2.47	0	0
ZNPP -4	3.21	11	0	2.48	0	0	2.63	0	0	2.8	0	0
ZNPP -5	3.28	9	8	2.89	1	1	2.76	0	0	2.68	0	0
ZNPP-6	3.15	1	0	2.7	0	0	2.45	2	0	2.45	0	0
SUNPP-1	3.5	0	0	2.59	0	0	2.26	0	0	2.29	0	0
SUNPP–2	3.72	32	1	2.64	0	0	2.42	0	0	2.37	0	0
SUNPP-3	3.55	19	0	2.01	0	0	2.29	0	0	2.27	0	0
RNPP-4	3.46	35	5	2.83	0	0	2.54	2	0	2.26	0	0
KhNPP-1	3.72	12	0	2.85	0	0	2.17	1	0	2.31	0	0

Operator had developed the Technical Decision based upon the data accumulated during regular testing on possibility of CPS absorbers concerning changing of periodicity and order of measurements as to identification of CPS control rods dropping into core of WWER-1000 duration. However, causes of CPS control rods caused violation of safe operational conditions of 23.02.2000 and 01.06.2000 at unit 1 at Khmelnitsky NPP show that this issue is still topical, thus requiring the further investigation.

SSTC NRS performed expert assessment of this decision upon request of the RA. In-depth and comprehensive investigation of this issue resulted in development of the recommendations upon which base the RA obliged the operator to establish the correcting measures of necessary scope for each unit which undertaking will allow to ensure the designed indicators of CPS reliability during between-repairing period.

According to the recommendations made by OKB "Hydropress", it is necessary to go on with the undertaking of measures to exclude irregular time of CPS control rods dropping. At all the units complete upgrading of all the CPS drive stems, including increase of their weight along with improvement of UPT.

4. CONCLUSION

Under upgrading of the units built according to the old designs the problems can and do arise connected with changing of conditions for operation of equipment. It is not far always possible to predict their arising. Therefore, despite the conservative approach used under regulatory decisions making, the procedure of technical decisions licensing shall include possibility of new factors arising in the process of upgrading at any stage of the licensing. Namely, under performing of each move towards implementation of the decision the alternative paths to resolving of the problem shall be envisaged. Consequently, the existing normative base shall be improved along with the regulatory requirements taking into account the real preconditions stipulated by all the factors connected with operation of NPP.

As the above addressed examples show, upgrading of NPP in Ukraine is many-sided problem requiring comprehensive and in-depth investigation. The RA faces with the proper assessment of the decisions made in terms of their impact upon safety of NPP, well grounded bases, possibility and expediency of their implementation. Substantial role as to this assessment is played by SSTC NRS. Scientific support to the RA allows to take into account both the newest study results under making of decision and the experience gained during previous operation of units for many years.

THE REVIEW OF LWR OPERATING EXPERIENCE IN UKRAINE

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Abstract

Most probably that in Ukraine the WWER-1000 reactors will generate up to 93-96% of all NPPs electric power and about 40-50% of the total electric power production for the period of ten years (2000-2010). The operating experience of Ukrainian NPPs with WWER-1000 is 137 reactor — years. At the beginning of 1999 a total quantity of the fuel assemblies (FAs) discharged during all operational time of 11 reactors was ***5819*** (110 fuel cycles). Economical improvement is reached by increase of fuel burnup using some of FAs of 3 annual fuel cycles design in 4-th fuel loading cycle. The main problem of core operation of the last years have been consisted in incomplete rod control cluster assembly (RCCA) insertion. There were RCCA jammed at intermediate position or RCCA drop time was longer than required 4 sec. The compensatory measures realization has allowed decreasing the probability of incomplete RCCA operation. As a result of compensatory measures (excluding some cases) RCCA drop time problem was almost solved. Periodic measurements of RCCA drop time are not necessary. The cost, allowable time of operation, and possibility of inexpensive disposal are the main consumer features of RCCA. In order to increase RCCA lifetime it is required to replace the bottom (300-500 mm) n/α absorber B_4C by unswelling n/γ absorber (Hf, Dy_2TiO_5 or In + Cd + Ag) and to use cladding material that will be more stable to radiation embrittlement.

1. INTRODUCTION

In Ukraine there are fourteen water-cooled power reactors in operation (11 WWER-1000, 2 WWER-440 and 1 RBMK-1000) with a total installed capacity of 12880 MW(e). It makes 25% of the total installed capacity of the electric power stations. Four WWER-1000 units are under construction. In 1999 NPPs produced 72,072 × 10^9 kW.h of electricity (42,1% of the country electricity). The share of electricity produced by NPPs constantly increases because of the economic crisis and the fact, that electric power production by the NPPs is much cheaper than that of the thermal power plant station (Fig.1). It is most probable that the WWER-1000 reactors in Ukraine will have been generated up to 93-96 per cent of all NPPs electric power and about 40-50 per cent of the total electricity produced during 2000-2010.

For the electricity utility today operating an existing nuclear power plant, economic optimization means minimizing the costs of producing electricity, based on a high degree of plant reliability, operational flexibility and maximum fuel utilization. This will include the costs of spent fuel management.

2. EXPERIENCE OF FUEL OPERATION

2.1. Fuel Cycle Economical Improvement

The WWER reactor units operating now were designed for base-load operation in the common electric system of the former Soviet Union. Originally, according to the design, the WWER-1000 power reactors were operating in 2 annual fuel cycle using the fuel of 3.3% enrichment, the burnup limit of which is 38 MWd/kgU. By the year 1989 3 annual fuel cycle was calculated and supplied by fuel. According to the decision of the chief reactor designer , the reactors started operating in 3 annual cycle using the fuel of 3.6+4.4% and 4.4% enrichment. The design burnup limit of the fuel is up to 49 MWd/kgU. After the split up of the USSR and of its energy systems it became impossible to implement the design duration of the fuel cycles and to hold the planned refuellings in spring and summer periods. The average WWER-1000 FA burnup in 3 annual fuel cycle in Ukraine is essentially lower than designed. The fuel utilization is ineffective (Fig. 2 — burnup distribution in the unloaded FA).

It was decided to use some of FAs for the forth year of operation for the purpose of higher burnup, better fuel utilization, flexibility of core layout solutions that allows to vary the duration of the fuel cycles. Up to 1997 significant experience of the so-called experimental-industrial FAs operation in 4 annual fuel cycles was gained. It allowed the chief designer of the reactor and FAs to change design limits for the FAs:
– Total FAs operational time in the reactor was extended from 21000 up to 28000 hours;
– The allowed amount of FAs in the core, which are operated in the 4-th year is 36.

Figure 2 shows burnup distribution in the unloaded FAs. Analyzing fuel loadings of Ukrainian reactors it is possible to state, that the use of some FAs in 4 annual fuel cycles has allowed to increase unloaded FAs average burnup from 38–39 MWd/kgU up to 41,5 MWd/kgU. It leads to a reduction of the fuel cycle costs by 10%.

2.2. Fuel pin failure

The fuel reliability should be considered in the context with the purpose to increase fuel burnup. At the beginning of 1999 a total quantity of the FA discharged during all time of operation of 11 reactors was ***5819*** (110 fuel cycles). 194 of them were identified as leaking.

Basing on the data of Ukrainian WWER-1000 FA operation we can state, that the probability of FA leaking detection after first, second, third and fourth year of operation will be distributed as follows:
- after the first year of operation — 42/194 = 21,6%
- after the second year of operation — 65/194 = 33,5%
- after the third year of operation — 76/ 194 = 39,2%
- after the fourth year of operation — 11/ 194 = 5,7%.

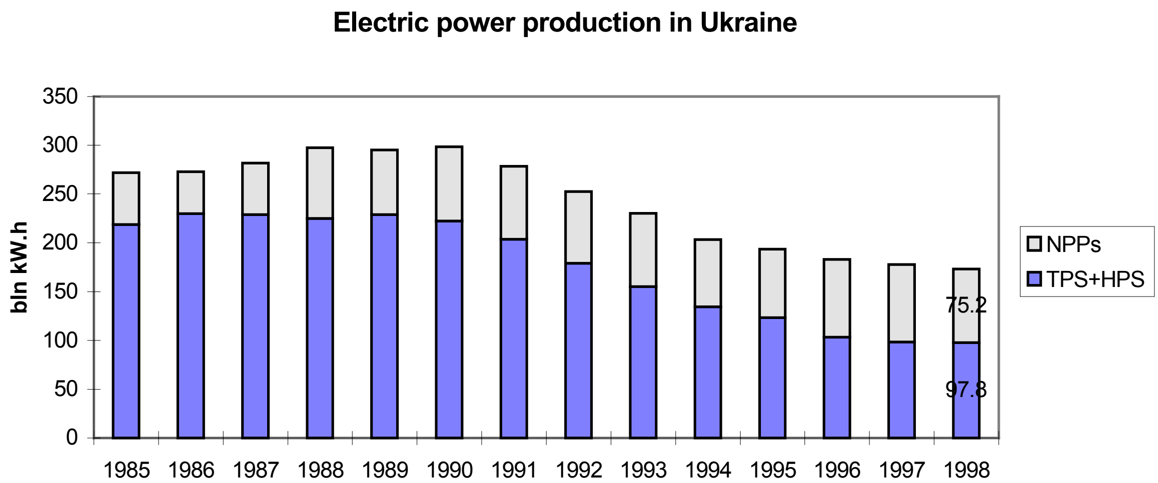

FIG. 1. Electric power production in Ukraine (TPS+HPS — Thermal and Hydro Power Plant).

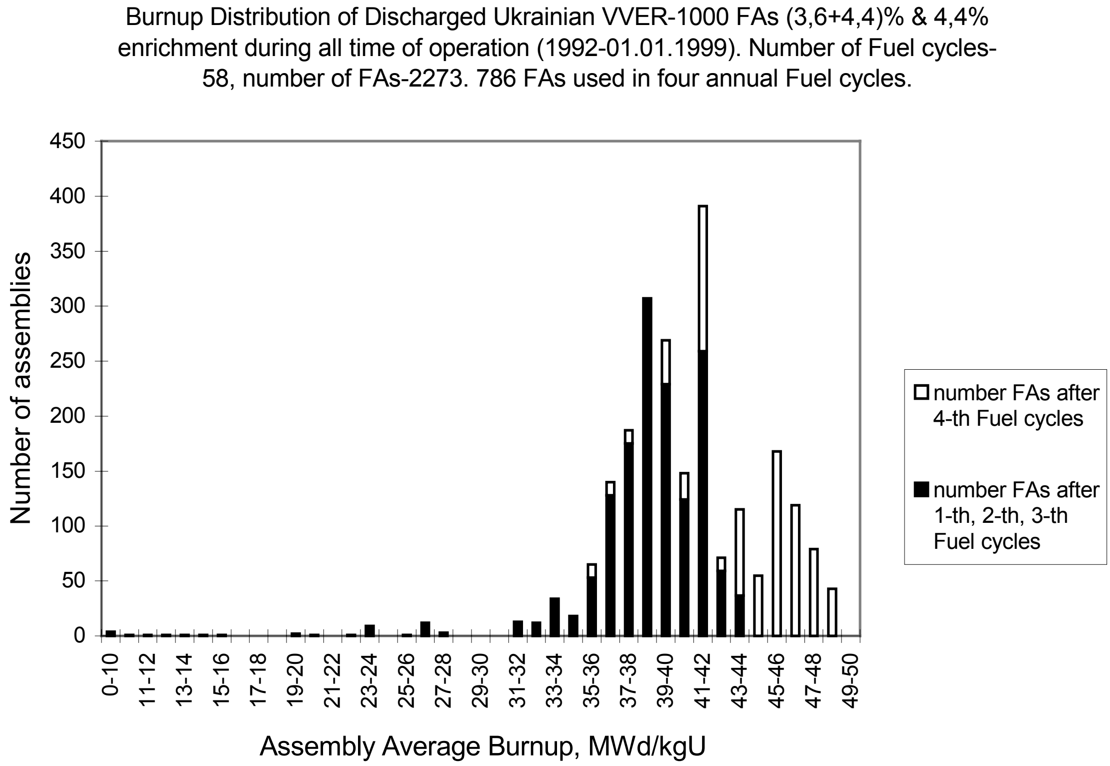

FIG. 2. FAs burnup distribution of Ukrainian WWER-1000.

However presented statistics is not correct for determination of fuel burnup influence on fuel pin failure rate, because the average share of tested FA after the first and second year of operation is 30-50%, after the third year of operation -60-80% and after the fourth year of operation -90–100%. (The rejection criterion of leaky FA is the value of iodine-131 activity $=1\cdot10^{-4}$ Ci/L in water of testing system.).

Detection of leaky spent fuel is put into practice under requirements of "Instruction for detection of the fuel rods tightness of WWER-1000 reactors (0401.00.00.000 DNG). Instruction was issued by Russian competent institutes, OKB Gidropress and Russian Research Center "Kurchatov Institute" respectively.
In the most of fuel cycles the primary circuit coolant activity was about (1-5)* 10^{-5} Ci/L iodine. The safety limit of $1.5*10^{-2}$ Ci/L iodine when the reactor must be shut down, has never been reached.

In 1998 the chief designer of the reactor and FAs changed (strengthened up) the safety limit of iodine primary coolant activity from $1.5*10^{-2}$ Ci/L iodine to $5*10^{-3}$ Ci/L iodine, and introduced new operational limit. The operational limit is equal to $1.0*10^{-3}$ Ci/L iodine. However the rejection criterion of leaky FA (I^{131} activity $\geq 1\cdot10^{-4}$ Ci/L in water of testing system) remains the same. This obliges the technical manager of NPP to use more conservative approach in FA failure determining and FA unloading ahead of schedule.

Fig. 3 show the time evolution of the average fuel pin failure rate (FPFR). The average FPFR is defined as follows:

FPFR = NFA/NA*312
NFA — the number of leaking FAs during a one year period.
NA — total quantity of FAs in reactors.
312 — number of pins in assembly. It is assumed that the leaking assembly has only one leaking pin.

Summarizing the presented data we can state that the increase of Ukrainian WWER-1000 fuel burnup has not worsened fuel reliability.

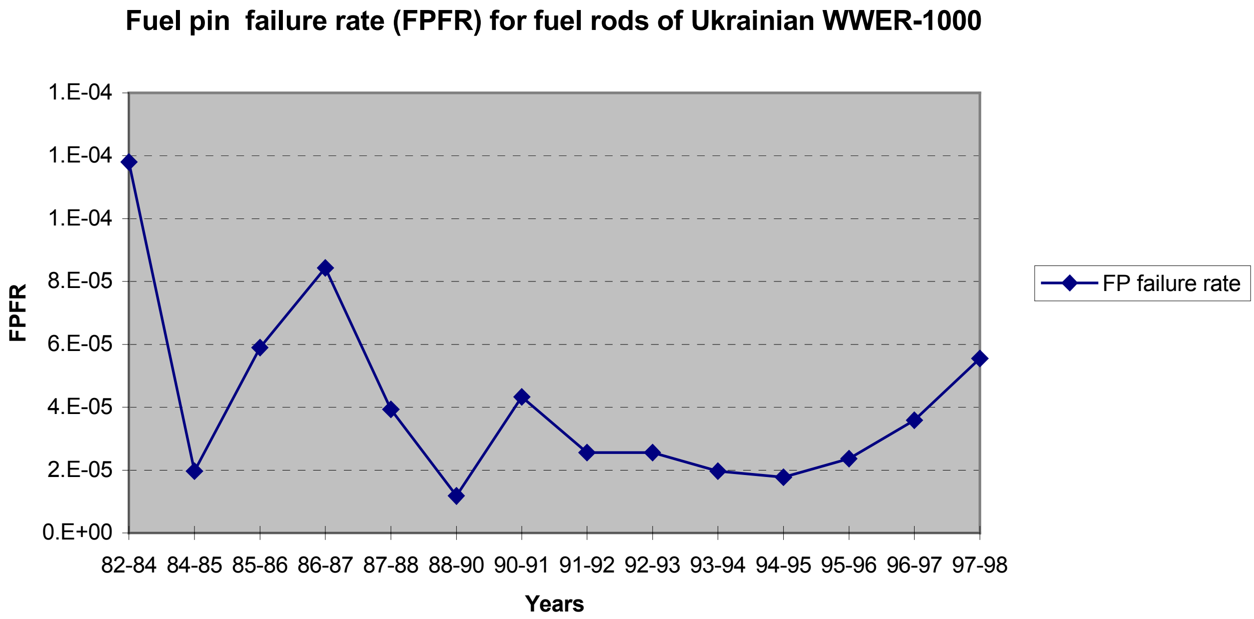

FIG. 3. Fuel pin failure rate for fuel of Ukrainian WWER-1000.

3. THE FURTHER FA IMPROVEMENT

It is necessary to implement advanced uranium- gadolinium (UO_2-Gd_2O_3) fuel assemblies with Zr alloy guide tubes (GT) and Zr alloy spacer grids (SG) for continuous improvements of WWER-1000 fuel utilization. It is expected to improve fuel assembly dimensional stability and to save uranium consumption by increasing uranium utilization.

3.1. The experimental industrial utilization of the advanced fuel

Ukrainian NPPs have been operating advanced fuel assemblies with Zr alloy-110 and Zr alloy-635 guide tubes (GT) and Zr alloy-110 spacer grids (SG) since 1995. According to the calculations the substitution of the steel by Zr alloy in materials of guide tubes and spacer grids increases the fuel utilization efficiency by 8.2%. Advanced fuel implementation allows to increase fuel burnup by 5–7%.

3.2. Results of the NPP site post irradiation inspection

During inspections of FAs with Zr GTs and SGs which have completed 3 cycles the displacement of SGs was revealed. In some FA the displacement was revealed after 2 fuel cycles. Structurally SGs were not reliably attached to GTs. Displacement of SGs happened in case where the fuel burnup exceeded 25–27 MWd/kgU.

4. SAFETY PROBLEMS OF REACTOR OPERATION UNDER INCOMPLETE RCCA INSERTION

In 1992 -1993 almost at all WWER-1000 of Ukraine, Russia and then in Bulgaria incomplete RCCA insertion (i.e. RCCA stuck in intermediate position), RCCA drop time exceeds 4s (design time) have been revealed. The program of additional quarterly measurements of RCCA drop time was developed and implemented. In the case of RCCA operation violation (and when there was no chance to cease one) the unit had been transferred to the operational mode with three loops coolant circulation and with preliminary power reduction to 67% of $N_{nominal}$.

The following units of Ukraine were transferred to the 3 loop operational mode:

- Zaporozhye NPP's unit 1; fuel cycle # 7 and 8; Generation loss-2,1* 10^9 kW hour;
- Zaporozhye NPP's unit 3; fuel cycle # 7, 8 and 9; Generation loss-3, 3* 10^9 kW hour;
- SU NPP's unit 2; fuel cycle # 7; Generation loss-1, 0* 10^9 kW hours.

In 1993 the chief reactor designer had calculated the neutron-physical and thermal hydraulic characteristics of reactor core for the RCCA drop time up to 10 sec, keeping in mind the case when all RCCA are inserted except the most effective RCCA. It was shown that requirements of regulatory safety documents were met. However, the results obtained were not used to cancel operational restrictions according to a conservative principle.

Fuel Assembly (FA) Bow is the main reason of incomplete RCCA insertion during reactor core operation.

In order to reduce probability of incomplete RCCA insertion and in order to decrease size of FA bow, to ensure the reactor's safe operation the following compensatory measures have been performed during the scheduled repairs in 1993–1997:

- Modification upgrading of the bundle safety tubes (BST) in order to correct BST position and to correct axial compression of FA;
- Drilling of RCCA drivers bars in order to reduce force of hydrodynamics friction during the input of RCCA in the reactor core;
- Utilization of the new designed heavier RCCA with gafnium or titanat of dysprosium in Rovno NPP and Zaporozye NPP and RCCA driver bar with increased dead load;
- Organizing the core loading pattern with advanced FA with zirconium (Zr-Nb and Zr-Nb-Sn-Fe alloy) guide thimbles and grids and with increased (comparatively with initially designed) FA head spring gain.

It has allowed to decrease the probability of incomplete RCCA operation and allows us to reduce FA bow in reactor core step-by-step. On Figure 4 (the example of a Z NPP-2) the curves of RCCA drop time change for the period between 1993 and December 1999 are presented.

As a result of compensatory measures (excluding some cases) RCCA drop time problem was almost solved (Table 1). Periodic measurements of RCCA drop time are not necessary. Indicated measurements are to be conducted at the moments of non-scheduled reactor shutdown as well as during the reloading activities.

TABLE I

Operating characteristic	Before the implementation of compensatory measures	After the implementation of measures
average RCCA drop time ($\tau_{average}$)	3,3s	2,49s
number of RCCA with drop time $\tau > 4s$	200	14
number of stuck (jammed) RCCA	28	0

Currently it is proved, that the increase of fuel burnup under compensatory technical measures implementation does not worsen FA bow situation. Analysing the WWER-1000 and PWR operational experience we may state that, at present we possess a more detailed and complete information on evolution of a FA bow and measurement of RCCA insertion for WWER-1000 than for PWR.[1,2,3]

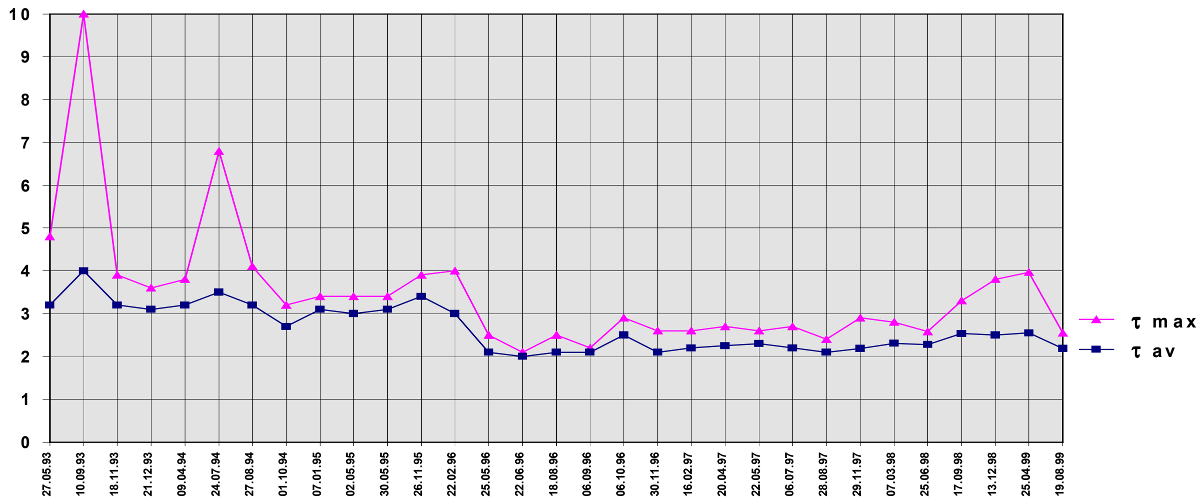

FIG. 4. Measurement of RCCA drop time at Zaporozhe-2.

5. EXPERIENCE OF CR AND RCCA OPERATION IN UKRAINIAN WWER-1000

The RCCAs of a reactor control protection system (CPS) are the most important elements of the reactor maintenance safe operation, which ensure control of reactor core power level and fast reactor core transfer from initial condition to sub critical condition during the accident. Share of RCCA in fuel reloading cost is 1–1,2%. However for CR materials choice it is necessary to take into account a possibility of inexpensive RCCA recovery or disposal.

5.1. The selection of RCCA design and materials

In the former USSR the Moscow manufacturing plant of polymetals (MPP) was the designer and manufacturer of CR for almost all reactor types. The carbide of boron (B_4C with a natural mixture of isotopes) was selected as an absorber material because of being accessible, cheap and quite effective material. The stainless steel 06 X18 N10T (C < 0,06%, Cr-18%, Ni-10%, Ti < 1,0%) was selected as a CR clad material.

5.2. The first result of improved RCCA based on combination absorber — Hf-B_4C basis operation

The cost, allowable time of operation, and possibility of inexpensive disposal are the main consumer features of RCCA. In order to increase RCCA lifetime it is required to replace the bottom (300–500 mm) n/α absorber B_4C by unswelling n/γ absorber (Hf, Dy_2TiO_5 or In + Cd + Ag) and to use cladding material that will be more stable against radiation embrittlement (according to the researches, as the calculation data of the thermal neutrons flux density above reactor core has appeared to be below actual).

12 RCCAs with combined Hf-B_4C absorber had been designed, manufactured, accepted by the interdepartmental commission, and in 1997 were loaded in WWER-1000 of Rovno NPP for the operation on the joint decision of GOSKOMATOM and with participation of Kurchatov institute, RIAR and chief reactor designer — OKB GP. Cr-Ni alloy EP-630Y (Cr-42%, Ni-56%, Mo-1%), which has high corrosion stability (resistance) in a water environment and has high plasticity during a long period of irradiation, was used as a cladding material.[4]

Differential and integral RCCA worth characteristics were measured at the time of RCCA insertion by using a reactivity meter. RCCA movement were made step by step from upper position (H = 100%) down to lower position (H = 0%) by step of 5%.

The results allow to conclude:
- Integral worth characteristics of RCCA with Hf and of RCCA without Hf are almost the same;
- At the moment of 20% insertion integral efficiency of lower part of RCCA with Hf was 24% less than that of ordinary RCCA;
- At the moment of 10% insertion efficiency of RCCA with Hf was equal to 0,53-0,58 against the ordinary RCCA efficiency (calculated value was 0,8);
- At the moment of reactor scram total RCCA worth was equal to 8,7% and exceeded the calculated value (7,5%).

In order to clarify the reasons of low Hf efficiency in existing RCCA it's useful to perform calculation of:
- Burn up effect (neutron spectrum influence),
- Research geometry effect on RCCA lower part efficiency,
- Research RCCA lower part position (above reactor core) effect on efficiency
- Safety analysis of RCCA with Hf as CG.

When RCCAs with Hf are used as SG all safety requirements are met.

It is necessary to note, that for last 15–20 years were also other problems, which were solved or are solved now.

The steam generator (SG) collector was found to be the weakest element of the reactor coolant boundary. There were 15 SGs replaced at Zaporozhe NPP and South-Ukraine NPP for the 1988–1992 period. One of them was repaired at the plant and not replaced[5,6].

The SG damages where caused by the cracks formation. In 16 WWER-1000 SGs the cracks where developed in cold collectors during operation. In three cases damages were indicated by increasing of the steam generator water activity. In the rest of cases the damages were detected visually during scheduled inspection.

The Instrumentation and Control System (I&C) projects were created in late 60-ths and implemented in 70-ths, so they have become outdated already. The available I&C which have been working at Ukrainian NPPs don't provide information display. They haven't interface designed for peripheral equipment switching on.

On some NPP units I&C have exhausted its term resources (have been working longer than projected lifetime). Such Systems have 20 years project lifetime but some of their separate components haven't more than 10 years (ionizing chambers, neutron detectors, pressure and temperature transducers, etc.). The Control and Protection Systems (CPS) exhausted their term resources up as well.

The new project of improved I&C was developed by Khartron Co. (Kharkov, Ukraine). There are detectors, instruments, gears and other equipment all necessary for project realisation, produced by industry of Ukraine, Russia and by some other countries as well. The pilot project is being implemented at Zaporizhia NPP nowadays. Also the Safety Parameter Display System is being introduced at all Ukrainian NPPs with participation of Khartron Co. and Westinghouse Electric Co.

The I&C upgrading plan has made up and its gradual implementation began.

6. CONCLUSIONS

The main problem of core operation of the last years consisted in incomplete RCCA insertion because of FAs bow in reactor core. As a result of compensatory measures (excluding some cases) RCCA drop time problem was almost solved. Periodic measurements of RCCA drop time are not necessary. Indicated measurements are to be conducted at the moments of non-scheduled reactor shutdown as well as during the reloading activities. Implementation of the given technical measures allows to reduce FA bow in reactor core.

The use of a hafnium as the material for the control rods (CR) under appropriate price policy can give certain economic advantages to Ukrainian NPPs. However additional researches are needed to confirm the specific CR physical characteristics and reliability.

The experience of WWER-1000 FA operation and principal results of the irradiated FAs examination allow to accept the possibility of further fuel burnup increase. Fuel reliability is satisfactory.

The further improvement of FAs is necessary. That will allow to reduce the front- end fuel cycle cost (specific natural uranium expenditure), to reduce amount of spent fuel and, consequently, the fuel cycle back- end costs, and to increase the fuel burnup.

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DESIGN FEATURES AND TECHNICAL MEANS FOR IMPROVING PERFORMANCE, ECONOMICS AND SAFETY

(Session I-b)

Chairperson

N.S. FIL
Russian Federation

IMPROVEMENTS OF THE NUCLEAR POWER PLANT ISAR 1

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Abstract

This paper reviews the improvements which have been made in the nuclear plant Isar-1 since it was first connected to the grid in 1977, and discusses the resultant performance of the plant.

1. SUMMARY DESCRIPTION OF NUCLEAR POWER PLANT ISAR 1

The main data of the Isar 1 plant are presented in Table I.

2. ATOMIC LICENSES FOR ISAR 1

Table II shows the history of the different licenses for the erection and operation of the Isar 1 plant. Due to recommendations of the Reactor Safety Commission (RSK) the operation of the plant had to be stopped after the fulfillment of license No. 7 to perform backfitting measures corresponding to license No. 8. This situation was untypical in our company.

3. SUMMARY OF THE MOST IMPORTANT IMPROVEMENTS

The most important improvements with respect to safety and operation are listed in Table III according to the corresponding licenses.

3.1. Improvements of systems

Several improvements have been made in existing systems. Table IV lists these improvements, and provides the reasons for making these improvements. One of the most important improvements was the removal of an auxiliary system for the main recirculation pumps.

3.2. Additional systems

Additional systems, which didn't exist in the initial plant design, have been added to the plant. These are listed in Table V, along with the reasons for adding these systems. The filtered pressure relief system of the containment was a recommendation of the RSK due to the Chernobyl accident.

3.3. Basic safety of tubes and systems

Additional backfitting measures have been made to improve the safety of tubes and systems (see Table VI). Some of these backfitting measures led to a long outage in 1994 of nearly 3 months especially for the installation of new steam and feedwater tubes of the primary coolant system.

TABLE I. ISAR 1 PLANT — MAIN DATA

- Boiling water reactor, type 69
- Design and general construction contractor: Siemens/KWU
- Owners: Bayernwerk AG (now E.ON Energie AG): 50 %
  Isar-Amperwerke AG: 50 %
- Operator: Kernkraftwerk Isar 1 GmbH (until end of 1997)
- Gross capacity: 912 MWel
- Start of erection: 16.5.1972
- First criticality: 20.11.1977
- First synchronisation with the grid: 3.12.1977
- Take over of the plant from Siemens/KWU by the Isar 1 GmbH: 21.3.1979
- Containment: pressure suppression system
- Number of fuel elements in the core: 592
- Main recirculation pumps in the pressure vessel: 8
- Main steam lines: 4
- Main feed water lines: 4

TABLE II. ATOMIC LICENSES FOR ERECTION AND UNLIMITED OPERATION

Number of license	Title	Application	Date of license
Site	Choice of the site	21.12.1970	25.11.1971
1. license	Erection of the plant	25.06.1971	16.05.1972
2. license	Handling of fuel elements and radioactive materials	26.09.1975	31.05.1976
3. license	First loading of the pressure vessel with fuel elements and neutron sources	15.11.1976	22.07.1977
4. license	Nuclear commissioning and operation until the first outage	14.01.1977	18.11.1977
5. license	Increase of the capacity of the fuel storage installation	20.09.1978	04.03.1980
6. license	Performance of the first fuel change	31.07.1979	15.04.1980
7. license	Further operation	31.07.1979	10.07.1980
8. license	Performance of backfitting measures	17.11.1980	31.08.1981
9. license	New nuclear commissioning and unlimited operation	17.12.1980	28.07.1982

TABLE III. ATOMIC LICENSES FOR IMPROVEMENTS OR CHANGES

Number of license	Title	Application	Date of license
1. license	Reduction of radioactive releases to air and water	02.11.1987	11.01.1988
2. license	Realization of measures for beyond design accidents	29.10.1987	11.07.1988
3. license	Backfitting of the crane in the reactor building	20.02.1989	20.07.1989
4. license	Backfitting of an additional pressure relief system for the pressure vessel	19.04.1990	25.07.1990
5. license	Use of uranium fuel elements with 10 × 10 rods	25.07.1990	26.06.1991
6. license	Change of steam and other tubes	31.08.1992	23.03.1994
7. license	Installation of a new process computer	02.04.1991	14.07.1994
	Table III. cont.		
8. license	Improvements of the feedwater- and emergency core cooling system	10.05.1994	27.10.1994
9. license	Use of uranium fuel elements with 10 × 10 rods (ATRIUM)	18.03.1994	29.05.1995
10. license	Use of uranium fuel elements with 10 × 10 rods (SVEA 96)	15.04.1996	29.04.1997
11. license	Joining of the Bayernwerk Kernenergie GmbH to the licenses	27.08.1997	24.10.1997
12. license	Common security installations for the plants Isar 1 and Isar 2	01.07.1997	11.09.1998
13. license	Joining of the Preussen Elektra Kernkraft GmbH to the licenses	17.04.2000	09.06.2000

3.4. Change or improvement of valves

Due to different reasons a lot of valves had to be changed or improved as can be seen in Table VII.

3.5. Improvement of fire protection installations

In the initial layout of the plant, fire protection installations did not have the standard corresponding to later regulations and recommendations of the safety authorities. Therefore a lot of improvements in nearly all parts of the plant had to be realized. These are listed in Table VIII.

TABLE IV. IMPROVEMENTS OF SYSTEMS

Improvement of systems	Reason for improvement	Year of realization
Cooling system for normal operation	Avoidance of possible spreading of radioaktive material out of the controlled area	1982
Change of different tubes and valves within the turbine hall	Better material, reduction of iron entry into the reactor pressure vessel, avoidance of corrosion	1981, 1982
Change of the process computer	Better supervision of the plant, new computer generation	1990–1994
Remove of an auxiliary system for the main recirculation pumps	Increase of safety, remove of faulty welds, reduction of radioactive doses of the staff	1993
Installation of a third grid connection	Optimization of the electrical supply by a nearby hydro power plant	1997–2000
Improvement of I and C components against design accidents	Deficiencies of the design	continuous
Improvement of lightning protection equipment	Deficiencies of the design	continuous

TABLE V. ADDITIONAL SYSTEMS

Additional systems	Reason for improvement	Year of realization
Installation of a recombiner system for hydrogen	Avoidance of an explosible mixture in the containment	1982
Filtered pressure relief system of the containment	Reduction of the risk of a hypothetical failure of the containment due to overpressure	1988
Filter and overpressure in the main control room	Protection of the main control room against consequences of hypothetical beyond design accidents	
Additional pressure limitation system of the reactor pressure vessel	Increase of the availability of the pressure relief installations	1994

3.6. Improvements of components

Due to different reasons components corresponding to the initial layout had to be improved. A list of improved components and the reasons for making these improvements is given in Table IX.

TABLE VI. BASIC SAFETY OF TUBES AND SYSTEMS

Basic safety of tubes and systems	Reason for improvement	Year of realization
Change of high pressure heater and steam dryer/reheater in the turbine hall	Material with basic safety, better possibility for periodic tests	1982
Change of main steam tubes, feed water tubes and valves within the containment, in the reactor building and in parts of the turbine hall	Material with basic safety, better possibility for periodic tests	1982–1999
Partly change of the core flooding system	Material with basic safety, better possibility for periodic tests	1994

TABLE VII. CHANGE OR IMPROVEMENT OF VALVES

Change or improvement of valves	Reason for improvement	Year of realization
About 350 valves with safety function	Optimization, resistance against design accidents, other materials etc.	1994–2000

TABLE VIII. IMPROVEMENT OF FIRE PROTECTION INSTALLATIONS

Fire protection installation	Reason for improvement	Year of realization
System for inertisation of the containment	Avoidance of fire within the containment	1988
Installation of fire extinguishing installations	Deficiencies of the layout	1992–1998
Improvement of fire walls	Deficiencies of the layout	
Improvement of ventilation systems, ventilation flaps	Deficiencies of the layout	
Replacement of the former fire detection installation by a more effective installation	Deficiencies of the layout	
Replacement of asbestos	Stronger regulations	1992–1998

TABLE IX. IMPROVEMENTS OF COMPONENTS

Component	Reason for improvement	Year of realization
Protective casing of the high pressure casks of the controll rod installation	Avoidance of hight pressure under the reactor pressure vessel in the case of a failure of the casks	1981/82
Backfitting of the main crane in the reactor building	Avoidance of the fall of loads	1989
Backfitting of the refueling maschine	Control of single failure	1996

TABLE X. IMPROVEMENTS OF SECURITY INSTALLATIONS

Security installation	Reason for improvement	Year of realization
Improvement of identification	Deficiencies of the layout	1997
Installation of a security control room	Not planned in the layout	1997
Improvement of the supervision and protection of the outer fence of the site of Isar 1 and Isar 2	Deficiencies of the layout, joining of the separate plants Isar 1 and Isar 2 to a common organization	1999

3.7. Improvements of security installations

Especially due to deficiencies of the initial layout and new recommendations of the authority, improvements of several installations had to be realized. These are summarized in Table X.

3.8. Improvement of operation

Computer-based maintenance systems are now installed in all nuclear and conventional power plants (without hydro power plants) of the former Bayernwerk AG, as summarized in Table XI.

3.9. Installation of a full scope simulator

In the last years the tendency in Germany was that a full-scope simulator should be well fitted to the layout of the control room of the plant. Therefore a new simulator for the plants Isar 1 and Philippsburg 1 was installed in the Simulator Center in Essen. For the plant Isar 1 also the control room had been reconstructed.

TABLE XI. IMPROVEMENTS OF OPERATION

Improvement	Reason for improvement	Year of realization
Change of the turbine control system	Adaption to modern technology, difficulties with spare parts, high service costs of the system	1994
Installation of a computer-supported maintenance system together with Isar 2	Adaption to modern technology, improvement of maintenance and documentation	1997/2000

3.10. Further improvements

Some further improvements, including some which are currently underway, which shall be finished within the next 2 years are shown in Table XII.

4. PERIODIC SAFETY ANALYSIS

Figures 1-4 show the most important aspects of this analysis which was the pilot project for all nuclear power plants in Germany.

TABLE XII. FURTHER IMPROVEMENTS

Improvement	Reason for improvement	Year of realization
Supervision installation of the pressure relief system of the containment	Demand of the authority	2000
Improvement of the supervision system of the core	Recommendation of the periodic safety analysis	2000
Improvement of the filter installation in the ventilation system of the reactor building and the turbine hall	Adaption of modern technology	2000
Sample system of the containment	Demand of the authority	2001/2002

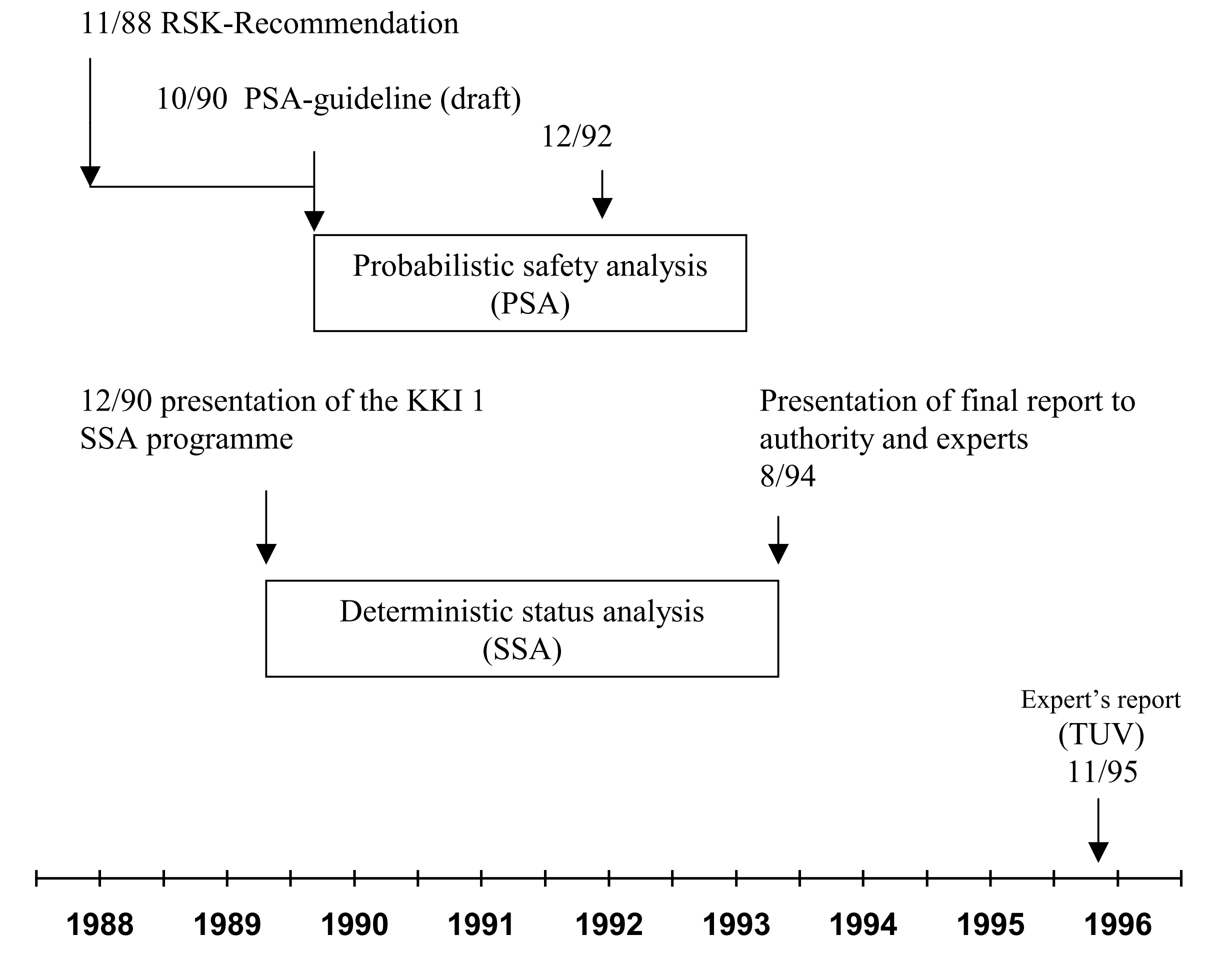

FIG. 1. Schedule: Periodic Safety Analysis for NPP Isar 1.

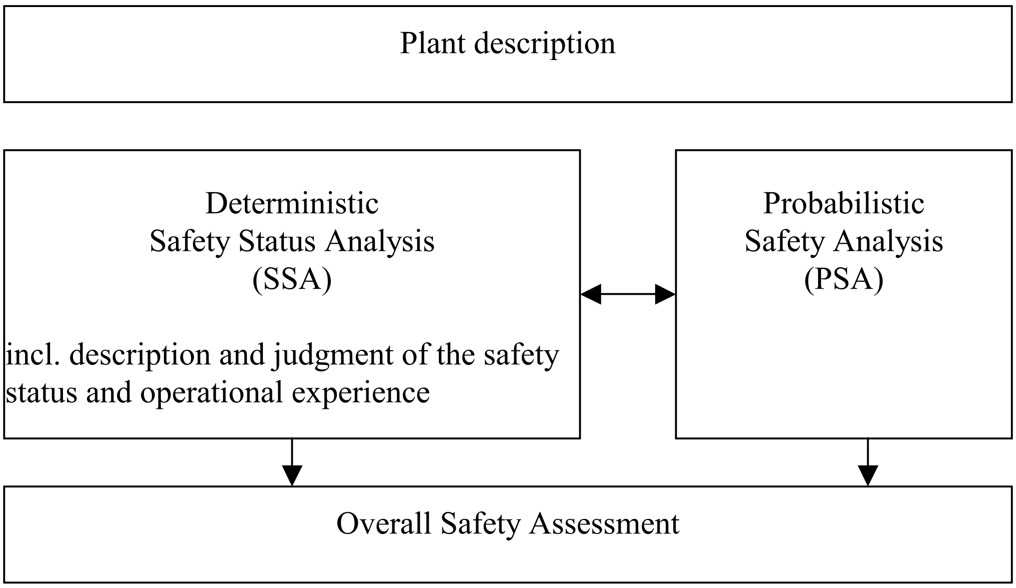

FIG. 2. Elements of the periodic safety analysis for NPP Isar 1.

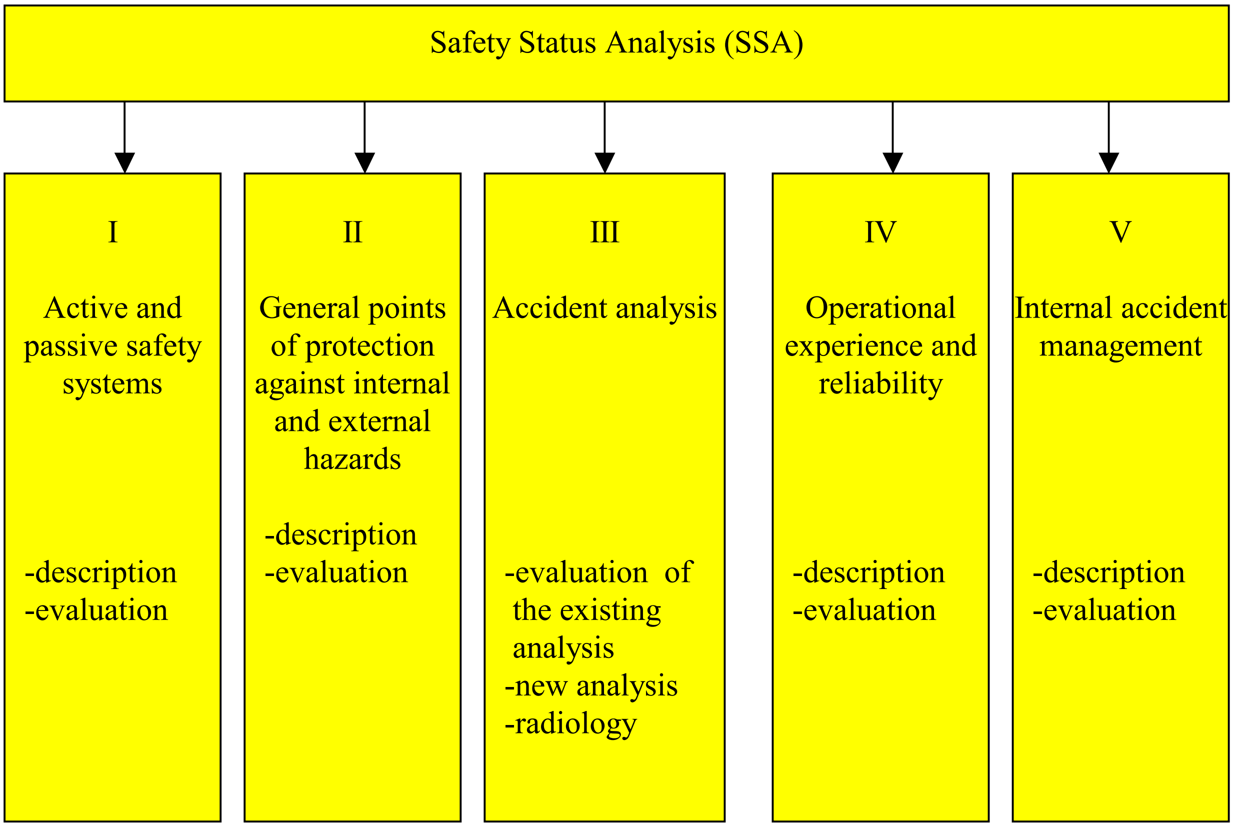

FIG. 3. Structure of safety status analysis for NPP Isar 1.

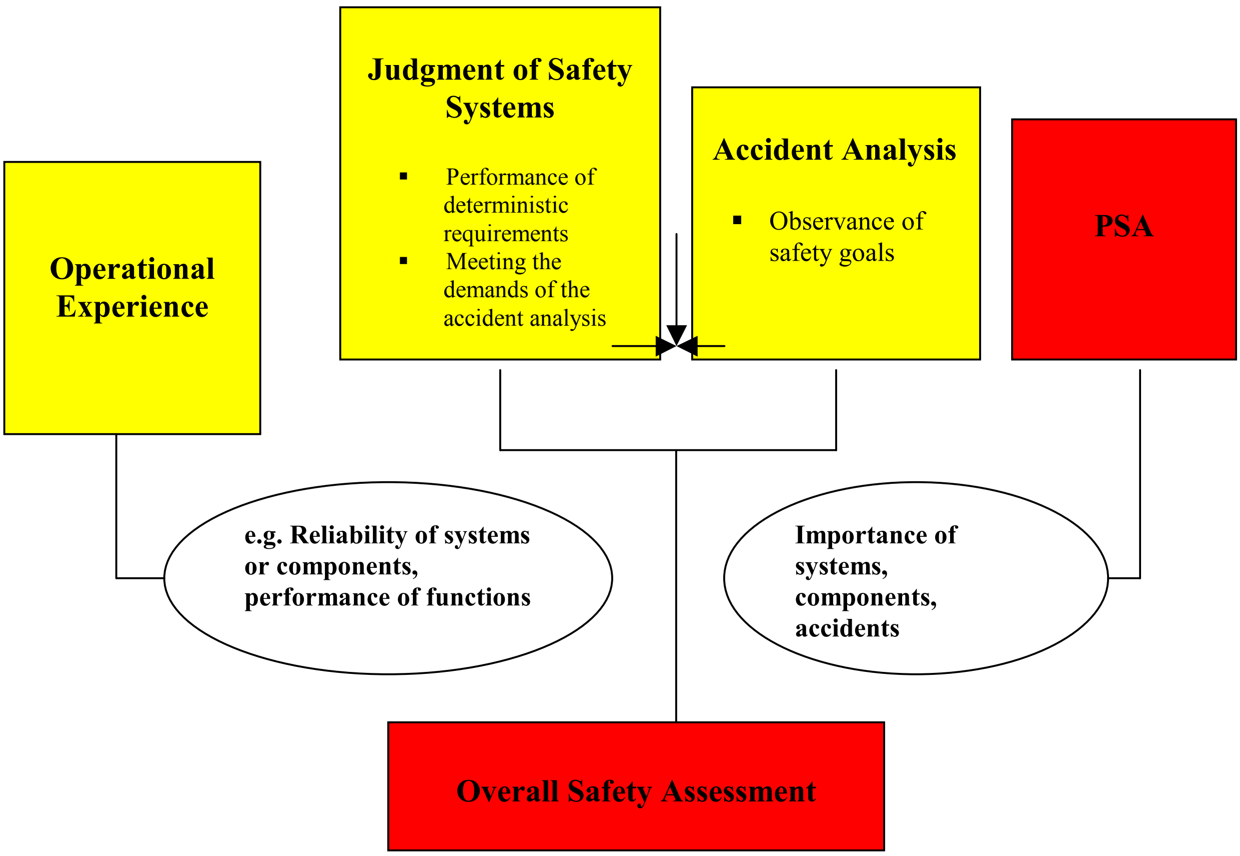

FIG. 4. General evaluation procedure applied for NPP Isar 1.

5. OPERATING RESULTS

The overall effect of the extensive modernization programme described above is the very good performance (e.g. availability and capacity factor) of the Isar-1 plant, as is shown graphically by the operating results (Figure 5) since 1995.

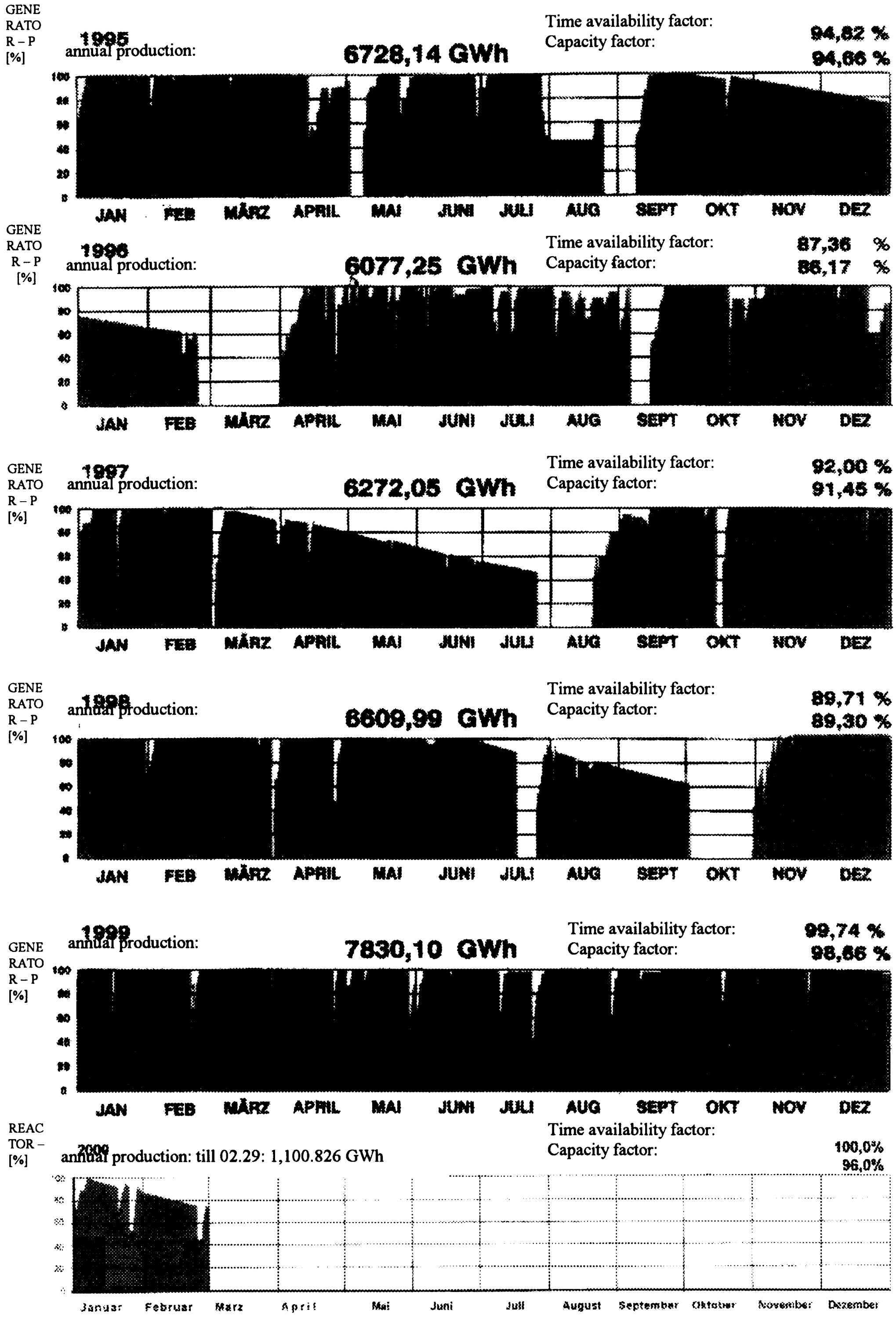

FIG. 5. KKI 1 — Operating results (1995–2000).

SAFETY ANALYSES SUPPORTING THE SYMPTOM ORIENTED EMERGENCY OPERATING PROCEDURES

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Abstract

New Emergency Operating Procedures were made for the Paks NPP by the leading of Westinghouse Electric Co. The Strategy Report of the optimal and functional recovery guidelines were made in the first stage of the work. During preparation of the Strategy Report safety analyses were defined the completion of which are essential for making the developed strategy definitive and for establishment of it. These analyses were carried out by the Atomic Energy Research Institute (KFKI-AEKI). In this paper 11 main topics are specified (list of prepared analyses in particular see in Appendix 4) which were essential for the development and working out of procedures according to the Strategic Report. During the specification of analyses the concrete guidelines supported by the presented analysis were always given; the task was also interpreted and those preliminary considerations and assumption were introduced which were used for making analyses.

1. GENERAL ORGANISATION OF THE EOPS

The EOPs are dealing with those events that actuate or require reactor trip or ESFAS operation. These events can further be categorised based on their features such as

- events could be diagnosed based on symptoms (i.e. plant parameters)
- events could readily not amenable to diagnose on the basis of assumptions.

Corresponding to these two different types of emergency situations there are also two different types of procedures: event-dependent and event-independent part (see Appendix 1).

2.1. Event-dependent part

They are those procedures addressing emergency situations that can be readily diagnosed. These procedures aim at the optimal recovery for the plant for specifically diagnosed events: Optimal Recovery Guidelines (ORGs).

The purpose of Optimal Recovery Guidelines is to bring the plant to the optimal end-status for all events covered. The optimal recovery concept is based on the premise that radiation release and equipment damage can both minimised through associating the symptoms of an emergency transient with a predefined plant condition and implementing an associated predefined event-related recovery strategy to achieve an optimal end status. The optimal end state varies with emergency transient but is the state in which radiation release and equipment damage are minimised and plant conditions are stable with plant equipment operating in long term alignments.

The ORGs are entered when the reactor is tripped or the Emergency Core Cooling System is actuated. An immediate verification of the automatic protection signals is performed and the accident diagnosis process is initiated. When the nature of the accident is identified, the operator is directed to the applicable recovery procedure.

Three levels of diagnosis are built in the ORGs: early diagnosis (E-0), continuous and rediagnosis. Guidelines are provided to the adequate procedure if an error in diagnosis process is identified.

The ORGs are addressed to 4 major event categories (see Appendix 2):
- Loss of primary or secondary coolant
- Faulted steam generator
- Steam generator tube rupture
- Loss of all AC power

They provide recovery instructions for the „classical" accidents and for those cases when they are combined with the equipment failures or other unexpected events.

2.2. Event-independent part

They are those procedures addressing emergency situations that can't be readily diagnosed. These procedures will be based on monitoring as well as restoring plant safety: Function Restoration Guidelines (FRGs)

a) Critical Safety Functions Monitoring

Early in the course of the accident, one has to initiate monitoring of the Critical Safety Functions. These CFSs are defined as a set of functions ensuring the integrity of the physical barriers against the releases of radioactive material:
- Fuel matrix/cladding
- Primary System Boundary
- Confinement Building

Monitoring of these functions is performed continuously through a cyclic application of the Status Trees:
- **S**ubcriticality
- **C**ore Cooling
- **H**eat Sink
- **P**rimary Integrity
- **C**onfinement Integrity
- **I**nventory

b) The Function Restoration Guidelines (FRGs)

The objective of the event-independent part of the Emergency Operating Procedures (EOPs) is to provide means to evaluate and restore the plant nuclear safety. The concept is based on the premise that radiation release to the environment can be minimised if the barriers to activity release are protected (barriers of defence in depth). In order to accomplish this goal, a set of functions has been defined which are critical from the plant nuclear safety point of view. These are the Critical Safety Functions. To be able to evaluate the status of these functions, Status Trees have been designed, one per CSF. Once the state of the CSF is evaluated, based on their state and the rules of priority one can designate a Function Restoration Guideline to be implemented for restoring CSF (see Appendix 3).

The Function Restoration Guidelines are entered when the Critical Safety Function monitoring identifies a challenge to one of the functions. Depending on the severity of the challenge, the transfer to a Function Restoration Guideline can be immediate for a severe challenge or delayed for a minor challenge. These guidelines are independent of the scenario of the accident, but only based on plant parameters and equipment availability.

3. PREPARED ANALYSES

There were carried out about 50 RELAP cases. In the following the results of required and prepared analyses and main conclusions are presented.

3.1. Evaluation of the use of main loop isolation valves during accidents/Isolation of primary leak by main loop isolation valves closure

In the current Paks procedure dealing with primary break accident, credit is given to the closure of the main loop isolation valves. The following considerations apply:

One of the fundamental basic rules of the EOPs is to not contradict nor violate the design bases of the plant. The loop isolation using the loop isolation valves has not been considered in the Paks accident analyses nor in the Final Safety Report, and as such, is not documented as a plant design basis. There were investigated some LOCA cases - medium and small sizes - with MLIV closure (see analysis No. 1. in Appendix 4). After MLIV closure on the broken loop RCS was filled up by 3 HPIS injection and there is less than 5 minutes to avoid the opening of PRZR relief valve. In conclusion, it was decided not to close MLIV if any of the HPIS pumps are in operation.

3.2. Upper plenum coolability during natural circulation conditions

Some analyses were made to determine the appropriate cooldown rate during natural circulation without steam void in vessel (see analysis No. 4. in Appendix 4). There were calculated NC cooldown with 20°C/h and 14 °C/h rate. In conclusion, the 20°C/h rate was too fast to assure RCS cooldown without void formation in the upper head, but 14°C/h was found appropriate.

The possible void formation in the SG primary collectors under natural circulation cooldown conditions was investigated. This phenomenon could be suppressed by using the degasifier lines.

To calculate the boron concentration in the primary system is necessary to compensate boron acid for the non-boration of the "passive parts" (pressurizer, reactor vessel head, SG collectors). Optimum boron mixing were calculate in the analysis.

Based on the transient analyses (see results of analysis No. 4. in Appendix 4) it was concluded that the 30°C/h cooldown rate with 5 t/h upper head injection is appropriate to assure RCS cooldown without void formation in the upper head, but not in the steam generators. The Plant Engineering Staff should evaluated whether using EVS (and discharging the primary coolant into confinement) or waiting some hours is required.

3.3. Malfunction of high pressure injection system at full power

The specific scenario of a spurious start of HPIS needs particular attention. In such case, the three trains would start injecting because of modifications in the new reactor protection system. Due to the flow from the three HPIS pumps, the level will increase in the pressurizer, thereby increasing the pressure, which is expected to result in actuation of the pressurizer spray. Based on the system characteristics (especially HPIS pumps capacity), there could be a potential for reaching the pressurizer solid condition (or at least spray nozzle flooding) and also of dead-heading for the HPIS pumps.

An analysis for a spurious HPIS actuation was made (see analysis No. 5. in Appendix 4). It was anyway be aimed at terminating the HPIS flow while protecting the RCS from the overpressure. Results show that primary pressure increasing fast caused by 3 HPIS injection and after 5 minutes PRZR relief valve opens (analysis was made with old-type-valve with opening setpoint 136 bar). After 15 minutes PRZR was filled up. In conclusion, on the units with old-type-valve the operator should intervene within 15 minutes. The new relief valves opening setpoint is 138 bar (this pressure is the end of HPIS injection) so the problem is eliminated.

3.4. Cooldown without letdown during natural circulation

Best estimate analysis could justify the strategy implemented in this procedure (see analysis No. 6. in Appendix 4) In conclusion, the primary boron concentration during the 14°C/h natural circulation cooldown without letdown, is sufficient to keep the core in safe shutdown conditions.

3.5. Review of criteria for safety injection termination

Guideline ES-1.1 safety injection termination criteria do not verify boron concentration. Paks has verified that safety injection termination criteria need not be modified to include boron concentration verification (see analysis No. 6. in Appendix 4). From these results it can be concluded that in controlled cooldown cases, when the PRZR level is maintained by make-up pump injecting coolant with 40 g/kg boron concentration, there is no danger of recriticality and safe shutdown core conditions are provided.

There was investigated another - more fast cooldown - streamline break transient (see analysis No. 8. in Appendix 4) whether subcriticality can be maintained also in this case. The calculation results indicate that HPIS injection could be terminated and there is no danger of recriticality. Due to the high amount of boron injected by 3 HPIS pumps the primary boron concentration increases very fast ensuring high margin when HPIS pumps can be stopped by the operator. Here should be noted, that there is no analysis for only the 1 HPIS injection (conservative accident analysis).

3.6. Total loss of power

There were investigated cases with or without cooldown by secondary side and with or without loss of primary cooling 5 t/h flow on each 6 MCP sealing (see analysis No. 10 and 11. in Appendix 4). In the cases without primary leakage the time available for recovery - even with secondary depressurisation - exceeds 10 hours. This time reduces to less than 5 hours without secondary bleed, if the pump seals are defected. In conclusion, the secondary bleed should be initiated if only a loss of primary coolant also exists. Otherwise the secondary side depressurisation can lead to shorter available time for recovery.

3.7. Main steam header rupture

The existing analysis for the uncontrolled depressurisation of all steam generators is based on a scenario where neider main steam header separation nor steam generator isolation occurs, enabling all steam generators to blow down through the ruptured main steam line (i.e. the isolation valve on the ruptured line fails to close), however the faulted SG is automatically isolated on feed side by the logic. For this scenario, the analysis was finalised to include safety injection termination criterion for boron concentration.

Compensation of shrinkage was investigated in the analysis No. 8. (see Appendix 4) to determine the time of possible feedwater de-isolation by the operator, when the RCS cooldown rate falls below the 60°C/h limit value. This occurs at 19 minutes when the RCPs are running and at 26 minutes when the MCPs are stopped. These time periods are sufficiently long to permit the operator to intervene and de-isolate feedwater flow according to this procedure. However, it should be noted that the normal feedwater reserves will be exhausted by about 25 minutes.

3.8. Pressure reduction by opening of pressurizer safety valves during hot leg/cold leg SBLOCA

The opening of the pressurizer safety valves or RCS pressure control has the lowest priority in the ERG. The effectiveness/benefit of opening the safety valves depends on its size. Paks specific analysis of the effectiveness of the safety valves was provided (see analysis No. 9. in Appendix 4). There were calculated cases with or without HA injection, with operator actions opening of PRZR safety valves and initiation secondary side depressurisation in FR-C.2. The results show that LPIS injection can be initiated temporarily by the primary bleed and core damage time could be delayed. However without early (starting in FR-C.2) secondary bleed this action could not provide long term LPIS flow because of increasing primary pressure by steam formation in the uncovered core. If the secondary side depressurisation is initiated only in FR-C.1 procedure that may occur faster reaching of 1200°C safety limit of cladding temperature.

There are some differences in that case when HA injection is available. Results show (see analysis No. 13. in Appendix 4) that in this case starting secondary bleed in FR-C.2 (at 370°C) is sufficient and stable LPIS injection could be provided.

Calculations show that, if HPIS injection is not available, early start of secondary bleed is very important to avoid core damage in FR-C.1, especially in case without HA injection. If so, the secondary side depressurisation initiated at 370°C core outlet temperature can avoid extended core damage, although short term violation of cladding temperature limit could occur.

3.9. Total loss of power with primary feed and bleed, analysis of the optimum B&F configuration

The following Paks-specific analyses and evaluations were required (see analysis No. 10. in Appendix 4)

- determine long term low pressure feed sources to SGs, and the capability to cool and depressurise the system to these low pressure sources with the SG inventory;

- based on the above analysis, determine the entry point of FR-H.1; it was decided to use symptoms: total feedwater flow <65 t/h and all SG wide range level <700 mm.
- determine the transition point to primary Bleed and Feed (based on B&F analysis).
- Based on analyses, the operator should initiate secondary bleed if there is no chance to restore any type of feedwater more than 8 hours.

There were 3 cases calculated using results of calculations of total loss of power. Primary B&F was initiated at different core outlet temperatures (320°C, 500°C and 650°C). Results show that bleed and feed process was effective in all 3 cases. The overall conclusion that operator can wait until core outlet temperature begin to rise and primary B&F should initiate at 300 mm in all SGs.

The optimal configuration for the B&F process was determined:
- 1 HPIS pump,
- 1 new PRZR relief valve (with 50 t/h capacity) or 1 PRZR safety valve,
- without HA injection.

The arguments are in following:
- even 1 HPIS cold water injection leads to violation of the 60 °C/h limit and is enough to borate the RCS and prevent core damage;
- with both configuration the process is successful;
- HA operation decreases the temperature quickly and fills up the PRZR early.

3.10. P-T diagram determination

The strategy given in the generic guideline should be applicable to Paks NPP, but specific analysis has been performed to establish the soak time requirements (see analysis No. 12. in Appendix 4)

3.11. Post LOCA cool-down

There were made some LOCA calculations with 33 mm breaksize - when the break flow can be compensated by 3 HPIS pumps at a primary pressure around 100 bar (see analysis No. 12. in Appendix 4) to examine the effectiveness of the major strategy of ES-1.2 procedure. It was checked whether sufficient subcooling and subcriticality can be assured throughout the coooldown process. Finally it was determined, that termination of the first HPIS pump is successful. But after the termination of the second HPIS pump it was restarted by the ECCS signal on low PRZR level. In conclusion, this breaksize is allowed to reduce HPIS injection but not to terminate all of 3 HPIS pumps.

4. CONCLUSIONS

All of the calculations were carried out by RELAP5/MOD 3.2 by KFKI-AEKI and on the basis of results most of questions have been answered. Consequences were included into the Paks-specific guidelines.

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[2] Isolation of primary leak by main loop isolation valves closure, Allorsz/R01/A, KFKI-AEKI, Budapest, May 1998.

[3] Summary of the PRISE project with respect to application of MGVs, Allorsz/R01/B, KFKI-AEKI, Budapest, September 1998.

[4] Evaluation of the use of primary loop isolation valves during accidents, Allorsz/R01/C, KFKI-AEKI, Budapest, January 1999.

[5] Upper plenum coolability during natural circulation conditions, Allorsz/R02/A, KFKI-AEKI, Budapest, December 1998.

[6] Steam generator collector head coolability during natural circulation conditions, Allorokt/R02/A, Budapest, February 1999.

[7] Malfunction of high pressure injection system at full power, Allorsz/R03/A, KFKI-AEKI, Budapest, August 1998.

[8] Cooldown without letdown during natural circulation including the boration aspect of SI termination, Allorsz/R04/A, KFKI-AEKI, Budapest, September 1998.

[9] Total loss of power, Allorsz/R06/A, KFKI-AEKI, Budapest, September/December 1998.

[10] Main steam header rupture, Allorsz/R07/A, KFKI-AEKI, Budapest, August/December 1998.

[11] Pressure reduction by opening of pressuriser safety valves during hot leg SB-LOCA, Allorsz/R08/A, KFKI-AEKI, Budapest, December 1998.

[12] Total loss of power with primary feed and bleed, Allorsz/R09/A, KFKI-AEKI, Budapest, September/December 1998.

[13] Total loss of power with primary feed and bleed, analysis of the optimum B&F configuration, Allorsz/R09/B, KFKI-AEKI, Budapest, March 1999.

[14] Pressure reduction by opening of pressurizer safety valves during cold leg SB-LOCA, Allorokt/R01/A, KFKI-AEKI, Budapest, June 1999.

[15] Specification of pressure-temperature limit-curves, Allorsz/R10/A, September 1998, KFKI-AEKI, Budapest,

[16] Post LOCA cooldown, Allorsz/R11/A, December 1998, Allorsz/R11/B, September 1999, KFKI-AEKI, Budapest.

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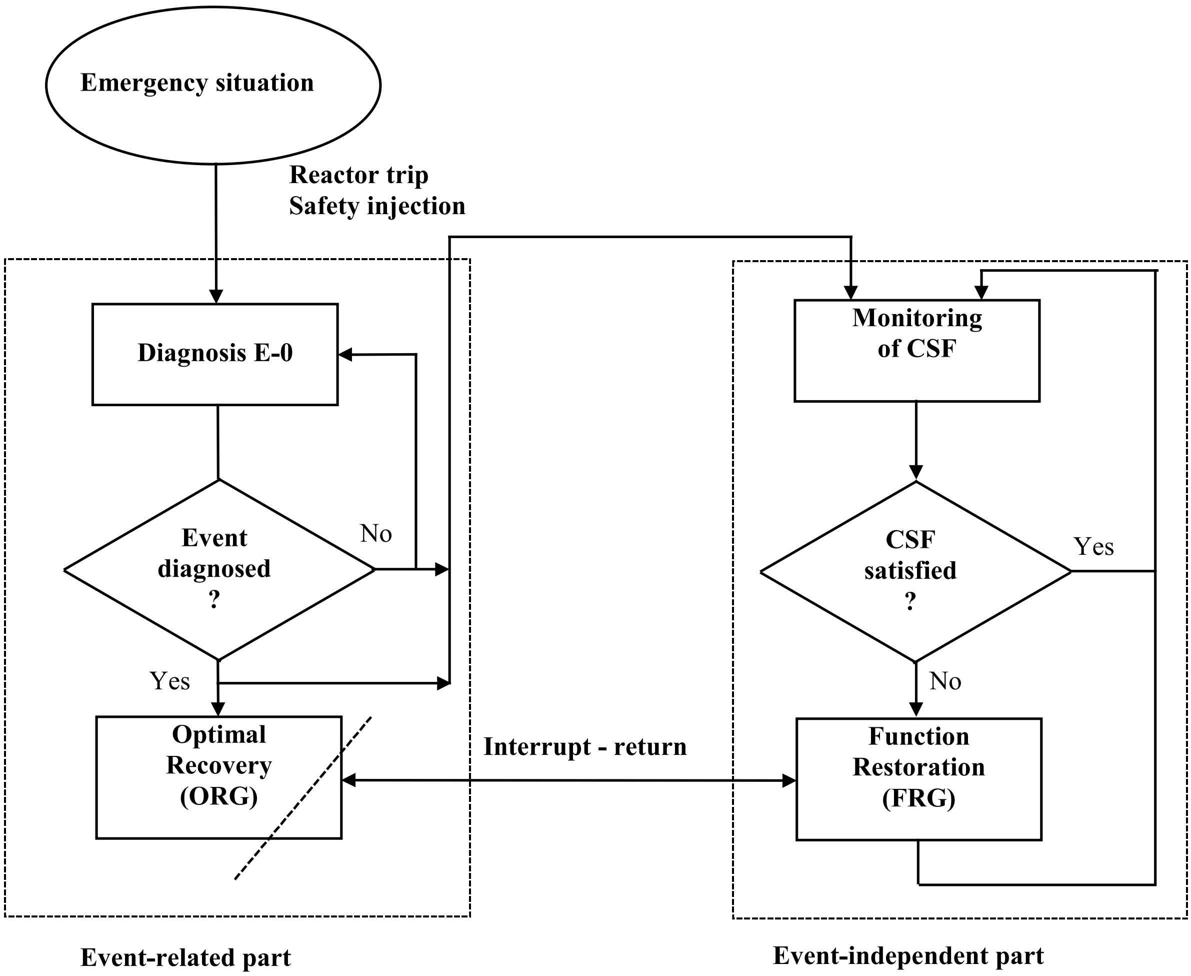
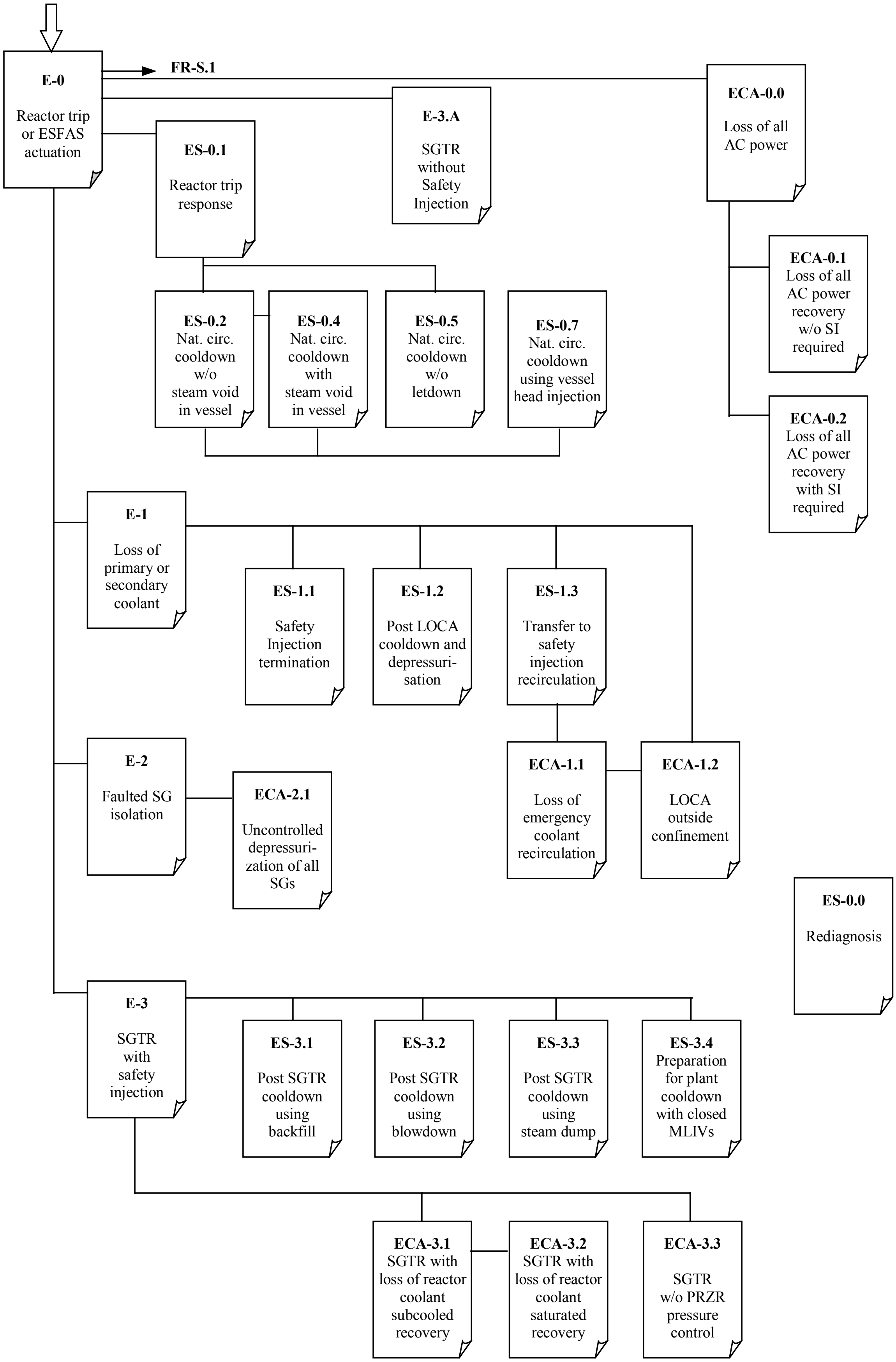
Appendix 1

GENERAL ORGANIZATION OF THE EOPs



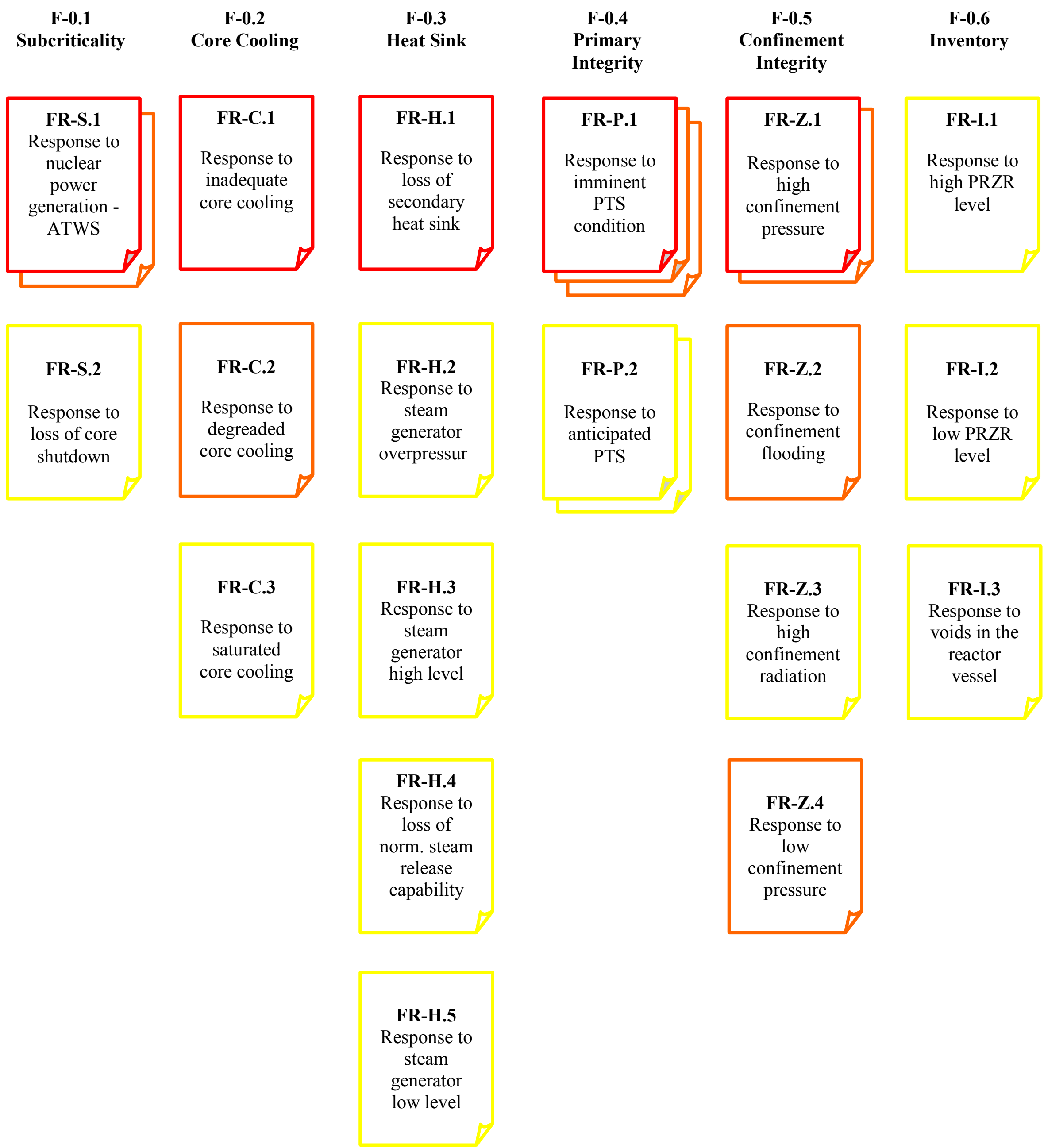
Appendix 2

ORGs STRUCTURE



Appendix 3

FRGs STRUCTURE



Appendix 4

LIST OF ANALYSES AND SPECIFIC STUDIES

1.	Allorsz/R01/A	Isolation of primary leak by main loop isolation valves closure.
2.	Allorsz/R01/B	Summary of the PRISE project with respect to applicationof MGVS
3.	Allorsz/R01/C	Evaluation of the use of primary loop isolation valves during accidents.
4.	Allorsz/R02/A	Upper plenum coolability during natural circulation conditions
5.	Allorsz/R03/A	Malfunction of high pressure injection system at full power.
6.	Allorsz/R04/A	Cooldown without letdown during natural circulation including the boration aspect of SI termination.
7.	Allorsz/R06/A	Total loss of power.
8.	Allorsz/R07/A	Main steam header rupture.
9.	Allorsz/R08/A	Pressure reduction by opening of pressureser safety valves during hot leg SBLOCA
10.	Allorsz/R09/A	Total loss of power with primary feed and bleed
11.	Allorsz/R09/B	Total loss of power with primary feed and bleed, analysis of the optimum B&F configuration.
12.	Allorsz/R10/A	P-T diagram determination.
13.	Allorsz/R11/B	Post LOCA cool-down.
14.	Allorokt/R01/A	Pressure reduction by opening of pressuriser safety valves during cold leg SBLOCA.
15.	Allorokt/R02	Steam Generator collector head coolability during natural circulation conditions.

MEANS OF ACHIEVING HIGH LOAD FACTORS AT OLKILUOTO 1 AND 2

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Abstract

Teollisuuden Voima Oy operates two BWR units Olkiluoto 1 and 2 that have achieved load factors typically higher than 90%. The operating experiences gained in the 1990s is summarised and the factors contributing to the high capacity factors are addressed. These include the general objectives for operation and maintenance, plant modernisation programme, maintenance principles, and outage policy and experiences. Finally, the international evaluations performed at Olkiluoto are mentioned.

1. INTRODUCTION

Teollisuuden Voima Oy (TVO) is a Finnish power company owned mainly by industry. The company was founded in 1969 primarily to satisfy the electrical power needs of the forest industry. The company aims at developing and operating large-scale power plants and producing electricity for its shareholders at cost price. TVO's first nuclear power plant unit, Olkiluoto 1, was connected to the national grid in September 1978, and the second unit, Olkiluoto 2, in February 1980. TVO's output of nuclear power covers about one fifth of the total national production in Finland.

Olkiluoto 1 and 2 are two identical BWR units with a net electrical output of 840 MW. They have single turbine-generator sets cooled by seawater. In addition to the power plant, the site incorporates an interim storage for spent fuel and a final repository for operating waste. A training centre with a full-scope replica simulator is also located at the site. The final disposal facility for spent fuel will be constructed in Olkiluoto in the vicinity of the power plant.

2. BACKGROUND: PRODUCTION COSTS

Production costs are the key issue in the competitiveness of electricity generation. Although nuclear power is a capital-intensive generation form, low operation and maintenance costs are a mandatory target for every power plant. At Olkiluoto we have been able to keep the fixed O&M costs at almost constant level during the past 15 years, as shown in Fig. 1.

Low O&M costs are the result of disturbance-free and predictable operation, which is promoted by several measures:

- clear organisations and responsibilities, less organisation levels, “simplified” functional processes
- optimisation of maintenance measures
- optimised outage policy
- key competences / outsourcing, partnership
- advanced data systems
- encouraging wages.

In our case, special emphasis is put on plant upgrading with related power uprating. This and other issues mentioned above are discussed below.

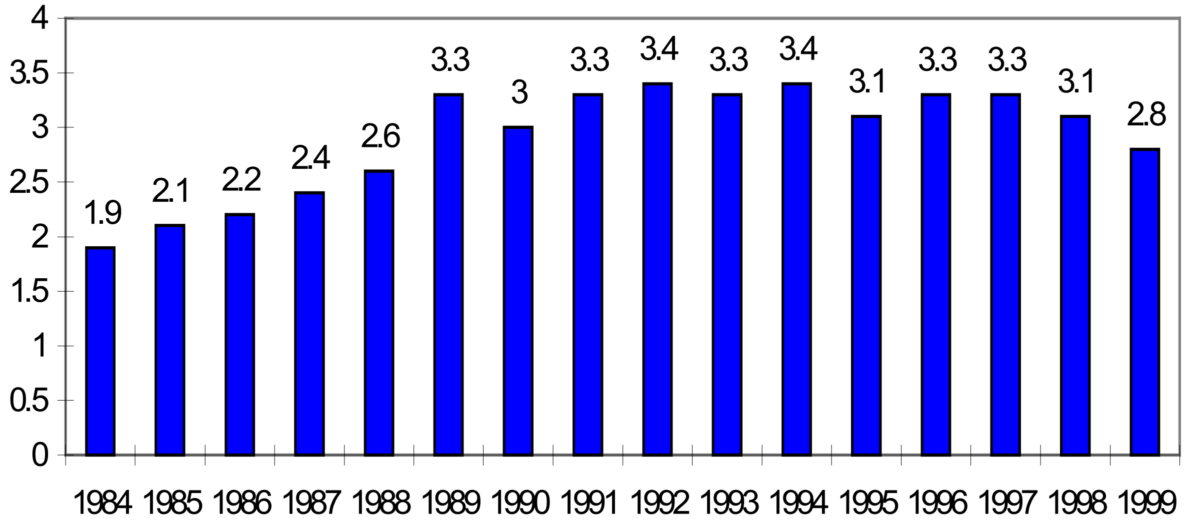

Fig. 1. O & M costs in Finnish p/kWh (including overhead costs, no fuel and capital costs).

3. OPERATING EXPERIENCE

Olkiluoto 1 and 2 have been able to keep very high load factors when compared internationally. The units have been in base-load operation, which means that capacity factor is a suitable performance indicator. The operation record of Olkiluoto 1 and 2 during the past 15 years is presented in Fig. 2. The annual electricity production has risen during the past few years, thanks to power uprating.

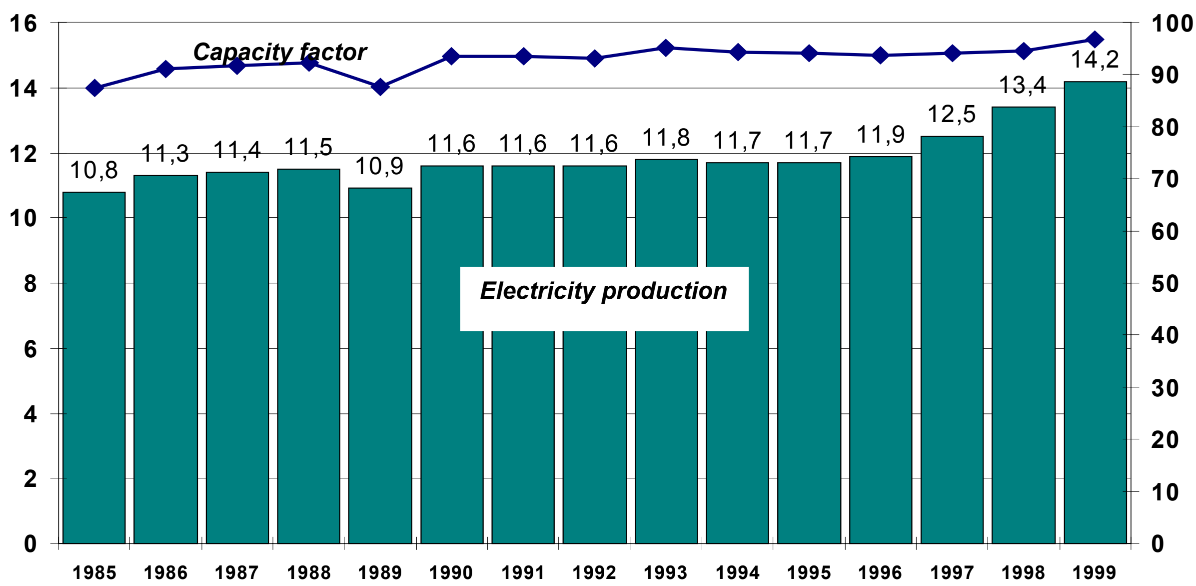

Fig. 2. Electricity production in TWH net (left) and capacity factor in% (right).

Keeping the average load factor for the last ten years over 93% has been possible because of good availability and short annual outages. The operational success is primary attributable to the early elimination of technical problems and preparation for defects beforehand. Development of outage planning and maintenance methods and a low rate of disturbances have also been very significant in keeping load factors high. The non-planned production losses have been typically between 0.2 to 2%, as shown by the statistics of production losses in Fig. 3.

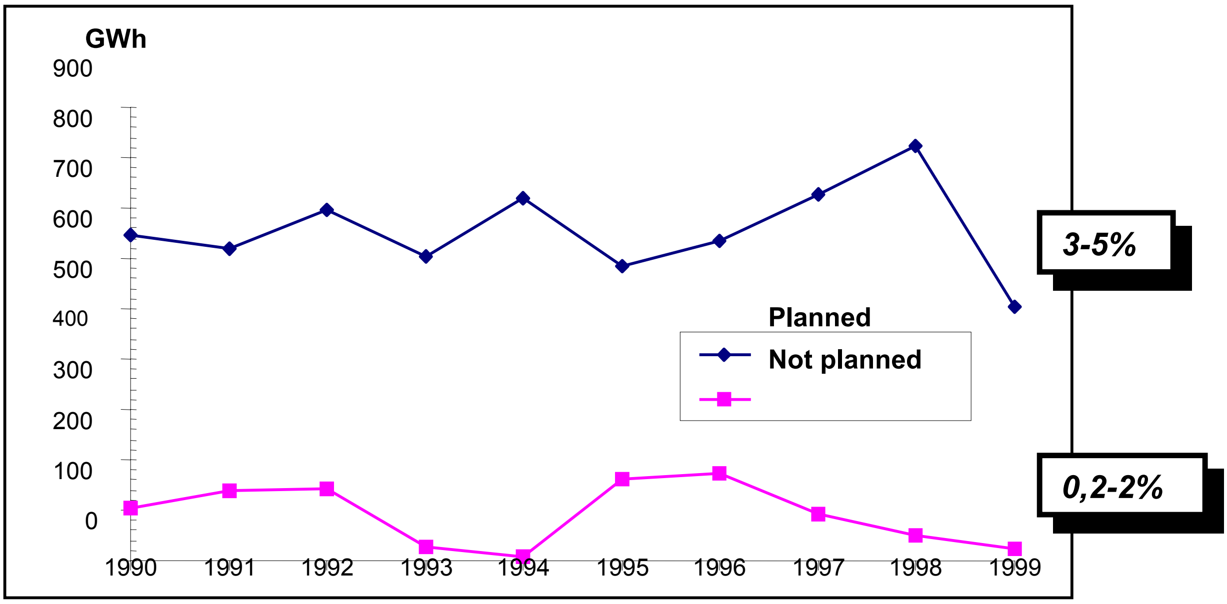

Fig. 3. Production losses in 1990-99 in GWh.

4. GENERAL OBJECTIVES FOR OPERATION AND MAINTENANCE

The following long-term goals are applied in the operation and maintenance of Olkiluoto 1 and 2:

- keeping the load factor over 90% and avoiding operational disturbances
- keeping the technical condition of the plant "as new"
- continuous learning and development of operation and maintenance functions
- reducing the production costs.

With good reason, we can claim that Olkiluoto plant is still reasonably modern. The original ASEA Atom design included several advanced features, and numerous improvements and modifications have been made since commissioning. This would not have been possible without substantial investments to improve the technical quality and design of the plant. Our modification and improvement policy is defined as follows:

- Annual investment level has been 100-150 MFIM (20-30 MUSD).

- 200-300 smaller and bigger modifications/improvements are made yearly. These include also investments to tooling, data systems, infrastructure, spare parts etc.

- The modifications are carried out without major impact on annual outage times.Examples are the exchange of generator and 2 LP turbines, modification of HP turbine and exchange of HP steam control/stop valves in outages of 15-19 days.

- The exchanges are performed in complete packages ("old parts out, new complete parts in") to keep short outage times.
- The design and preparation of outage activities starts one year in advance.

5. MODERNISATION PROGRAMME

A good manifestation of the afore-mentioned company policy is the wide-ranging modernisation programme which was carried out in 1994-1998 with the following objectives:

- reviewing safety features and enhancing safety, if feasible
- improving the production-related performance
- identifying factors limiting the plant's lifetime and eliminating them, if feasible
- enhancing the expertise of the own staff and improving productivity.

Four ways were followed to increase the electricity production:

- reducing the unplanned capability loss factor
- shortening annual outages
- improving thermal efficiency
- uprating reactor thermal power.

In the start-up phase of the programme, the technical development in the branch was exploited. This included mapping of new safety requirements and advanced design solutions. In the decision-making both our own operational experiences and experiences from other plants were considered.

In the execution of the programme, own staff was used as much as possible. Losses in electricity production were avoided. This meant that the plant modifications presupposing shutdown were implemented during normal refuelling and maintenance outages. In this way the average capacity factor stayed at 94% during the programme. Cost/benefit approach was applied, of course.

The programme consisted of about 40 separate projects, and the total costs amounted to 784 MFIM.

The expenditures were balanced by the increase of plant electrical output. In fact, the power level of the Olkiluoto units has been uprated twice:

- in 1984 by 8% from 2000 MW (660 MW(e)) to 2160 MW (710 MW(e)) and
- in 1998 by 15.7% from 2160 MW to 2500 MW (840 MW(e)).

Net electrical power has also increased due to modifications improving plant efficiency. The most important plant modifications contributing to the reactor power uprating are shown in Fig. 4. It should be noted that safety margins were not deteriorated, thanks to technical modifications including new fuel types. Plant technical lifetime of 60 years was found feasible in the studies performed.

6. MAINTENANCE PRINCIPLES

TVO has the philosophy of ”rolling 40 years' plant lifetime”. This requires that the technical condition of plant is kept at good level and maintenance and modernisation measures are conducted at right time by a cost-effective way, controlling the O&M costs. PSA, modern maintenance analyses methods and operation experience are used to allocate maintenance measures to right components and to keep up the desired plant safety and availability level.

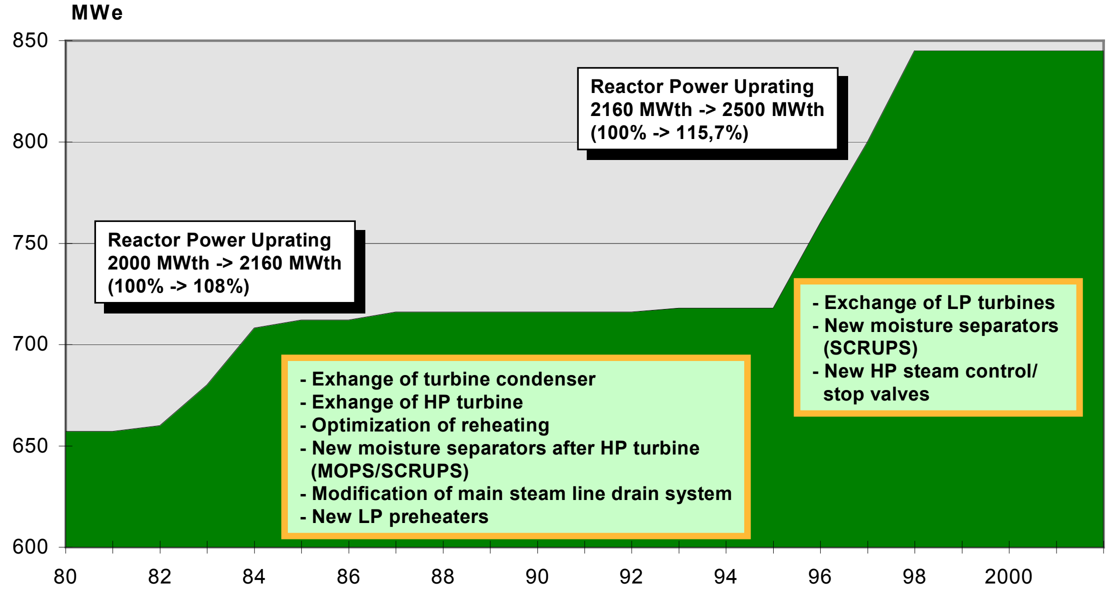

Fig. 4. Power uprating with related main plant modifications

Maintenance planning, execution and follow-up are supported by new and effective data systems.

The maintenance policy is verified in the distribution of maintenance man-hours in Fig. 5. The increased share of modifications in the late 1990s was caused by the modernisation programme.

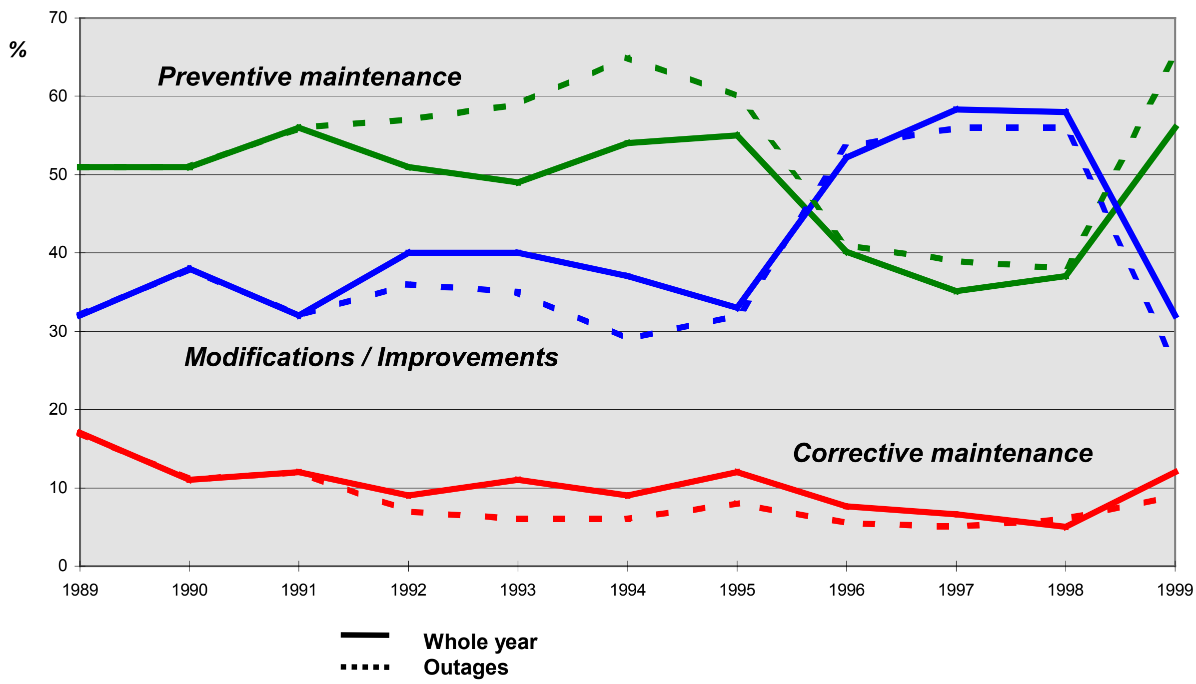

Fig. 5. Distribution of maintenance man-hours

7. OUTAGE POLICY AND EXPERIENCES

The price and availability of the replacement electricity for owners and the fuel economy are the factors determining the length of operating cycle and the time period to carry out outages. At Olkiluoto, regular 12-month cycles are most advantageous. 24-month cycle is not economically justified because of fuel costs and some technical restrictions. 18-month cycle should lead to outages during cold season with higher electricity price. The most important single reason to start outages in the beginning of May is the price of replacement electricity in the Nordic electric market. This time is also convenient for the availability of the competent and common resources used at Nordic nuclear power plants.

The lengths of annual outages in Olkiluoto NPP have decreased during the years from 40 days outage in 1980 to the 10 days (Olkiluoto 2) and 8.5 days (Olkiluoto 1) achieved in 1999, which is shown in Fig. 6.

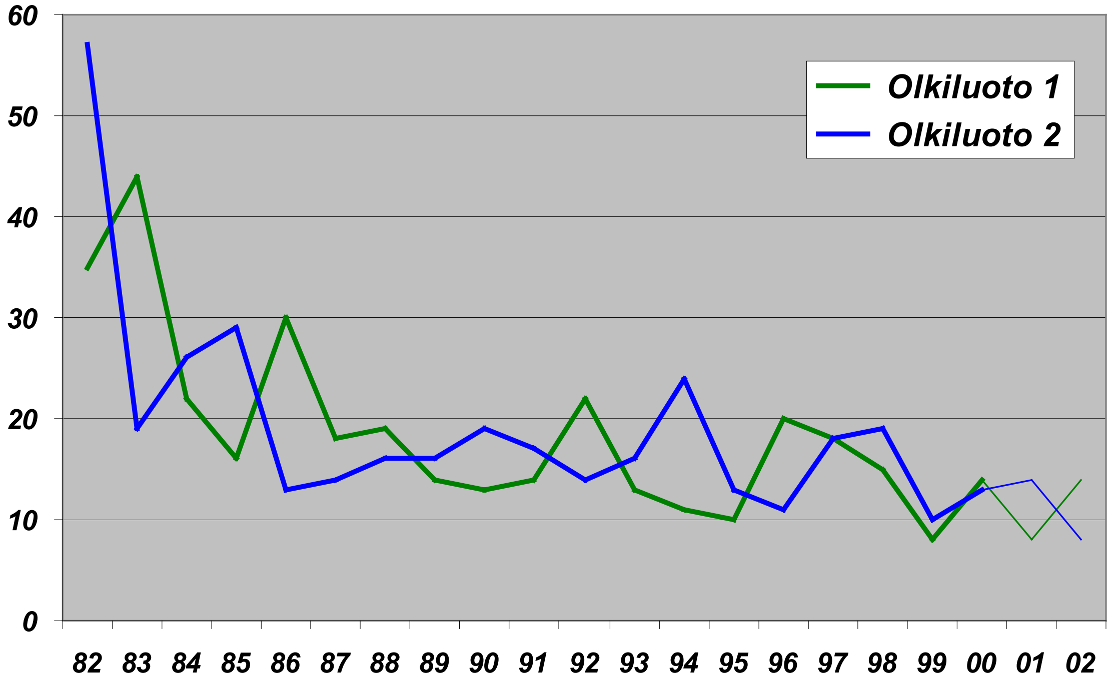

Fig. 6. Lengths of annual outages in days.

In order to optimise the long-term outage lengths and costs the outages are divided to two types: refuelling-only outages (about 9 days) and service outages (about 14 days, including for example the opening and inspection of turbine). They follow each other alternately. Major modification and repair works are carried out in 20-30 day's service outages every 5 years if needed. An interval of 2-4 days is enough between the short two outages.

Planning of outages must be done on several levels: long-term planning (about ten years), mid-term planning (about three years) and detailed planning of the following outages (1-2 years). Of course, short-term planning must also be done during the outages as well as preparedness maintained for unexpected repairs.

Continuous training of own and contractor personnel assure the availability of professional staff. Long-term cooperation with plant vendors and other affiliated companies is ensured by long-term agreements. Special training for plant components has been given to some Finnish affiliated companies. Multi-cycle fixed price contracts are also used, and over 70 per cent of the personnel have experience based on previous outages. Result and quality bonus systems are used for own personnel and partly also for contractor personnel. Coordination of the planning and execution of outage activities must always be the responsibility of the utility, and it must be done in a "partnership" arrangement with the main contractors having the common goals.

8. EVALUATION OF O&M QUALITY AND PRODUCTIVITY

An increasing number of NPPs have enforced different operation evaluation methods. These methods have been very beneficial and increased the understanding of operation and maintenance functions. At Olkiluoto, the following evaluations have been performed by various international organisations:

- IAEA/OSART in 1986
- evaluations by STUK (Finish regulatory body) in connection with the renewal of operation license in 1983 and 1997
- WANO Peer Review type self evaluation
- SCEMM (Scandinavian Center for Maintenance Management) in 1996-97
- SFS in 1997, TTK/INSPECTA in 1997 (Mechanical Maintenance)
- IAEA/ASSET in 1997-98
- INPO (limited extent) in 1998
- WANO Peer Review in 1999.

IMPLEMENTATION OF THE RCM APPROACH AT EDF NPPs: CURRENT STATUS

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Abstract

To ensure safest possible operation and to get best overall economic performance of its Nuclear Power Plants, 10 years ago Electricité de France launched a Reliability Centered Maintenance (RCM) project to optimize the preventive maintenance programs. The principles underlying the RCM approach are based on common sense: failures must be prevented by preventive maintenance operations in all cases when the repercussions for the installation could be serious or critical in term of safety, availability or maintenance costs. The approach is a 3-phases process:

1. Evaluation of the functional consequences of failures
2. Evaluation of performances based on the analysis of experience feedback
3. Optimization of the preventive maintenance tasks

The new preventive maintenance programs are presently almost completed and progressively implemented. Here and now, the implementation of the RCM approach allows to emphasize some benefits:

- Same or increased level of safety
- Same or increased level of plant availability
- ALARA principles better taken into account
- Cost control
- Positive change in maintenance culture

Some new studies are in progress, such as the development of a "Risk Based In Service Inspection" for passive components.

1. INTRODUCTION

As every nuclear power plant operator, Electricité de France have two permanent concerns: to ensure safest possible generation and to get best overall economic performance. These two concerns are the basement of the maintenance policy and have lead EDF to clarify maintenance concepts and to initiate large studies aiming to optimize maintenance activities.

For this purpose, ten years ago EDF launched a Reliability Centered Maintenance (RCM) project to optimize preventive maintenance for its Nuclear Power.

2. MAINTENANCE AT EDF NUCLEAR POWER PLANTS

2.1. Maintenance concepts

It is essential to clarify the concepts before starting any study on the optimization of maintenance activities. The definition of the different concepts of maintenance used at EDF are summarized on the figure 1. They can be defined as follows:

Maintenance

All the tasks which enable equipment to be kept in a given state or be able to assure a given service, or which restore it, so that it can meet these requirements.

Corrective maintenance

All the tasks carried out on a piece of equipment, so that its operating capability can be restore after a functional failure. It therefore always has an unplanned nature.

Preventive maintenance

Involves all the tasks carried out on a piece of equipment in order to reduce the likelihood of a functional failure. The aim of the preventive maintenance is to prevent functional failure and, consequently, to ensure the possibility of using the equipment for a given time.

Systematic preventive maintenance

Involves the systematic replacement of components, according to a predetermined schedule. This means the replacement of components whose life span is known. It also includes systematic lubrication jobs as well as minor upkeep work. This definition avoids confusion between systematic checking or even continual checking to determine the state of the equipment (conditional preventive maintenance) and systematic preventive maintenance.

Conditional preventive maintenance

Set of actions including:

- *observations* which enable the state to be known, under operating or shutdown conditions, with or without dismantling. including the checks, inspections, tests and surveillance under operating conditions,
- *analyses* which assess this state and forecast its evolution, depending on the use of the equipment (expert opinion, operating experience, harmfulness studies…)
- *measures* which prevent failures by correcting degradation that cannot be left in the existing state.

2.2. Main figures

To operate 58 units needs huge financial and human resources. A large part of these resources is assigned to maintenance.

The main figures which characterize maintenance at EDF NPPs are:

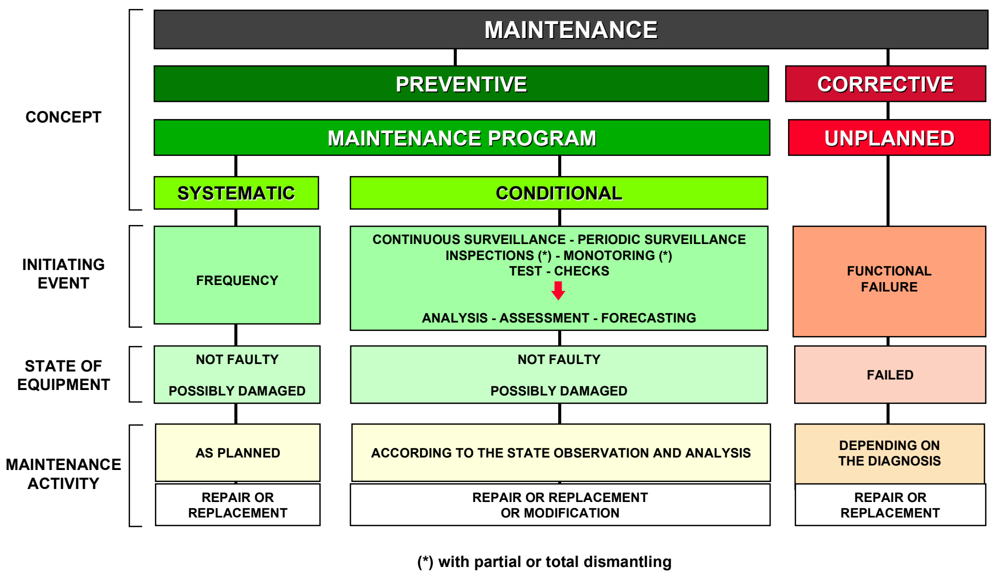

FIG. 1. Maintenance concepts.

Maintenance expenses:

- Overall Maintenance Costs: 1500 M€ per year for 58 units (including 900 M€ for subcontractors) which represent 50% of total O&M costs and 17% of kW.h cost
- 900 M€ spent on Equipment:
  - 2/3 Preventive Maintenance (major part carried out during refueling outages)
  - 1/3 Corrective Maintenance
  - 35% spent on Primary Circuit Components
  - 50% spent on 50 Major Systems (i.e. safety and /or availability related)
  - 15% spent on the remaining 150 systems
- <u>Personnel</u>:
  - 10 000 from EDF
  - 20 000 from Contractors
- Capability loss factor due to component failure: 1.5%

3. RCM: EDF APPROACH

3.1. Why a RCM approach?

EDF developed its own Reliability Centered Maintenance Method to better address the choice between Preventive and Corrective Maintenance. Therefore, the problem is to answer the following question:

- On which equipment is Preventive Maintenance to be performed?
- What Preventive Maintenance tasks are to be performed on these equipment?
- How to optimize Preventive Maintenance Programs with respect to Safety/Plant Performance/Costs/Radiation Exposure?

3.2. Methodology

The principles underlying the RCM approach are based on common sense: failures must be prevented by preventive maintenance operations in all cases when the repercussions for the installation could be serious or critical in term of safety, availability or maintenance costs.

The methodology developed by EDF to select maintenance activities is based on the assessment of the couple: *"Likelihood of failure - Functional effects of the failure"*. It follows a logical sequence represented in figure 2.

It is a 3-phase process.

Phase 1: Stakes assessment

The first step of this phase consists in analysing the role of the selected system and the possible failure modes and causes. For systems with large stakes (safety, availability and/or costs) this is made from the system level to the component level. When the component itself is considered as a "large stake component" the analysis is performed to the level of detail required for maintenance. This allows to draw up the list of significant failure modes and components. The reliability assessment technique used in this phase is Failure Mode and Effect Analysis (FMEA).

The second step aims to identify among the significant failure modes and components those which are critical in term of safety, availability and maintenance.

For this purpose, information on operating experience is necessary and is provided by the second phase of the approach.

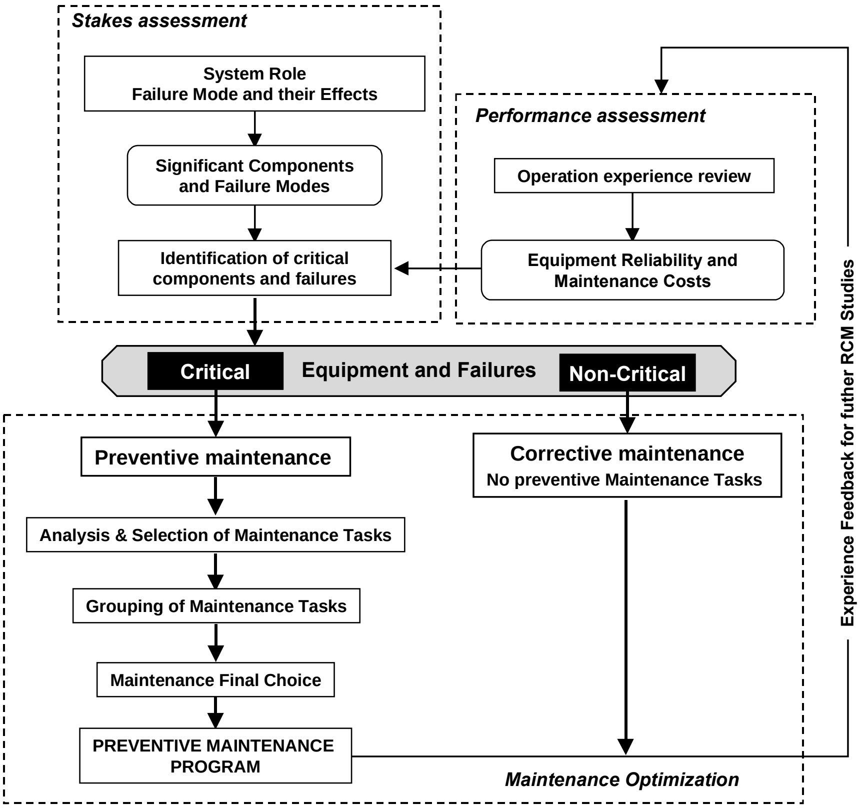

FIG. 2. EDF RCM methodology on a selected system.

Phase 2: Performance assessment

This phase, fundamental for the RCM approach, consists in analyzing the operating experience. It provides information on the failures and degradations which have actually occurred on current systems. It is the only mean of showing that an expected failure or degradation has only rarely occurred and that a given preventive maintenance task is therefore not needed and may even be harmful.

Phase 3: Maintenance optimization

The final step is the selection of preventive maintenance tasks for all components found critical. Maintenance tasks are selected according to a specific selection logic which views all preventive maintenance operation in a ascending order of complexity: lubrication, in-service monitoring, checks and tests, complete overhaul and scheduled replacement.

At the present time, this final step is essentially based on expert judgment.

3.3. Implementation

The project is managed at the corporate level but involves in every plant specialists of different skills such as maintenance, operation, safety, management.

Systems with large stakes (use of complex PRA studies): the studies are carried out at the Corporate level to deal in priority with the 50 most important systems. Presently 43 out of 50 Preventive Maintenance Programs are performed.

The Sites have been in charge of carrying out the studies of all the other remaining major systems; 62 out of 70 Preventive Maintenance Programs are already performed.

On every plant, a multi-skills team supported by corporate level departments, analyses 1 or 2 systems per year and defines the related maintenance programs which become prescriptive after validation.

The Sites have taken over the methodology concepts. The nuclear plant personnel become a major actor in the development of the preventive maintenance policy and will be ready to implement RCM on other systems.
Most of the preventive maintenance programs, which only concern active components at this time, are drawn up using the RCM method, both on 900 and 1300 MW series.

4. MAIN BENEFITS

Here and now the experience feedback of the implementation of the RCM methodology over a few years period allows to emphasize some benefits, in term of Safety, Availability, Radiation Protection and Maintenance Costs.

4.1. Same or increased level of safety

Regarding the safety, the application of the RCM to draw up Preventive Maintenance provides the following benefits:

- Preventive Maintenance programs are now established on new systems which have not previously (Safety Injection & Containment Spray Pump Motor Room Ventilation System)
- Maintenance re-allocation: more maintenance is required on some components, less on others (RHRS: on Reactor Heat Removal System more maintenance on some valves, less on others)
- Periodic tests are better taken into account

4.2. Same or increased level of plant availability

Concerning the plant availability, the positive repercussion of RCM are:

- Not so many Maintenance Tasks are to be performed during outages

  Preventive Maintenance Programs are being streamlined: No more "Nice but not necessary Jobs" filling outages programs,

  Fewer problems are encountered when restarting the plant,

  No more tight schedules: outages with reduced maintenance are easier to schedule.

- Surveillance Tasks have replaced some Maintenance Tasks

  Operation people are conducting periodical tests allowing a better anticipation of the problems and consequently a reduction of components failures.

4.3. ALARA principles better taken into account

A rationalization of the preventive maintenance tasks lead to a reduction of the radiation exposure The main reasons are:

- Some dose costly maintenance tasks are suppressed or their periodicity is lengthened (Sebim valves, RHR Pumps, RCP pumps.)
- Radiation Exposure is considered in decision making between Preventive and Corrective Maintenance (Fuel Handling and Storage System)

4.4. Cost control

A first estimation of the reduction of the maintenance costs gives the following results (Figure 3):

- 15% to 20% cost reduction in Preventive Maintenance
- 50 M€ total saving expected by year 2000

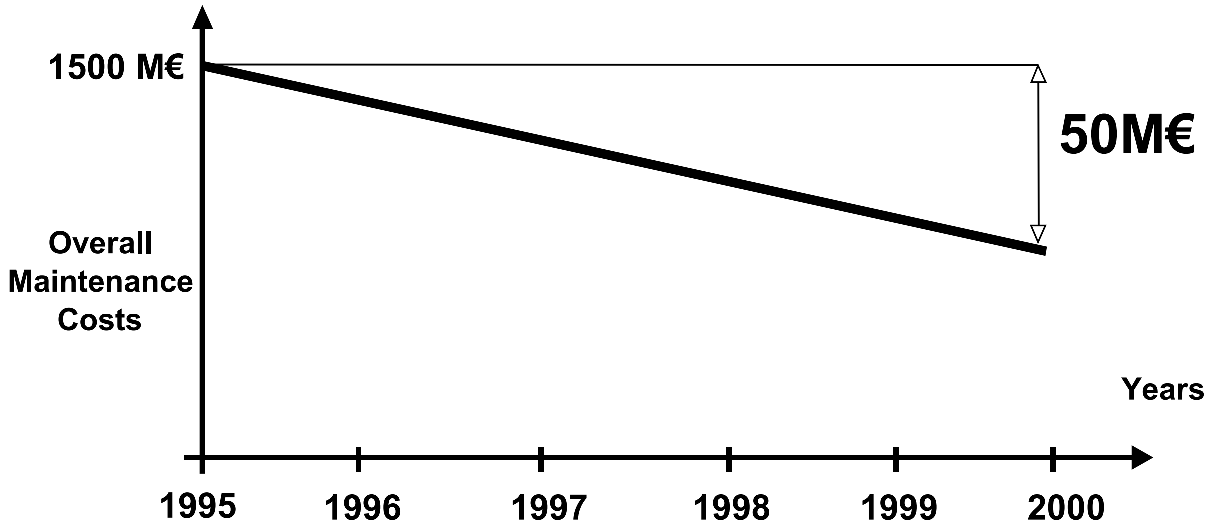

FIG. 3. Evolution of Maintenance costs over the 5 past years.

This preliminary trend has to be confirmed as and when the implementation of the new preventive maintenance programs will be effective.

4.5. Example of changes on a system: Auxiliary Feedwater System

The table 1 below gives an example of changes on the Auxiliary Feedwater System.

For this system, some valves which was not initially subjected to maintenance activities are now inspected. On the contrary, the maintenance tasks of some other components have been canceled (check valves) or their periodicity lengthened (electric motor).

TABLE I: EXAMPLE OF CHANGES ON THE AUXILIARY FEEDWATER SYSTEM MAINTENANCE PROGRAMME

Equipement	Maintenance Task	Frequency
Check Valves: 5, 6 & 11 (down stream of Pumps)	Complete periodical inspection (overhaul)	5 years → Task canceled
Valves 131, 132, 133	Complete periodical inspection (overhaul)	None → 10 years (PSA)
Valve 115 VD (Tank feed line manual V.)	Surveillance task and operability test	None → 1 year (PSA)
Valves 30 to 35 VD (Steam Genartor isolation)	Operability test	None → 5 years
Elec. Motor driven Pump	Complete overhaul	8 years → 16 years
Pump Packing replacement	Time based replacement	Every year to 4 years Condition based

4.6. Other benefits

RCM methodology infers a Positive Change in maintenance culture, mainly due to the functional nature of the approach:

- maintenance people and operation people work together during the studies,
- maintenance is more process oriented,
- maintenance optimization calls for a good knowledge of the required functions, then better every day decisions are expected.

In an other hand, the maintenance optimization calls for a detailed analysis of experience feedback, a better quality of experience feedback is expected with regard to

- collection and analysis of events or maintenance costs
- extraction of reliability data
- definition of indicators (availability, costs).

5. NEW DEVELOPMENTS

New developments are currently in progress, such as

- Development of a methodology "Risk Based In Service Inspection" for passive components (pipes). The studies started in 1997 and the first preventive maintenance programs will be issued soon,
- Development of a "simplified" methodology for components or systems with smaller stakes,
- Until end of 2000, the maintenance optimization of the 50 important systems will be completed
- From 2001 to 2005, the maintenance optimization of less important systems will be carried out.

6. CONCLUSION

The implementation of optimized Preventive Maintenance Programs drawn up using the RCM methodology is now effective at EDF nuclear plants. The first experience feedback shows some appreciable benefits regarding safety, performance and cost. The impact on the operator culture is proven positive as well.

RCM can be considered as an important contributor to the safety, competitiveness and acceptance of nuclear power.

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MAINTENANCE MANAGEMENT FOR NUCLEAR POWER PLANT "INTEGRATED VALVE MAINTENANCE"

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Abstract

The deregulation of Europe`s power market does force many utilities, and especially nuclear power plant operators, to introduce extensive cost-cutting measures in order to be able to compete within this new environment. The optimization of plant outages provides considerable potential for raising plant availability but can also lower operating costs by reducing e.g. expenditure on maintenance. Siemens Nuclear Power GmbH, in cooperation with plant operators, is currently implementing new and improved service concepts which can have a major effect on the way in which maintenance will be performed in the future. Innovative service packages for maintenance in nuclear power plants are available which can be used to perform a time- and cost-effective maintenance. The concepts encompass optimization of the overall sequence from planning in advance to the individual measures including reduction of the scope of maintenance activities, identification of cost cutting potential and bundling of maintenance activities. The main features of these maintenance activities are illustrated here using the examples of outage planning and integrated valve maintenance. In nuclear power plants approx. 5000 valves are periodically preventively, condition-based or breakdown-based maintained. Because of this large number of valves to be maintained a high potential of improvements and cost reductions can be achieved by performing an optimized, cost-effective maintenance based on innovative methods and tools. Siemens Nuclear Power GmbH has developed and qualified such tools which allow to reduce service costs while maintaining high standards of safety and availability. By changing from preventive to predictive (condition-based) maintenance - the number of valves to be maintained may be reduced considerably. The predictive maintenance is based on the Siemens Nuclear Power GmbH diagnostic and evaluation method (ADAM). ADAM is used to monitor the operability of valves by analytical verification of the functioning capability and the readiness for function which is to be verified periodically during lifetime (trending).

1. INTRODUCTION

The deregulation of Europe`s power market does force nuclear power plant operators to introduce extensive cost-cutting measures in order to be able to compete within this new environment.

The optimization of plant outages provides considerable potential for raising plant availability but can also lower operating costs by reducing expenditure on maintenance. Economic advantages also resulted from the substantially shorter duration of refueling outages achieved through advanced equipment used for inspection and repair.

For example, only one traverse per area is required for examination of the outer surface of the cylindrical section of boiling water reactor pressure vessel when phased-array probes are used. In this way, the time for the reactor pressure vessel inspection could be reduced by 60%.

Siemens offers a comprehensive range of innovative services packages for all maintenance activities, including component service for example for valves, pumps and piping, as well peripheral activities such as e.g. radiology and radiation protection.

2. OPTIMIZATION OF MAINTENANCE

Siemens Nuclear Power GmbH, in cooperation with plant operators, is currently implementing new and improved service concepts which can have a major effect on the way in which maintenance will be performed in the future. Innovative service packages for maintenance in nuclear power plants are available which can be used to perform a time- and cost-effective maintenance.

The concepts include optimization of the overall sequence from planning in advance to the individual measures including reduction of the scope of maintenance activities, identification of cost-cutting potential and bundling of maintenance activities.

The goal of any maintenance optimization program is to reduce maintenance costs while maintaining the same high level of availability.

The main features of these maintenance activities are illustrated here using the examples of outage planning and integrated valve maintenance. The advantages of bundling of individual packages are also briefly discussed.

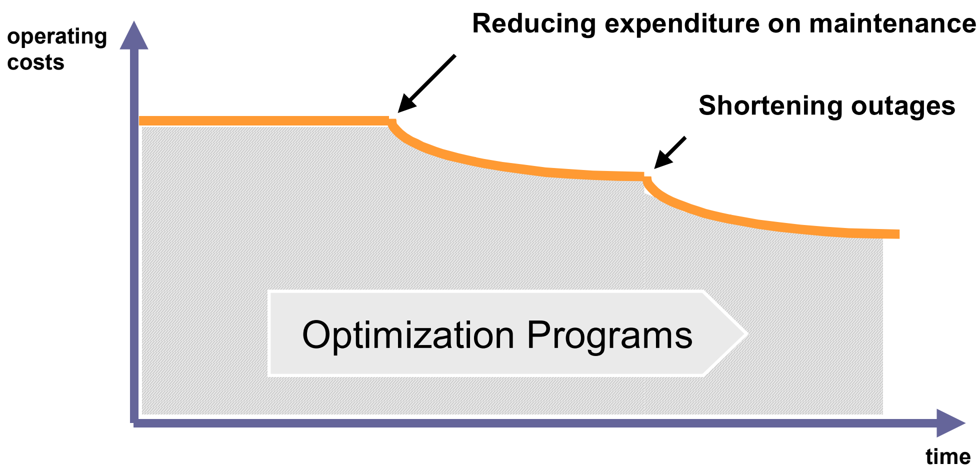

FIG. 1. Reduction of maintenance costs.

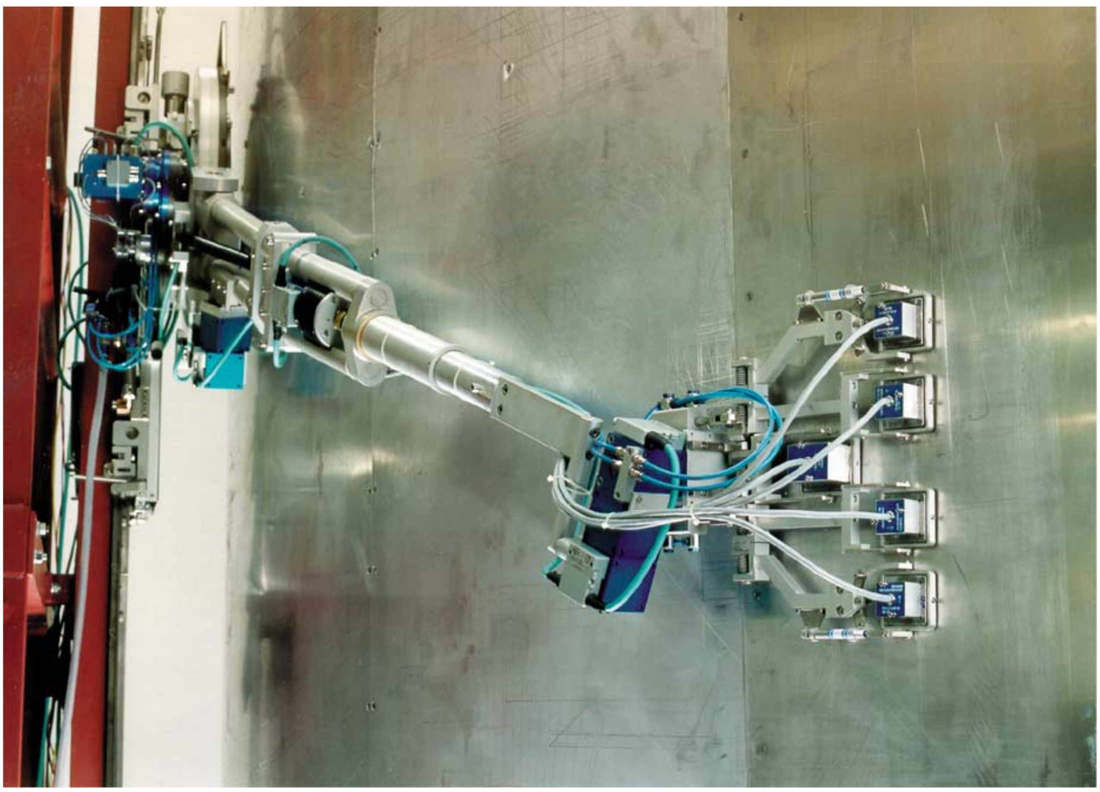

FIG. 2. Examination of outer surface of cylindrical section of BWR reactor pressure vessel by a phased-array probe.

Service package	Primary objectives and measures
Valve maintenance	Work sequence optimization through bundling of activities; proof of fulfillment of safety standards; extension of service intervals; condition-based maintenance
Radiology and radiation protection	Consultation on dose rate reduction, in particular on highly contaminated plant items and on work procedures
Pump maintenance	Optimization of overall work sequence through coordination of activities and persons involved
Steam generator maintenance	Long-term retention or improvement of features through appropriate measures up to full service
Piping maintenance	Performance of engineering tasks such as preparation of analyses, elaboration of measures, generation of design review documents; piping replacement
Maintenance manuals	Work documentation using advanced information technology, which simplifies future use of information
Condition-based maintenance	Reduction of service effort by adapting measures to the current conditions of plant items
Water chemistry diagnostics	Simplification of diagnostics through computer-aided analyses and evaluation; localization of causes in the event of deviations from optimal condition
Teleservice	Generation of remote diagnoses of current plant and component conditions using advanced communications technology
Outage consultation	Reduction of time and cost of work through involvement in planning of measures

FIG. 3. Innovative service packages for maintenance activities in nuclear power plants.

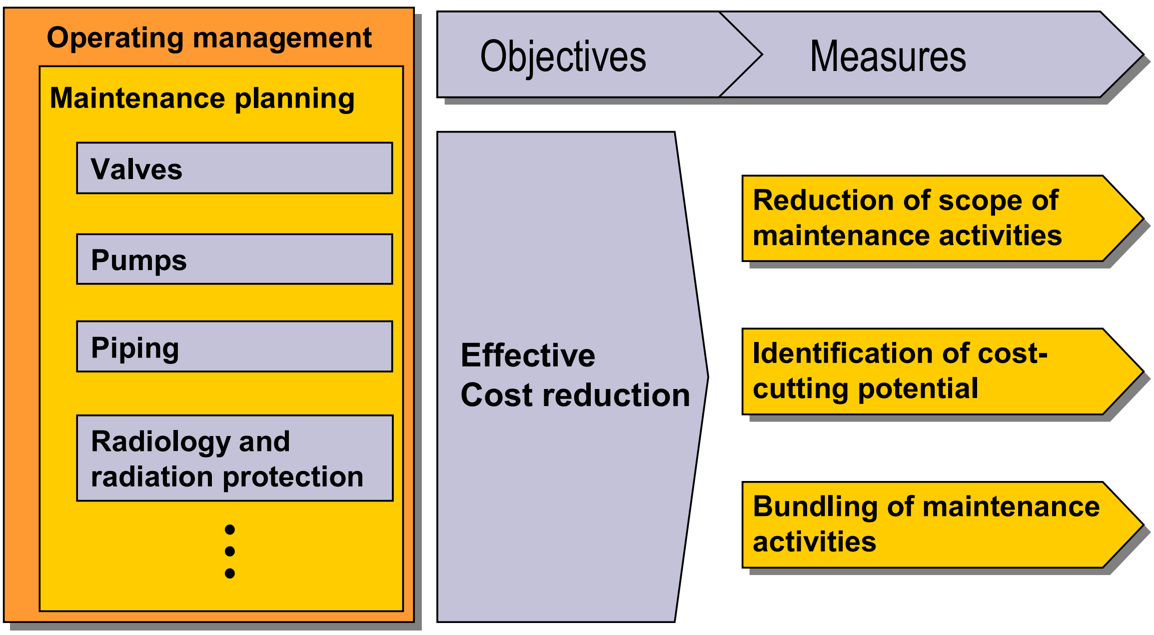

FIG. 4. Optimization of maintenance processes.

2.1 Integrated valve maintenance

In nuclear power plants approx. 5000 valves are periodically preventively, condition-based (predictive) or breakdown-based maintained. Because of this large number of valves to be maintained a high potential of improvements and cost reductions can be achieved by performing optimized maintenance methods and using innovative tools.

Siemens Nuclear Power GmbH has developed and qualified innovative methods and tools which allow to reduced service costs while maintaining high standards of safety and availability. The functioning capability and readiness for function of valves are proved and ensured by the integrated valve concept. Basic elements are:

- calculation
- design evaluation
- monitoring and maintenance of valves.

2.1.1. Verification of functioning capability

The verification of functioning capability is to be made once as a follow-up for existing valves or for new designs in advance. The analytical concept includes load calculations for the actuator and for the valve components as well as the evaluation of the function-related features of the design including the mechanics. The analytical models used are validated by tests in power plants or in appropriate test rigs. The model for function determines the required actuating thrust and torque considering the relevant parameters and tolerances. The model for load analysis covers the load capacity of the valve.

Measures to Ensure Functioning Capability and Readiness for Function

Calculation	Design Evaluation	Monitoring, Maintenance
•Preparation of guidelines for calculations	•Preparation of guidelines for design	•Active power measurement
•Testing support (laboratories)	•Overall evaluation of design	•Condition-based maintenance based on ADAM® (Valve dia-gnosis and evaluation method)
•Validation of evaluation methods based on test results	•Experimental verification of design (e.g. blowdown test)	•Special measurements
•Re-evaluation and modification of design review documents	•Coordination of modifications with plant operator, manufacturers and authorities	•Expert support
•Integration of systems engineering	•Dimensional data sheets	•Optimization of valve maintenance

FIG. 5. Integrated valve maintenance concept.

The design evaluation covers the limits and weak points of the parameters relevant function and ensures the validity of the calculation assumptions.

2.1.2. Verification of readiness for function

The methodology of the integrated valve concept includes the guarantee for readiness for function during lifetime. In nuclear power plants inspections of the function of the whole system is performed periodically beginning with power supply and I&C including the shut-off element in the valve. The results are compared with the base measurements. For evaluation of the results of periodic examinations and inspection the model for function and the model for load analysis as well as the design evaluation are used.

2.2 Diagnosis software ADAM (valve diagnosis and evaluation methodology)

By changing from preventive to condition-based maintenance - based on the Siemens Nuclear Power GmbH diagnostic and evaluation method ADAM (Armaturen-Diagnose und Auswerte-Methode) — the number of valves to be maintained may be reduced considerably. The application of such sophisticated software tool will lead to cost reduction and can also extend the service intervals.

ADAM can be used to monitor the operability of valves by analytical verification of the functioning capability and the readiness for function which is to be verified periodically during lifetime (trending).

The structure and the evaluation method of ADAM is shown in Figure 6.

The individual modules “calculation” (see verification of functioning capability), “baseline measurement” and “diagnostic” implemented in ADAM are used to determine deviations from permissible tolerances of certain parameters (e.g. zero loads, running power etc.) or trends in the function of the valve and the actuator.

The monitoring of the readiness for function during lifetime is performed in the following sequence with the methods described below:

Normally, the active power of the AC-motor measured from the switchgear room and the baseline measurement allow the monitoring of all functional relevant parameters to verify readiness for function. Degradation or deviations in the function of the valve and the actuator with respect to upcoming failures can be detected in advance. If there are any indications of deviations from certain tolerances further parameters are measured in-situ like torque, stem thrust and the displacement of the worm gear.

Through the systematic monitoring of the motor-operated valves degradation can be detected and corrected by means of maintenance measures so that function-related parameters remain within design limitations.
The reliability of the function of valves and actuators defines the safety and availability of the power plants.

The benefit from the application of ADAM is illustrated in Figure 7.

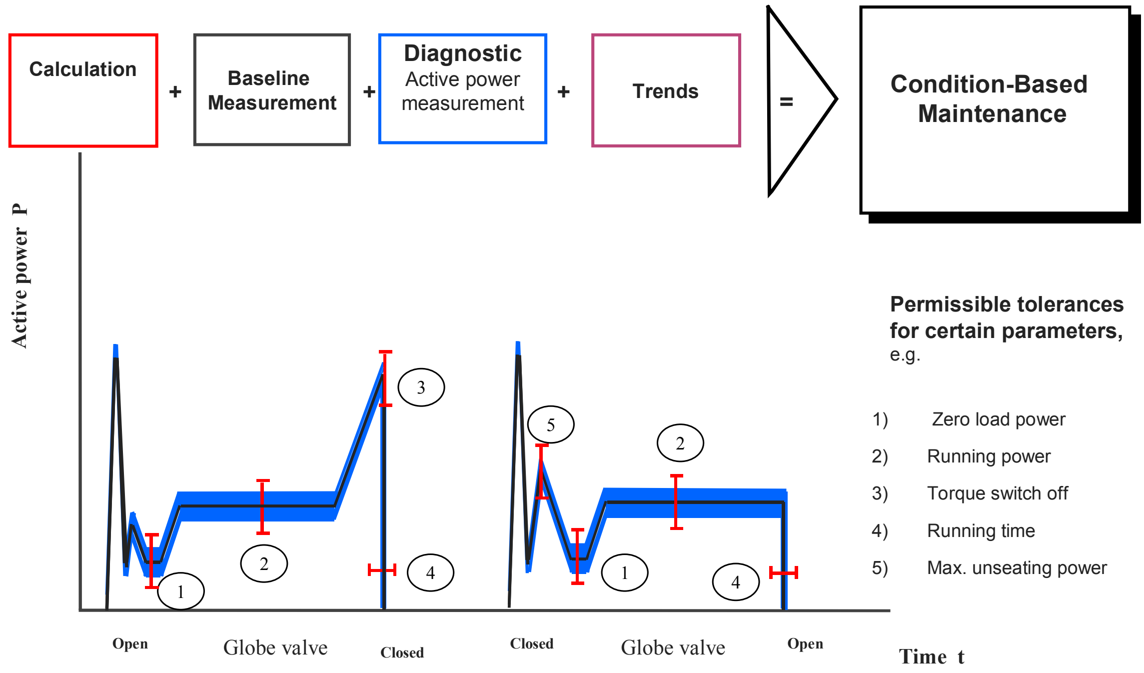

FIG. 6. ADAM®: The way to condition-based maintenance of MOV's evaluation method of ADAM.

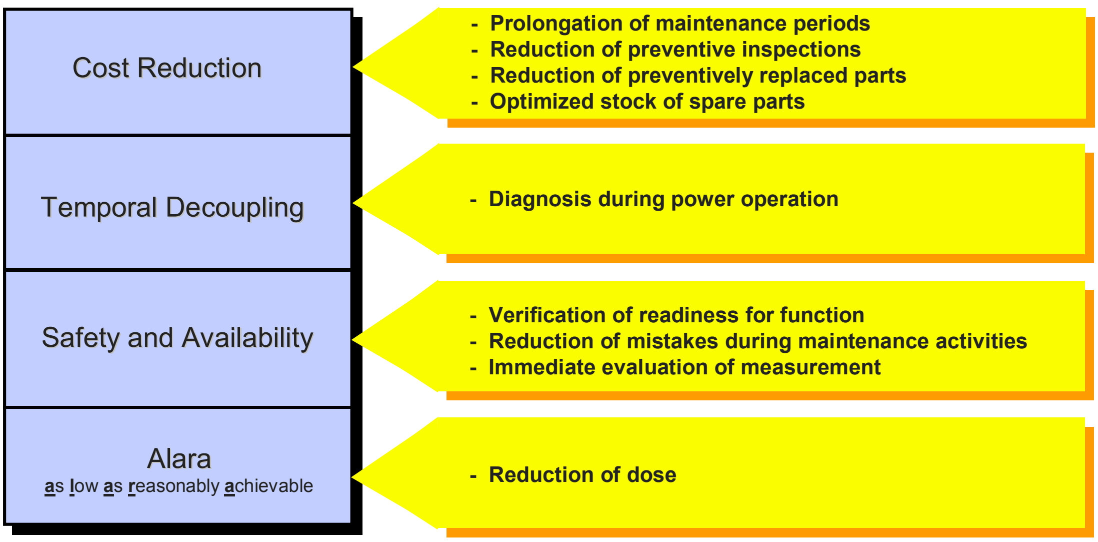

FIG. 7. ADAM®: Benefit from ADAM.

Owing to the periodic monitoring of valves including actuators and the evaluation of the measured data by ADAM the actual conditions of the valve can be determined to ensure the readiness for function. Based on the information obtained from ADAM and operational experience over a longer time period a condition-based maintenance can be performed in order to extend the maintenance periods. That means, preventive measures as inspections and replacement of parts can be reduced.

An additional advantage is also to perform the diagnosis during power operation by temporal decoupling of the valve system. The measurement of the active power from the switchgear room, e.g. with the advanced tool SIPLUC resulting in a diminished radiation exposure of the performing staff.

With these described measures costs will be reduced and high availability as well as safety of nuclear power plants can be ensured.

2.3 Diagnosis tool SIPLUG

The handling and application of the advanced diagnosis tool SIPLUG are shown on the photographs in Figure 8.

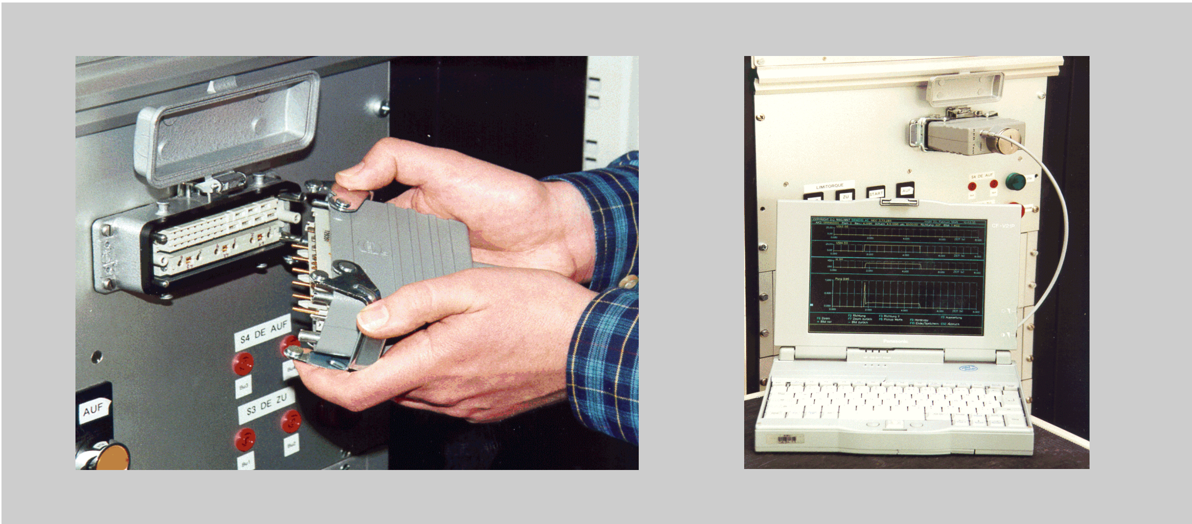

FIG. 8. ADAM®: SIPLUG®.

SIPLUG is designed as a sophisticated data acquisition tool which is connected to the electronics of the valve system located in the switchgear room. The significant features of SIPLUG can be described as follows:

- automatic recording of the course of active power for up to 10 valve strokes
- because of an own power supply the recorded data are stored within SIPLUG which can be evaluated afterwards by a separate computer
- the single plug (SIPLUG) is expandable to an on-line monitoring system.

2.4 Process analyses

A comprehensive maintenance strategy includes all measures needed to identify cost-cutting potential. Analyses are performed in order to estimate all maintenance processes beginning with the maintenance organization of the power plant operator, planning of activities, time scheduling, preparation and coordination of work as well as work execution. In addition, the coordination of work of all subcontractors and service providers should be also analyzed.

An exact estimation of the on-site maintenance activities based on costs and schedule is possible by comparison of the planned costs and activities with the actual expenditure and time needed to perform the maintenance.

Such procedures mentioned above have been partly established in different nuclear power plants. To achieve a large cost reduction great efforts have to be made to implement and use the identified cost-cutting measures obtained from the process analyses performed in advance. Through detailed time scheduling and planning of resources, optimization of maintenance organization and technical conditions as well as focusing on few competent subcontractors the expenditure can be cut down in order to ensure cost reductions.

Generally, the benefit for the customer is:

long-term reduction of the maintenance expenditure and

effective utilization of the comprehensive maintenance experience and know-how.

2.5 Performance of valve maintenance as main contractor

Economically, cost-effective performance of valve maintenance means performing and bundling of all individual activities under the umbrella of an experienced service provider which act as a main contractor. The great advantage of such a contract for the customer is to order the complete maintenance service on a fixed price which is lower as the total cost obtained from a large number of individual services. Competent and reliable subcontractors which are familiar with special features of components are involved to ensure good quality of work. Under such conditions the mutual cooperation can preserve the maintenance competence and know-how to ensure the safety and availability as long as the nuclear power plant is in service. Effective improvements based on process analyses can also be implemented.

Siemens can provide an integrated single-source concept for comprehensive valve maintenance which ensures long-term security of supply for all the services offered.

Essential valve maintenance activities of Siemens Nuclear Power GmbH are shown in Figure 11. The maintenance was successfully performed in cooperation with competent subcontractors and plant operators based on experience and know-how of the service provider Siemens Nuclear Power GmbH.

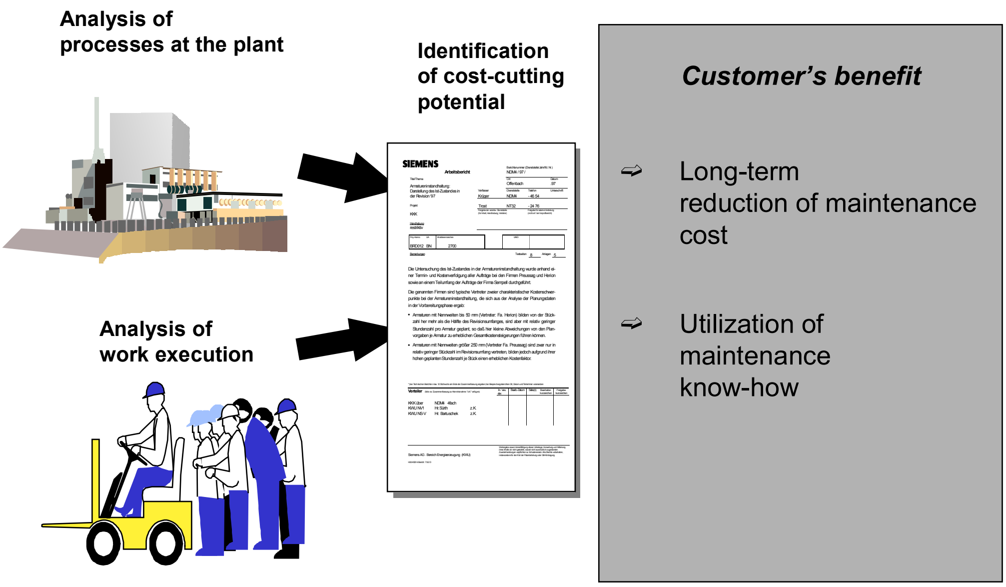

FIG. 9. Process analyses identification of cost-cutting potential.

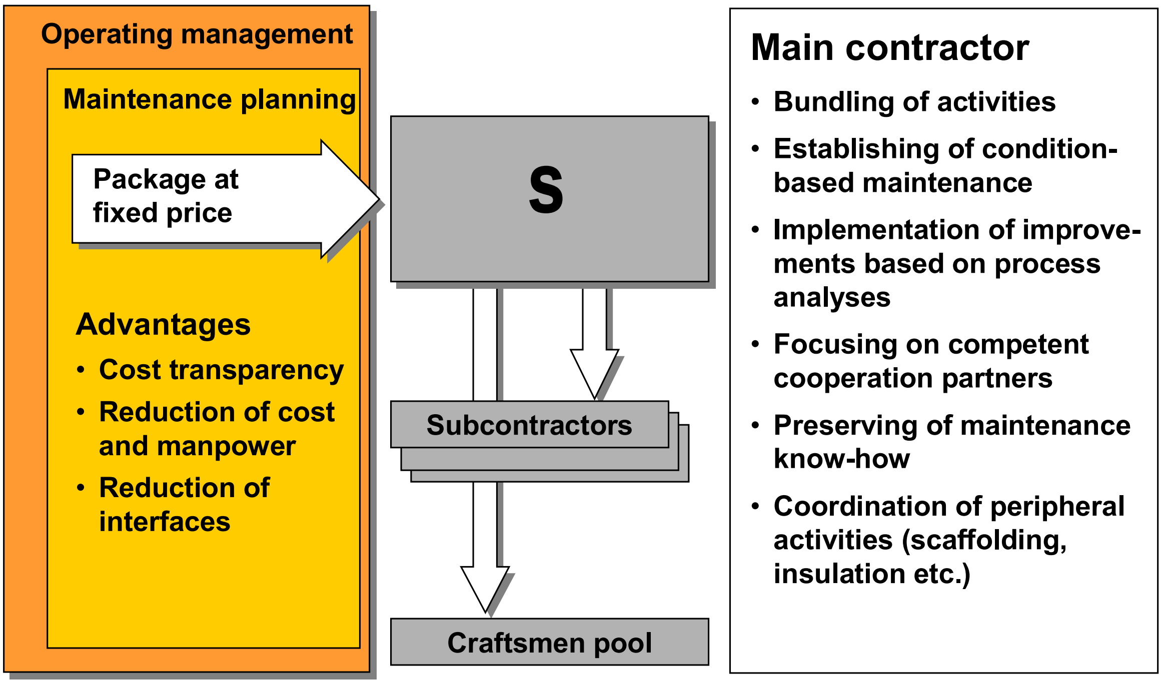

FIG. 10. Management of valve maintenance performance of valve maintenance as main contractor.

Germany	Europe
• Emsland	• Garoña (Spain)
• Grohnde	• Cofrentes (Spain)
• Isar 1+2	• Vandellos (Spain)
• Kruemmel	• Asco (Spain)
• Brunsbuettel	• Almaraz (Spain)
• Biblis A+B	• Leibstadt (Switzerland)
• Phillipsburg 1+2	• Beznau (Switzerland)
• Neckarwestheim 1+2	• Oskarshamn (Sweden)
• Obrigheim	• Ringhals (Sweden)
• Gundremmingen	• Tihange (Belgium)
	• Loviisa (Finland)

FIG. 11. References valve maintenance activities performed.

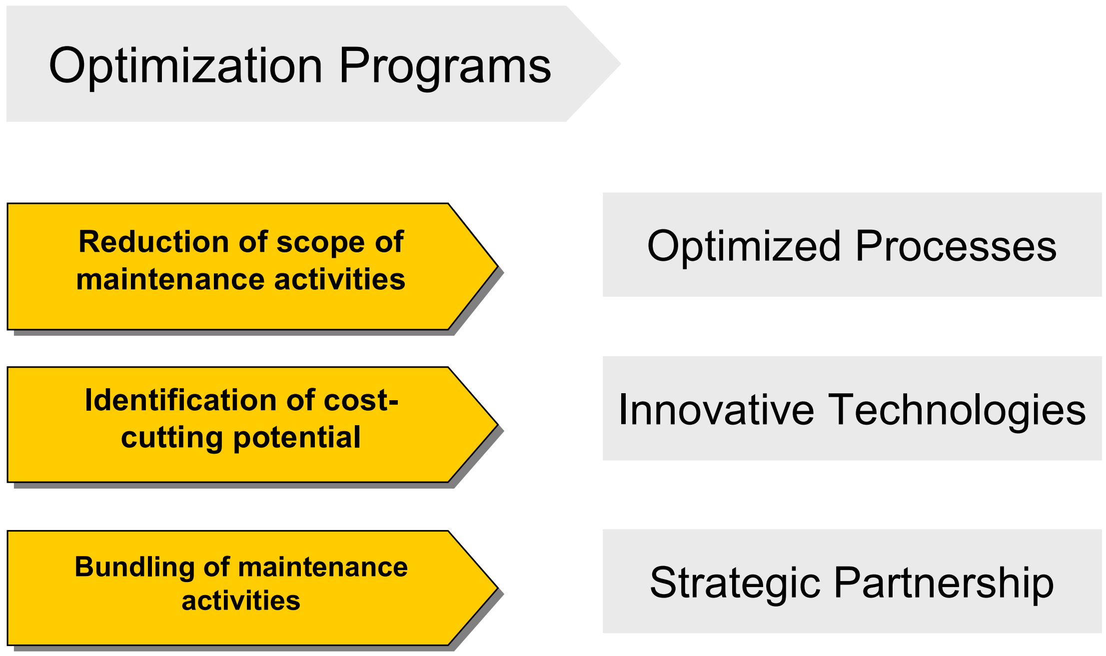

FIG. 12. Outlook.

3. SUMMARY AND CONCLUSIONS

Today, Siemens Nuclear Power GmbH already has a number of innovative tools and service packages available for future nuclear power plant maintenance. The potential for cost reduction in the field of maintenance of nuclear power plants has been exploited thanks to the use of optimized processes and these innovative technologies. The maintenance services provided by Siemens Nuclear Power GmbH enabled to make an important contribution to this success.

Siemens Nuclear Power GmbH is able to offer expert solutions to implement cost-effective, extensive maintenance packages for the nuclear power plant maintenance. The aim is to ensure the ongoing technological development of maintenance activities while achieving considerable cost reductions, through a partnership between power plant operators and service providers. Siemens Nuclear Power GmbH offers a range of innovative service packages for the maintenance of the future, one key example of such innovative service package shown in detail was the integrated valve maintenance.

Siemens' intensive efforts in this field are aimed at a helping drive for the technical and economic optimization of future maintenance activities at nuclear power plants, while ensuring that high safety levels are maintained.

LAGUNA VERDE NUCLEAR POWER PLANT:
AN EXPERIENCE TO CONSIDER IN ADVANCED BWR DESIGN

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Abstract

Laguna Verde is a BWR 5 containment Mark II. Designed by GE, two external re-circulation loops, each of them having two speed re-circulation pump and a flow control valve to define the drive flow and consequently the total core flow an power control by total core flow. Laguna Verde Design and operational experience has shown some insights to be considering in design for advanced BRW reactors in order to improve the potential of nuclear power plants. NSSS and Balance of plant design, codes used to perform nuclear core design, margins derived from engineering judgment, at the time Laguna Verde designed and constructed had conducted to have a plant with an operational license, generating with a very good performance and availability. Nevertheless, some design characteristics and operational experience have shown that potential improvements or areas of opportunity shall be focused in the advanced BWR design. Computer codes used to design the nuclear core have been evolved relatively fast. The computers are faster and powerful than those used during the design process, also instrumentation and control are becoming part of this amazing technical evolution in the industry. The Laguna Verde experience is the subject to share in this paper.

1. INTRODUCTION

Laguna Verde Unit One began commercial operation in 1990 about fourteen years after the original design was developed.

During this period many changes happened in the nuclear industry that were incorporated in Laguna Verde design and operational philosophy. The line of production of the fuel design of the first core was out of production. The first reload was designed with new fuel design that was commercially available in the industry at that spot in time. New load strategies were developed in the industry that change the control rod pattern interchange, this means the control cell core (CCC), the spectral shift was also adopted to improve the fuel bum up efficiency, the total core flow window to operate at full power was introduced, the window includes a range from 87% to 107% rated core flow, this means an extension of the power flow map was adopted by using ELLLA and ICF. In order to satisfy the power demand the availability of each unit were increased and the fuel cycle length were extended from 12 to 18 months, including operational flexibilities such as cycle coast-down. Such changes require an increase in the design power density for the reloads, this implies a complication in satisfying thermal limits, hot excess reactivity and shutdown margin, even the fuel exposure limit was a restriction for reload design, considering the energy expectations, the reload design process required more advanced fuel design.

A power up-rate process was implemented for both units that were operating commercially at 1931 MW·th , to increase up to 2027 MW·th . This process was an opportunity to have, not only a higher capacity but to update the plant design using more advanced computer codes and methodologies than those used in the original design process.

The introduction of operational flexibilities and new design characteristics are demanding changes in the knowledge, skills and procedures that operator must have, including the operation in an extended power flow map related to the original design of the power flow map.

2. OPERATIONAL FLEXIBILITIES

2.1. ELLLA

Among the first set of operational flexibilities introduced in the Laguna Verde Nuclear Power station we found the extension of the power flow map by using ELLLA, this operational flexibility introduces a lower limit in the total core flow window, used to operate at full power. This mode of operations requires besides changes in the license including technical specifications changes, specific operators training. The lower limit of 87% rated core flow requires partial closure of the re-circulation flow control valve. This closure of the flow control valve has at least two consequences, the first one is related to the increase in the turbulences in the re-circulation pipes and fittings downstream of the control flow valves that shall face different conditions than those assumed in the original design process. The turbulent flow induced vibrations are generating an unexpected performance in the re-circulation system, drain pipes attached to the valves were broken and their design has to be modified, isolation valves downstream of the flow control valves were showing an unexpected performance with degradation in such way that the unit has to be dated during the final part of the fuel cycle, the design change in the isolation valves of the recirculation system was also introduced.

The second consequence is related to the status change in the operational alarms in the control room panels. The original design did not take into consideration during normal operation at full power the presence of alarms such as rod block alarm, that was originated in operation at the lower limit of the total core flow window and full power, this alarm is derived from the neutron noise originated from the boiling process in a high core average void content in the moderator system.

Operators are requiring measures to avoid operations in conditions such that alarms associated to or derived from normal operating conditions can appear, operating conditions that are having a potential decreasing in the operational margins must be avoided. Operators prefer to have a control room with panels free of alarms derived from the normal operation.

The degradation of the performance of the isolation valves in the re-circulation loop conducted to anticipate the potential operation in a single loop, this is another flexibility that was not assumed in the original design. This mode of operation requires specific operator training and also changes in the license and in the technical specifications.

2.2. ICF

The other flexibility to extend the power flow map was the introduction of the named Increased core flow. This flexibility allows operation at full power with a window of total core flow beyond the 100% rated core flow and until 107% rated core flow. This flexibility required a specific evaluation of the flow induced vibrations in the reactor vessel internals and in the re-circulation loop. This flexibility introduces a change in the performance of the fuel allowing the extension of the fuel cycle. However transient scenarios are changing imposing more restrictions in the thermal limits. In order to evaluate the flow-induced vibrations, extended data must be collected during the startup test program, also specific instrumentation and very oriented testing must be used. This flexibility can have impact on the life span of reactor vessel internals, the upper limit for the total core flow window, in Laguna Verde this upper limit was selected such that it will not have impact on the life span.

2.3. Stability

As the core design considers higher power density, the operations maneuvers during the startup, mainly during shift speed of the re-circulation pump can conduct to a situation of potential instabilities, such as density power oscillations.

Laguna Verde unit one during a startup after a scram in cycle four, experienced such kind of instabilities, with peak to peak of the order of 10% of rated power, this unexpected performance, required an analysis of the core stability in the power flow map. Consequences such reduction of the operational area in the power flow map was established, changes to the procedures for shift speed of re-circulation pump were launched. Specific operator training was developed.

The new design concepts under feasibility studies are considering variable speed in the re-circulation pump. This concept requires evaluation of different transients scenarios that those analyzed in the original design, because of the consequences of re-circulation flow transient having different impact in the core performance during transients and also in LOCA analysis.

Such kind of changes also will impact the peak clad temperature, and consequently thermal limits will be modified, if such is the case.

Nuclear reactor stability has been an issue since the beginning of nuclear reactor design, however earlier stability evaluations made using computer codes based on the frequency domain with simplifications did not identify the potential instability associated wit specific scenarios. Nuclear industry made effort to have most accurate calculation tools, and also developed new systems to satisfy the criteria of detect and suppress. Utilities are evaluating the use of such systems that are requiring in some cases a complete change in systems like average power monitor, rod block monitor to include the oscillation power monitor.

Actually there is under feasibility study the introduction of stability monitoring system, based on identification and suppress concept.

Stability issue is an example of industry issues that are requiring more powerful, robust and best estimated computational codes, to make a permanent and a systematic evaluation program of the nuclear reactor performance.

2.4. Fuel design

As was earlier mentioned the increase in the capacity factors, and the extension of fuel cycles from 12 to 18 months, are demanding advanced fuel design. The fuel design must also consider that the main condenser material (Cu-Ni alloy) is imposing some restrictions in during startup following a reload period. The new design must have higher bum up limit, lower thermal limits, higher power density and seismic qualification required form the seismic criteria for Laguna Verde. Laguna Verde is located in a high seismic acceleration zone, and the combined loads form LOCA — Seismic, requires specific qualifications for fuel design.

Introduction of the new fuel design with different geometric characteristics also requires neutronic and thermal hydraulic compatibility with the fuel design remaining form the previous generation.

Evaluation of the power flow map lines were evaluated to avoid any of the earlier problems that operators were facing, such as stability, and other maneuvers in the control rod adjustment maneuvers

Laguna Verde had introduced the use of zinc and noble metals in the reactor water chemistry, this new environmental for the fuel are having impact on the water control chemistry, operational procedures and care operators must have during start up and others operational maneuvers. It is obvious that the new fuel design must be compatible with the water chemistry practices and capabilities. Main condenser system, condenser de-mineralizer beds are having impact on the reactor water chemistry and also on the control cooper contained in the reactor water. This design is under evaluation for future maintenance of the main condenser, feasibility studies are going to be developed in order to change the main condenser material, but other materials could have other restrictions in the fuel design, or operational restrictions during power adjustment.

2.5. Power up-rate

Laguna Verde nuclear power plant performed the design evaluation to re-dimension the reactor power generation capability, raising from 1931 MW·th to 2027 MW·th . This was an opportunity to reevaluate the design, update the methodologies and the calculation tools used in the process. The challenge was to increase the power capability without decreasing any safety margin and avoiding any plant backfiting or modification. The following identifies, not all but some of the systems and the areas of opportunity that Laguna Verde would like to share.

First of all the fuel design was identified as a potential area of change to avoid the restrictions imposed by the new power capability and plant capacity factors required by operational programs developed to satisfy the power demand.

The main turbine and turbine control valves, were analyzed to identify capability to mange the new steam generation rate from the nuclear reactor, control capability of the turbine control valves, considering that the pressure in the reactor vessel was unchanged related to the original design pressure. For such purposes tests were successfully performed to show the capability of the control system.

Reliability of the safety relief valves were developed to show that the new steam generation capability has not impact in the relief system. Re-circulation system and the performance and capability of the pumps and control system were evaluated, changes in the pump performance were identified, it also has impact in the heat balance.

New computer codes related to those used in the original design were used to evaluate ATWS and LOCA analysis. The new computer codes required new methodology, however the process show that the tools used in the design process and the criteria used for the original design were having relatively large margins derived from uncertainties, this large margin allows us to up-rate the power capability without changing safety margins. The use of best estimated computer codes in constant development in the industry, combined with a continuous evaluation of the design will give us additional power capabilities, nevertheless some additional restrictions would also appear.

Main condenser has two boxes in serial arrangement, cooled by sea water, and it is having some restrictions mainly during hot sea water season. From my personal point of view this is one of the main challenges not only in the nuclear industry, the availability and the performance of the heat sink shall include advanced features in design. Thermal efficiency and economic performance are highly dependent on the heat sink. Parallel or serial arrangements, tube pitch and arrangement, heat transfer area and pipe availability, auxiliary systems such as cold condensed water aspersion, materials research are area of opportunity to have more efficient systems.

2.6. Spent fuel pool

As the energy requirements are satisfied there are other problems emerging, the original design of the spent fuel pool was a standard one, however the national program of the spent fuel management is waiting for more advanced technologies in the industry, the potential accumulation of the spent fuel discharged from the reactors are demanding temporal storage, Laguna Verde modified the spent fuel pool design, using the high density racks.

The actual capacity of each spent fuel pool is 3177 spent fuel assemblies — about 28 equilibrium cycle reloads. The new capability of the spent fuel pool required at least evaluation to assure sub-criticality, and heat load mainly when the full core is discharged from the core. Also every new fuel design is requiring an evaluation of the impact on the spent fuel capability and safety. Both spent fuel pools are completely independent, and this design concept has advantages, form the safety point of view. However there are some restrictions from the fuel management point of view. Since both units started up in different time and the independence of spent fuel pools make difficult to use some of the low burned fuel assemblies available of unit one to load in the first cycle of unit two, the need for neutron sources can not be avoided during start-up of unit two. Additionally fuel management became more restricted due to the fact that not all the highest reactivity fuel assemblies available at the site can be used in the reload. This means that the interchange of fuel assemblies between the spent fuel pool of both units must be evaluated in the new advanced reactor multiple units sites.

Spent fuel pool capacity and interchange in multiple units sites constitutes an area of opportunity in the advanced reactors design for countries developing slowly their final spent fuel management program, in order to give enough time to analyze and evaluate the feasibility of potential technologically and economically solutions to this issue.

2.7. Instrumentation and control

Laguna Verde Unit two started three fuel cycles after the unit one, during the construction process and start-up program of unit one some of the components of unit two were used as spare parts of unit one, for that reason during construction program of unit two such components were required but such components were out of the market, companies closed, or new advanced systems were developed for such purposes. That was the case of the Transverse In core Probe (TIP), Source (SRNM) and Intermediate range (IRNM) neutron monitoring systems.

The original design for TIP includes neutron detector and collecting information, however in the market were available an automatic TIP with Gamma detector, using computerized automatic data collecting system, that makes faster the total core power range monitor calibration, and faster startups. This concept improves the capacity factor of the plant.

The original concept for neutron monitoring includes three systems, Source range, Intermediate range and Power range. During start-up the operator has to select and operate each system in the adequate range manually. That set of systems evolved to a new concept considering only the Wide range and the Power range. Wide range substitutes Source and intermediate and became computerized, this also makes faster start-ups and improve capacity factors.

The set of new systems were installed in unit two and are actually operating showing the operational and unit availability advantage.

The non-programmed SCRAMS during the start-up program of unit two became significantly reduced in comparison to unit one.

As the neutron monitoring components in unit one approach to their end of life, substitution for spare parts are almost impossible, new concept design and computerized must be considered.

The use of computerized systems are imposing different failure modes and consequences of the failures must be analyzed to assure that they are not endangered the safety margins of the design. Quality Control of software design, are evolving as the potential of errors are identified. Computer platforms and software and firmware are evolving really fast, component and system substitution must be taken into account in the advanced design in order to have an amicable management program to update such kind of systems.

2.8. Core monitoring system

The core monitoring system originally was based on a Honeywell computer, the program for core monitoring system was based on the TIP readings and Local Power reading detectors, interpolation of the corrected reading were used to evaluate neutron flux and power distributions and from that thermal limits were derived. This first concept has not had prediction capability to evaluate some adjustment power manoeuvres.

Simulators based on the diffusion theory were evolving and the computer platforms made reality the use of such programs with restrictions, however they include the prediction capability.

Nowadays faster computers and more robust and powerful programs considered as best estimated are available to monitoring and predicting the core performance. Substitution of one monitoring system for an advanced system is becoming a challenge not only for management of life components but also for engineering and replacement, this is due to the interfaces among different systems. This evolution and retirement of the market of components or systems must be addressed in the advanced nuclear reactors design.

2.9. Components out of service

Nuclear power plants are often facing situations in which a component or a system is not available or is out of service, and the original design requires de-rating the unit or shutdown within a certain period of time. There is the possibility to evaluate the performance of the unit in a scenario that is not having the component or the affected system out of service, to show that there is not safety impact or to identify the restrictions to be applied in such operational scenario, and by this analysis to apply for a license exception in order to continue in line until the programmed shutdown or when spare parts are available.

This temporal permit to operate with equipment out of service is becoming a standard practice to perform the core analysis and reload design and licensing. We found among others SRV OOS, MSIV OOS, SLO, EOC RPTOOS.

It is very important that such potential operational scenarios must be taken into account from the beginning of the advanced nuclear reactor design process.

3. CONCLUSIONS

The experiences of the operating units are giving insights that must be taken into account in advanced nuclear power reactor design process.

Operating units experience is also showing that every concept in design must be considered in at least two perspectives, one is the local system benefits and problems associated, the second is related to the integral unit and plant related systems and components that will experience impact derived from the specific local system.

Evolution of the operational flexibilities must be evaluated and considered from the beginning of the design process to take into consideration the integral impact on the systems associated in the unit.

Developing new computer tools and best-estimate computer programs and the methodologies associated with them must be used systematically to identify safety margins, or potential restrictions derived from the use of old ones that did not identify restrictions at the moment of the original evaluation.

Experience shows that out of market components requires a new approach for process such as: design, built, maintenance, engineering and life component management for the new and advanced nuclear reactor design.

Nuclear industry in some countries are requiring support and time to evaluate technically and economically different options to launch their spent fuel management program. Adequate design criteria for the temporal storage capability in the spent fuel pools must be developed for advanced reactors.

Finally, it is very important to emphasize that nuclear industry and specially nuclear reactor must be under permanent and systematic evaluation, using not only the standard methods and computer codes, the nuclear community must support development of new computing tools, integration of qualified evaluation teams to assure that all aspects and specially all safety concern of the nuclear industry are permanently.

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DEVELOPMENT OF NEW DESIGNS AND TECHNOLOGIES WITH A FOCUS ON PERFORMANCE AND ECONOMIC VIABILITY

(Session II)

Chairperson

E. PATRAKKA
Finland

US DEPARTMENT OF ENERGY NUCLEAR ENERGY RESEARCH INITIATIVE

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Abstract

This paper describes the Department of Energy's (DOE's) Nuclear Energy Research Initiative (NERI) that has been established to address and help overcome the principal technical and scientific issues affecting the future use of nuclear energy in the United States.

1. BACKGROUND

In January 1997 the President tasked his Committee of Advisors on Science and Technology (PCAST) to evaluate the current national energy R&D portfolio and provide a strategy to ensure the USA has a program to address the nation's energy and environmental needs for the next century.

In its November 1997 report, the PCAST Panel on Energy Research and Development determined that establishing nuclear energy as a viable and expandable option was important and that a properly focused R&D effort to address the potential long-term barriers to expanded use of nuclear power (e.g. nuclear waste, proliferation, safety and economics) was appropriate. The PCAST panel further recommended that DOE reinvigorate its nuclear energy research and development activities in an R&D effort to address these potential barriers with a new Nuclear Energy Research Initiative (NERI). This new initiative should fund research based on competitive selection of proposals from the national laboratories, universities and industry.

The 1999 PCAST report on International Cooperation on Energy Innovation recommended that an international component to NERI be created to promote bilateral and multilateral research focused on advanced technologies for improving the cost, safety, waste management, and proliferation resistance of nuclear fission energy systems.

The Department endorsed the PCAST recommendation and received appropriations for NERI beginning in FY 1999, to sponsor new and innovative scientific and engineering R&D. In addition to the established NERI program, the Department proposed in FY 2001 to launch a new initiative within NERI, the International Nuclear Energy Research Initiative (I-NERI), to provide for cooperative international research and development of new technologies to address the key issues affecting the future of nuclear energy. I-NERI will give the United States and DOE greater credibility and influence in international discussions regarding the future implementation of nuclear technologies. It will allow us to leverage international resources, foster international cooperation, and work with countries on research already underway on a variety of advanced reactor types and proliferation-resistant fuel cycles. To achieve this long-range goal, the following objectives have been established:

- Develop advanced reactor and fuel cycle concepts and scientific breakthroughs in nuclear technology to overcome the principal scientific and technical obstacles to expand future use of nuclear energy in the United States, including issues involving nuclear material proliferation, unfavorable economics, and nuclear waste disposition;

- Advance the state of US nuclear technology to maintain a competitive position in overseas markets and a future domestic market;
- Promote and maintain a nuclear science and engineering infrastructure to meet future technical challenges.

NERI sponsors innovative scientific and engineering research and development in the following areas:

- Proliferation-resistant reactors and fuel cycles;
- New reactor designs with higher efficiency, lower-cost, and improved safety to compete in the global market; low output power reactors for use where large reactors are not attractive;
- Advanced nuclear fuels;
- New technologies for management of nuclear waste; and
- Fundamental nuclear science.

The NERI/I-NERI programs feature a *competitive, peer-reviewed R&D selection process to fund researcher initiated R&D proposals* from universities, national laboratories and industry.

The NERI research areas are also of critical importance to foreign countries with civilian nuclear programs. I-NERI encourages foreign participation with US institutions to help maintain the nuclear option worldwide and to leverage scarce research dollars.

Finally, the Department established an independent Nuclear Energy Research Advisory Committee to provide advice and recommendations to assist the Department on the direction and focus of its energy R&D activities.

2. FY 1999 ACCOMPLISHMENTS:

- The initial NERI procurement was completed with the award and issuance of grants and laboratory cooperative agreements for 47 R&D projects involving research participants from 45 US universities, laboratories and industrial organizations, and 11 foreign collaborating organizations.

- Initiated innovative scientific and engineering R&D for 46 NERI projects to enhance the performance, efficiency, reliability, proliferation resistance, and economics of future nuclear power systems.

3. FY 2000 ACCOMPLISHMENTS:

- Advanced the state of scientific knowledge and technology to enable incorporation of improved proliferation resistance, safety and economics in the design and development of advanced reactor and nuclear fuel systems through the award of ten new R&D projects.

- Continued the second phase of research for 45 continuing R&D projects awarded in FY 1999 to improve the scientific and technical understanding of new reactor and fuel cycle concepts and nuclear waste technologies, and the underlying fundamental science.

4. FY 2001 PLANNED ACCOMPLISHMENTS:

- Complete the first three-year round of NERI research and development by identifying feasible and important reactor and fuel cycle concepts for continued development.

- Establish the International Nuclear Energy Research Initiative (I-NERI) to promote bilateral research to improve the cost, enhance the safety, non-proliferation and waste of future nuclear energy systems.

5. PROGRAM BUDGET (in millions)

FY 1999 **Appropriation**	FY 2000 **Appropriation**	FY 2001 **Appropriation**
$19.0	$22.5	$35.0 (includes $7M for I-NERI)

DEVELOPMENT ACTIVITIES ON ADVANCED LWR IN ARGENTINA

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Abstract

CAREM, an Argentinean project, consists on the development, design and construction of a small Nuclear Power Plant. CAREM is an advance reactor conceived with new generation design solutions and standing on the large experience accumulated in the safe operation of Light Water Reactors in the world. The CAREM is an indirect cycle reactor with some distinctive features that greatly simplify the reactor and also contribute to a high level of safety: integrated primary cooling system, self-pressurized, primary cooling by natural circulation and safety system relying on passive features. In this paper a brief description of the CAREM distinctive features and associated development activities are presented.

1. INTRODUCTION

Argentine Nuclear Development started in early fifties. Initially the activities of the Comisión Nacional de Energía Atómica (CNEA) of Argentina were oriented to research in nuclear physics, radiochemical studies, material science among others subjects. In 1957, the CNEA decided to build a Research Reactor. The RA-1 was the first nuclear reactor to be put in service in South America. Since then, Argentina has designed and constructed several Research Reactors in Argentina and another countries, and at the present competes with foreign developed countries as supplier of this technology.

In 1964, CNEA initiated the feasibility study for the construction of Atucha I Nuclear Power Plant (CNA I) which would be the first nuclear power plant in Argentina and Latin America designed for electric power generation. In 1967 entrusted its design and construction to Siemens. The construction began in June 1968 and the commercial operation started in June 1974. The station contains a reactor of the pressure vessel type and it is heavy water moderated and cooled being of the PHWR type; it is periodically refueled on power. CNA I 's original design considered only natural uranium as fuel, being its electric power of 340 MW(e). The station suffered two essential modifications that improved its performance:

- In 1977 the electric power was increased to 357 MW(e).
- Since 1995 a progressive loading with slightly enriched uranium (0.85% wt) began, so that at present the core contains not only natural uranium fuel elements but also slightly enriched ones.

In 1967, CNEA initiated the feasibility study for the construction of Embalse Nuclear Power Plant (CNE) and in 1973 signed a contract with Atomic Energy of Canada Limited (AECL) and Societa Italiani Impianti P.A. (IT) for a 600 MW(e) CANDU–PHW (pressurized heavy water) type nuclear power plant. The construction of the station began in May 1974 and the commercial operation started in January 1984.

On the other hand, Argentina started the design of its own nuclear power plant, CAREM. The CAREM concept was first presented in March 1984 in Lima, Peru, during the IAEA conference on small and medium size reactor. CAREM design criteria or similar ones have since been adopted by other plant designers, thus originating a new generation of reactor design, of which the CAREM was, chronologically, one of the first. The Argentinean CAREM project consists on the development, design and construction of an advanced, simple and small Nuclear Power Plant (NPP) conceived with new generation design solutions and standing on the large world wide experience accumulated in the safe operation of Light Water Reactors. This project allows Argentina to sustain activities in the nuclear power plant design area, assuring the availability of updated technology in the mid-term. This implies working with technology acquired in Research Reactors design, construction and operation, and Pressurized Heavy Water Reactors (PHWR) Nuclear Power Plant operation as well as developing advanced design solutions.

2. FEATURES OF CAREM DESIGN

CAREM is an indirect cycle reactor (100 MWt, approximately 27 MW(e)) with some distinctive features that greatly simplify the reactor and also contribute to a high level of safety:

The CAREM NPP is a light water integrated reactor. The whole high-energy primary system and the absorbers rods drive mechanisms are contained inside a single pressure vessel.

The flow rate in the reactor primary systems is achieved by natural circulation. The driving force for the coolant's natural circulation is produced by the location of the steam generators above the core.

Self-pressurization of the primary system is the result of the liquid-steam equilibrium.

The main criteria used in the design of safety systems were simplicity, reliability, redundancy and passivity.

2.1. Integrated primary cooling system

The CAREM reactor pressure vessel (RPV) contains the core, steam generators, the whole primary coolant and the absorber rods drive mechanisms (figure 1). The RPV diameter is about 3.2 m and the overall length is about 11 m.

The core has 61 fuel elements of hexagonal cross section and about 1.4 m active length. Each fuel element contains 108 fuel rods, 18 guide thimbles and 1 instrumentation thimble. Its components are typical of the PWR fuel assemblies. The fuel is enriched UO_2. Core reactivity is controlled by the utilisation of Gd_2O_3 as burnuble poison in specific fuel rods and movable absorbing elements belonging to the Adjust and Control System. Chemical shim is not used for reactivity control during normal operation. Absorbing elements belonging to the Fast Extinction System are used to produce the sudden interruption of the nuclear chain reaction when required. Each neutron absorbing element is a cluster composed of a maximum of 18 individual Ag-In-Cd rods which are put together in a single unit. Each unit fits well into the fuel assembly guide tubes.

The primary coolant is light water that also acts as moderator. The large coolant inventory provides relatively smooth transients and large response time for recovery actions. Strong negative temperature coefficients are the consequence of no boron use for reactivity control during normal operation, allowing the control system to keep control of the reactor power through transients and load variations with minimum control rod motion.

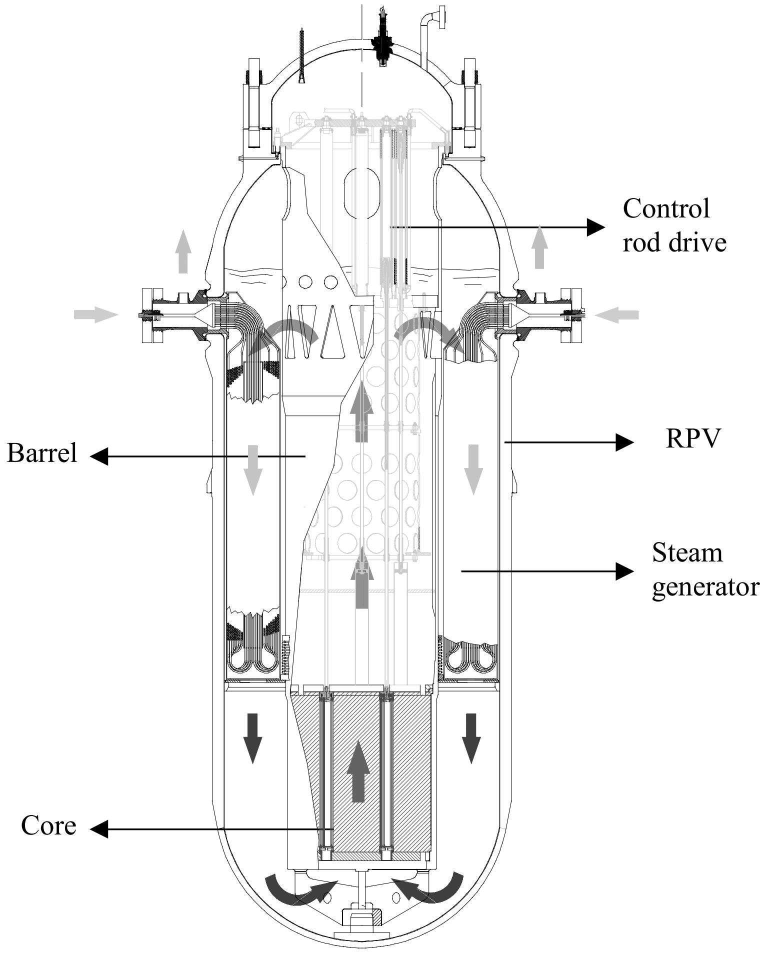


FIG. 1. CAREM primary circuit

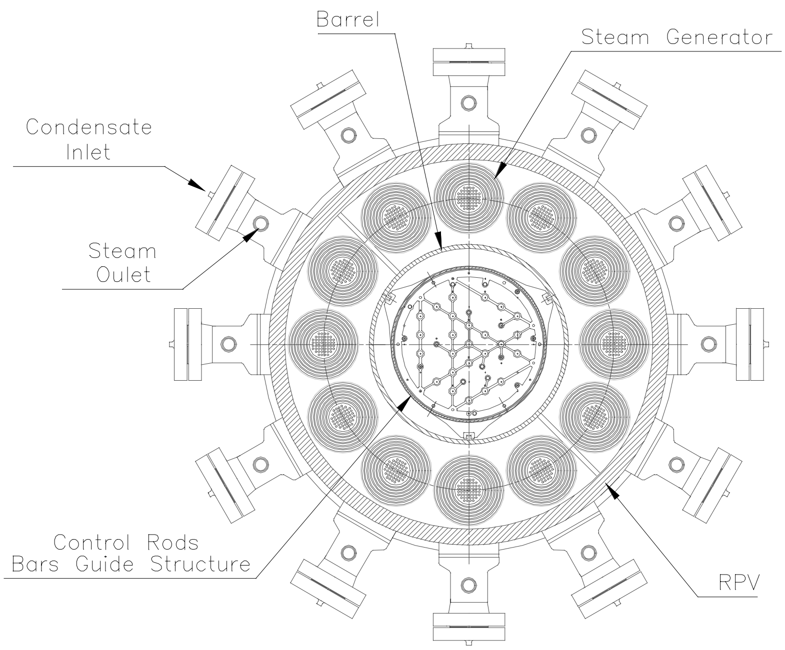


FIG. 2. Steam generators layout inside the RPV

The CAREM steam generators are twelve identical 'Mini-helical' of the "once-through" type placed equally distant from each other along the inner surface of the RPV area (figure 2). The secondary system circulates upwards within the tubes, while the primary does so in counter-current flow. An external shell surrounding the outer coil layer and adequate seals form the flow separation system, that guarantees that the entire stream of the primary system flows through the steam generators. In order to achieve a rather uniform pressure-loss and superheating on the secondary side, the number of tubes per coil layer is changed to equalize the length of all the tubes. Thus, the outer coil layers will hold a larger number of tubes than the inner ones. For safety reasons, steam generators are designed to withstand the pressure from the primary up to the steam outlet / water inlet valves even without pressure in the secondary.

The control rod drives (CRD) are of the hydraulic type. They are wholly contained in the RPV avoiding the use of mechanical shafts passing through the primary pressure boundary. Rods are kept in position by an external hydraulic circuit that pumps water to a lower chamber of a piston/cylinder assembly. Absorbing elements belonging to the Fast Extinction System are kept in the upper position during all normal operation and, at that position, the piston partially closes the outlet orifice and reduces the flow to a leakage. The CRD for this system is designed using a large gap between piston and cylinder in order to obtain a minimum dropping time. The inner wall of the cylinder and the outer wall of the piston of the CRD of the Adjust and Control System have grooves which produces changes in the pressure losses depending on their relative position. A steady flow keeps the piston in a fixed position and stepwise movement is achieved by applying pressure/flow pulses. It is designed to guarantee that each rod can be moved step by step so manufacturing and assemblies' allowances are stricter and clearances are narrower. In this case there is not a stringent requirement on dropping time. Both devices type, perform their SCRAM function by the same principle: "rod drops by gravity when flow is interrupted", so malfunction of any powered part of the hydraulic circuit will cause the immediate shutdown of the reactor.

2.2. Natural circulation

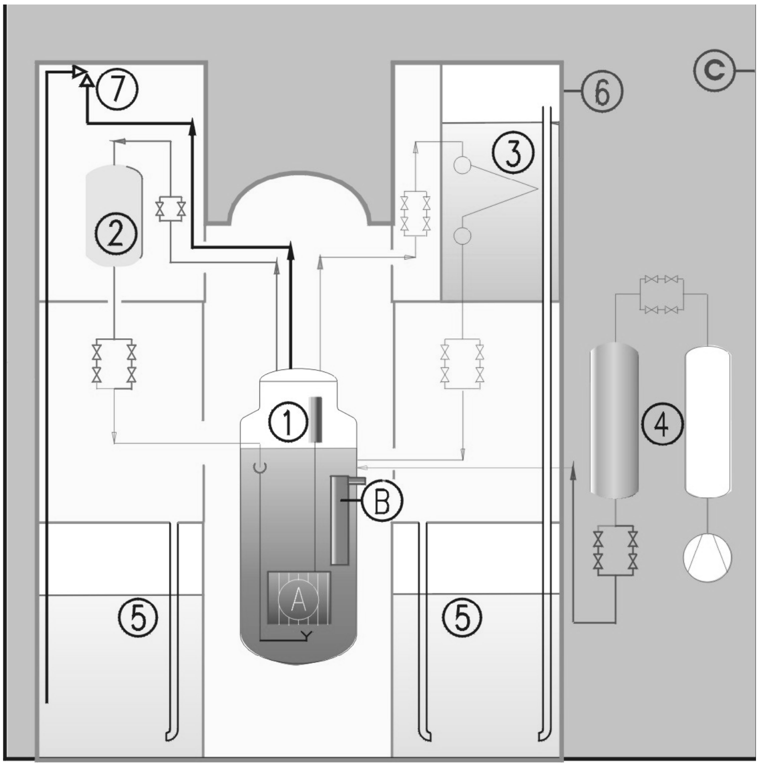
The flow rate in the reactor primary systems is achieved by natural circulation. Figure 1 shows a diagram of the natural circulation of the coolant in the primary system. Water enters the core from the lower plenum. After been heated, the coolant exits the core and flows up through the riser to the upper dome. In the upper part, water leaves the riser through lateral windows to the external region. Then it flows down through modular steam generators, decreasing its enthalpy. Finally, the coolant exits the steam generators and flows down through the down-comer to the lower plenum, closing the circuit. The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form losses, producing a flow rate in the core that allows to have a sufficient thermal margin to critical phenomena. Core internals are designed to minimize the pressure drops. The natural circulation of the coolant produces different values of the flow rate in the primary system according to the power generated (and removed). Under different power transients a self-correcting response in the flow rate is obtained. CAREM has not primary pumps therefore there is not possibility of Loss of Flow Accident (LOFA).

2.3. Self pressurization

Self-pressurization of the primary system is the result of the liquid-steam equilibrium. Due to self-pressurization, bulk temperature at core outlet corresponds to saturation temperature at primary pressure. The steam dome pressure is very close to the saturation pressure, and at all the operating conditions this has proved to be sufficient to guarantee a remarkable stability of the RPV pressure response. The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in case of power ramps. The large volume of the integral pressurizer also contributes to the damping of eventual pressure perturbations. Heaters and sprinkles typical of conventional PWRs are thus eliminated.

2.4. Passive safety systems

The main criteria used in the design of safety systems were simplicity, reliability, redundancy and passivity. Special emphasis has been put on minimizing the dependence on active components and operators' actions (figure 3).



SAFETY SYSTEMS

1- First shutdown system
2- Second shutdown system
3- Residual heat removal system
4- Emergency injection system
6- Containment
7- Relief valves

REFERENCES

A- Core
B- Steam generator
C-Secondary containment
5- Suppression pool

FIG. 3. CAREM safety systems

The First Shutdown System (FSS) is designed to shut down the core, when abnormal or deviated from normal situations occur, and to maintain the core sub-critical during all shutdown states. This function is achieved by dropping the neutron-absorbing elements into the core by the action of gravity when the water flow in the CRD mechanism is interrupted, so malfunction of any powered part of the hydraulic circuit will cause the immediate shutdown of the reactor. Six out of twenty-five absorbing elements are part of the Fast Extinction

System capable of shutdown the reactor immediately. The rest belongs to the Adjust and Control System capable to introduce enough negative reactivity to keep the reactor in shutdown mode, with appropriate safety margin, during all cooling conditions.
Gravity driven injection system of borated water at high pressure makes up the Second Shutdown System. It actuates automatically when the Reactor Protection System detects the failure of the First Shutdown System or in case of LOCA. The system consists of tanks connected to the reactor vessel by two piping lines which valves are opened automatically when the system is triggered. Then one of the pipes -from the steam dome to the upper part of the tank- equalizes pressures, and the other -from a position below the reactor water level to the lower part of the tank- discharges the borated water into the primary system by gravity.

The Residual Heat Removal System has been designed to reduce the pressure on the primary system and to remove the decay heat in case of loss of heat sink. It is a simple and reliable system that operates condensing steam from the primary system in emergency condensers. The emergency condensers are heat exchangers consisting of an arrangement of parallel horizontal U tubes between two common headers. The top header is connected via piping to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. The condensers are located in a pool filled with cold water inside of the containment building. The inlet valves in the steam line are always open, while the outlet valves are normally closed, therefore the tube bundles are filled with condensation. When the system is triggered, the outlet valves open automatically. The water drains from the tubes and steam from the primary system enters the tube bundles and is condensed on the cold surface of the tubes. The condensation is returned to the reactor vessel forming a natural circulation circuit. In this way, heat is removed from the reactor coolant, and simultaneously water is supplied to the reactor vessel. During the condensation process the heat is transferred to the water of the pool by a boiling process. This evaporated water is then condensed in the pressure suppression pool of the containment.

The Emergency Injection System prevents core exposure in case of LOCA. The system consists of tanks with borated water connected to the RPV. In the event of such accident, the primary system is depressurized with the help of the emergency condensers and at low pressure the rupture disks break starting the RPV flooding with borated water.

Three safety relief valves protect the integrity of the reactor pressure vessel against overpressure, in case of strong unbalances between the generated and removed power.

The pressure-suppression type primary containment is a cylindrical concrete structure with an embedded steel liner type with two major compartments: a drywell and wetwell. The lower part of wetwell volume is filled with water that works as the condensation pool, and the upper part is a gas compression chamber. The blow-down pipes from the safety relief valves are routed to the pressure suppression pool.

3. RESEARCH AND DEVELOPMENT

The CAREM project involves technological and engineering solutions, as well as several innovative design features that have been properly proved during the design phase. Within CAREM project, the effort was focused mainly on the nuclear island (inside containment and safety systems) where several innovative design solutions require developments. This comprises mainly: the reactor core cooling system, the reactor core and fuel assembly, the reactor pressure vessel internals and the hydraulic control rod drive mechanisms.

The utilisation of specific codes (modelling tools) is required to obtain design parameters of some important systems (e.g. primary cooling system, reactor core, etc.). In some cases these codes must be developed and/or validated with experimental data to build confidence on their results.

A High Pressure Natural Circulation Loop (CAPCN) was constructed and operated to produce data in order to verify the thermal hydraulic tools used to design CAREM reactor. The CAPCN reproduces the dynamics phenomena of the CAREM primary cooling system, except for the three-dimensional effects. Dynamical experiment data are being used to test our numerical procedures and codes [1].

The neutronic and thermal-hydraulic behaviors of CAREM are strongly coupled, so to take this effect into account, the neutronic code CITVAP and the 3D two fluid model THERMIT code were linked. Benchmarck data available worldwide were used to validate the neutronic data, codes and modeling. Ad-hoc experiments, to generate a substantial database in the operate range and fuel geometry of the CAREM core, to develop a prediction methodology for critical heat flux, were performed at the Institute of Physics and Power Engineering (Obninsk, Russian Federation).

Hydrodynamic and structural test are being conducted to qualify the fuel assembly design.

The development of the hydraulic control drive for the Fast Extinction System and the Adjust and Control System comprises different stages. First, a series of test using prototypes to determine preliminary operating parameters were finished. After that, a series of test were and are being conducted in the Low Pressure CRD Rig (CEM) to characterize the mechanism and the hydraulic circuit used to control the mechanism movement. Now, a High Pressure CRD Rig (CAPEM) is being designed and it will be used to qualify the mechanism at the CAREM RPV operation conditions including some test under abnormal conditions and to test the drive mechanism instrumentation developed for determining the position of the absorber elements in the core.

Related to mechanical design (structural, dynamic, seismic, etc.) of the core and other reactor pressure vessel internals, different mock-up facilities are being constructed. Among others, the evaluation of manufacturing and assembly process for the “Mini-helical” steam generators is being done using mock-up. The design of the cinematic chain of the First Shutdown System is of particular interest. A dummy of the core, up to the extension of three fuel assemblies, core support and upper structure including control rod guides were used to study the behavior of the control rod movement (stepwise and rapid fall) under horizontal seismic load on a wide range of frequencies and magnitudes. One vertical full-scale model of the control rod drive structure with the absorbing elements and a dummy fuel assembly were used to study the static/dynamic friction loads of the cinematic chain for different insertion/extraction velocities of the absorbing elements and different misalignment of the structure.

4. CONCLUSIONS

The development activities on advanced LWR in Argentina are carried on in connection with CAREM, a reactor conceived with new generation design solutions and standing on the large experience accumulated in the safe operation of Light Water Reactors in the world.

Technical and economical advantages are obtained with the CAREM design compared to the traditional design. The large loss of coolant accident was eliminated due to the absence of large diameter piping associated to the primary system and the rod ejection accident was also eliminated due to the development of the innovative hydraulic mechanisms located completely inside the reactor pressure vessel. In addition, hydraulic control rod drives mechanism significantly cost down compared with the traditional PWR's control rod drive mechanisms. The large coolant inventory in the primary results in large thermal inertia and long response time in case of transients or severe accidents and the large water volume between the core and the wall leads to a very low fast neutron dose over the reactor pressure vessel wall. The elimination of chemical shim (borated water) during normal operation reduces the maintenance and occupational doses and the operational costs and results in strong negative temperature coefficients that simplify the operation control. The integration of the primary system inside a single pressure vessel reduces the shielding requirements, by the elimination of gamma sources of dispersed primary piping and parts and the elimination of primary circuit branch pipes systems reduces the primary circuit failure probability. The elimination of primary pumps and pressurizer results in lower costs, added safety, and advantages for maintenance and availability.

Improvements of the reliability of CAREM reactor come partly from the use of natural circulation in the primary circuit.

Since CAREM is a LWR conceived with new generation design solution, an extensive research and development program related with the innovative design solutions are been carried on to properly verify them during the design phase.

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DEVELOPMENT, OPERATING EXPERIENCE AND FUTURE PLAN OF ABWR IN JAPAN

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Abstract

From 1974 to 1978 nuclear power plant's capacity factor decreased to low level in Tokyo Electric Power Company (TEPCO) for stress corrosion cracking of stainless steal pipe, thermal fatigue crack of feed water spurger and vibration trouble of LPRM, etc. So, for improving that situation, we gathered most useful BWR technology in the world at that time and we started ABWR project to develop ideal BWR. The goals of this development were (1) Enhanced safety and reliability, (2) Reduced occupational radiation exposure and radioactive waste, (3) Enhanced operability and maneuverability and (4) Improved economy. Significant improvements were induced by the adoption of the reactor internal pump, fine motion control rod drive mechanism, integral type reinforced concrete containment vessel and digital control system. After about 10 years R&D, we succeeded the development of the compact and economic plant, ABWR (1356MW(e)) whose building volume had been reduced to 76%, compared with previous BWR (1100MW(e)). The construction of Kashiwazaki-Kariwa Nuclear Power Plant unit No. 6 that is the first ABWR plant in the world was started September 1991, and that of unit No. 7 was started February 1992. We used effective construction methods, such as large block construction method and all weather construction method, to shorten the construction duration from first concrete to commercial operation start to 51.5 months. The commercial operation start of Kashiwazaki-Kariwa Nuclear Power Plant unit No. 6 was November 1996 and that of unit No. 7 was July 1997. After that, unit No. 6 had three times of annual outage and unit No. 7 had two times of annual outage. Both plants have good operating experiences. From commercial operation start to end of August 2000, the capacity factor of unit No. 6 is 86.4% and that of unit No. 7 is 86.9%. Now, in Taiwan, ABWR plants are under construction at Lungmen unit No. 1 and 2. In Japan, ABWR plants are under construction at Hamaoka unit No. 5, Shika unit No. 2, and in plan at Ohma unit No. 1, Shimane unit No. 3. TEPCO is now progressing cost reduction of ABWR with other Japanese utilities. The most significant point is standardization. And we also plan to adopt some design improvements. By using these measures, TEPCO intends 30% cost reduction of our next ABWR plants, compared with Kashiwazaki-Kariwa No. 6 & 7 of ABWR first plant. We would like to further develop ABWR for moreover cost reduction. And we start to develop ABWR- II as a large-scale (1700 MW(e)) next generation reactor in aim to commence plant operation in the late 2010s.

1. DEVELOPMENT OF ABWR

1.1. Reason of development

Tokyo Electric Power Company (TEPCO)'s first nuclear power plant (Fukushima-Daiichi unit No. 1) operation was started in 1971 and second plant (Fukushima-Daiichi unit No. 2) operation was started in 1973. From 1974 to 1978 nuclear power plant's capacity factor decreased to low level in TEPCO for stress corrosion cracking of stainless steal pipe, thermal fatigue crack of feed water spurger and vibration trouble of LPRM, etc. So, for improving that situation, we gathered most useful BWR technology in the world at that time and we started ABWR project to develop ideal BWR. From 1978 to 1979 TEPCO carried out feasibility study for ABWR with GE, Toshiba, Hitachi, ASEA-ATOM and Ansaldo Mechanica Nucleare of Italy. That was called AET (Advanced Engineering team) study. Upon receipt of the results of the feasibility study, TEPCO decided to proceed with development of the ABWR. From 1981 to 1985 TEPCO carried out basic design study for ABWR with GE, Toshiba, Hitachi and test & development study for ABWR with Japanese BWR utilities, GE, Toshiba and Hitachi.

The development of the ABWR was adopted as one of the main tasks of Japan's Third Light Water Improvement and Standardization Program. The verification test of the internal pump was assigned by the government to the Nuclear Power Engineering Corporation and was completed. Also, various researches in connection with the ABWR were undertaken jointly by BWR utilities.

After about 10 years study, we succeeded the development of the compact and economic plant. Milestone of ABWR development is shown in Figure 1.

1.2. Design of ABWR

The goals of this development were (1) Enhanced safety and reliability, (2) Reduced occupational radiation exposure and radioactive waste, (3) Enhanced operability and maneuverability and (4) Improved economy. Significant improvements were induced by the adoption of the reactor internal pump, fine motion control rod drive mechanism, integral type reinforced concrete containment vessel and digital control system. Cross-section of reactor buildings for 1100MW(e)-class BWR and ABWR are described in Figure 2.

1.2.1. Reactor internal pump (RIP)

The reactor coolant recirculation pump, located outside of the reactor pressure vessel (RPV) in BWR/1-6, has been installed in the RPV. This eliminates recirculation piping outside the vessel, resulting in improvement in operability and safety. The reactor core will remain

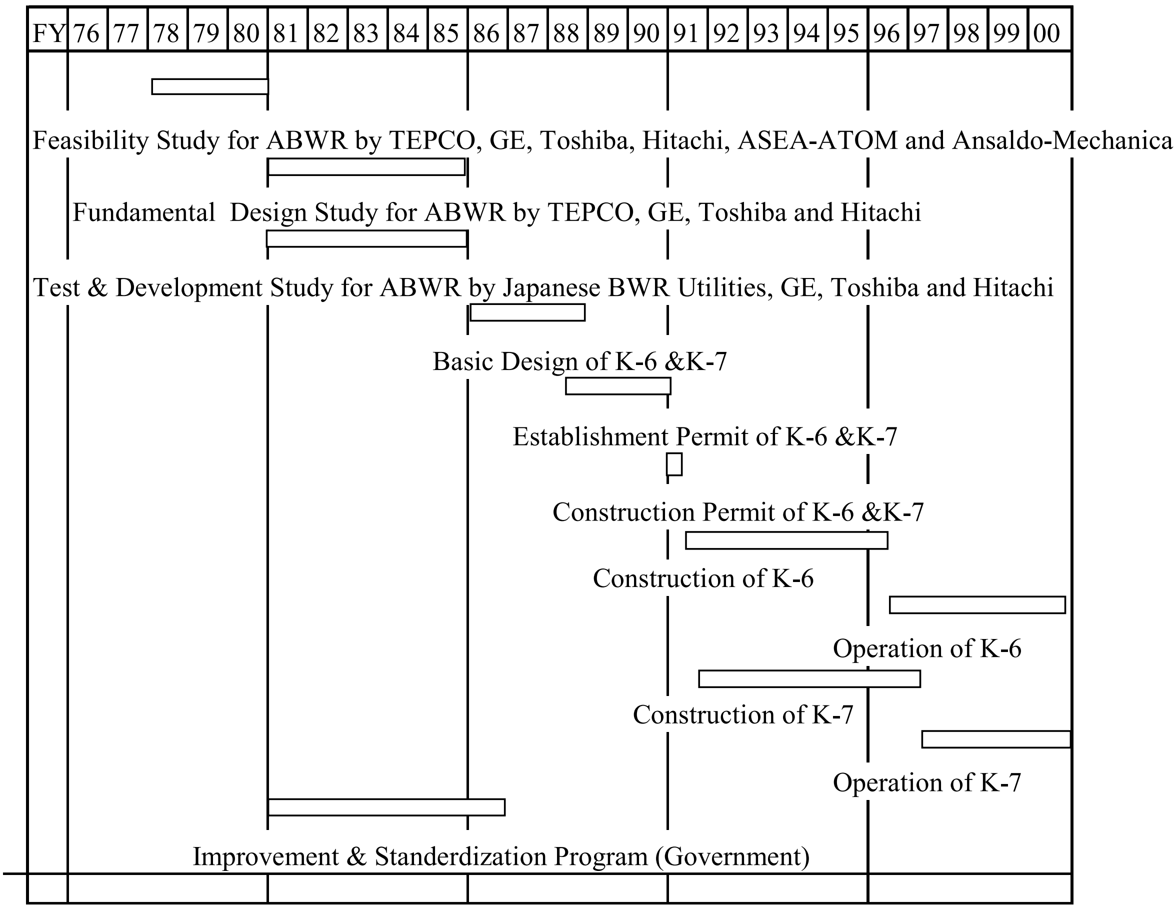

FIG. 1. Milestone of ABWR development.

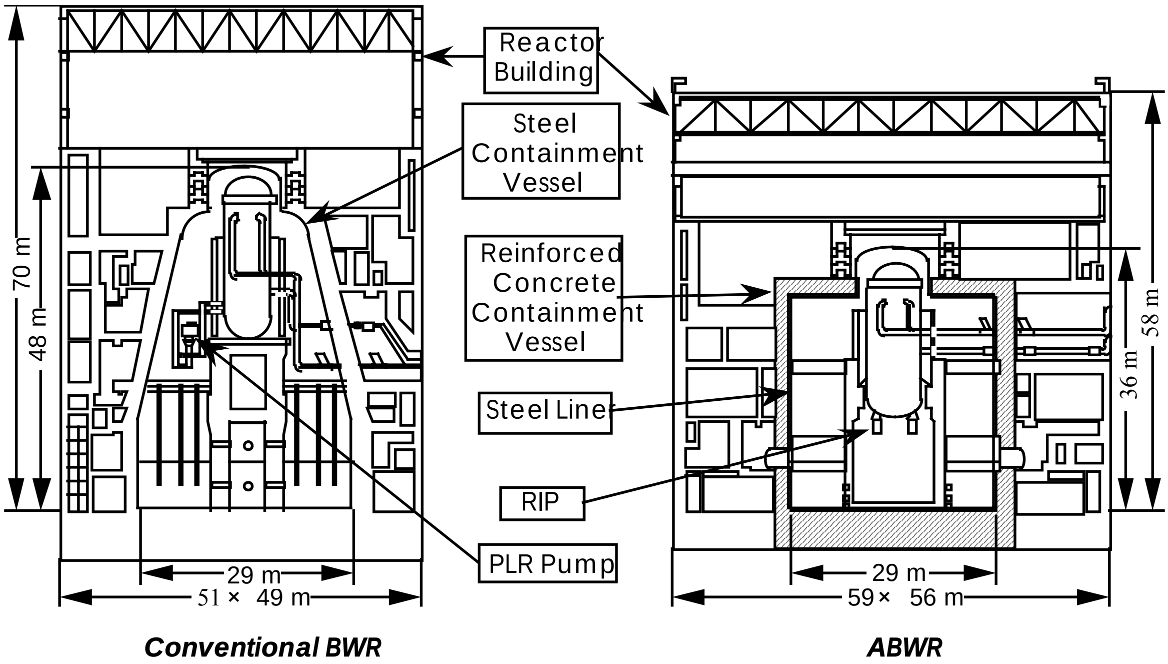

FIG. 2. Cross-section of reactor buildings for 1100MW(e)-class BWR and ABWR.

covered with water during accident conditions. Wet motor without shaft seals decreases probability of leakage. Primary containment and reactor building are made more compact by the elimination of recirculation piping. The adoption of small inertia reactor internal pump and solid-state, adjustable speed drive systems improve controlability. We reduce in-service inspection and radiation exposure through elimination of recirculation piping. Lowered the center of gravity of the RPV improved aseismic strength.

1.2.2. Fine motion control rod drive mechanism (FMCRD)

The ABWR incorporates the FMCRD, which enables fine control rod position adjustment by using an electric motor. It also retains a traditional hydraulic system for the scram function, and the dual drive capability increases safety. Simultaneous operation of up to 26 control rods reduces the time required for plant start-up. ABWR adopts simplified and optimized hydraulic system, so one Hydraulic Control Unit (HCU) scrams 2 control rods. Split type housing makes the seal inspection easy. That reduced occupational radiation exposure and shortened refueling and maintenance outage period. Diversity of CRD methods and bayonet coupling improve reliability. A joint study program confirmed normal operation characteristics; scram characteristics; durability for 40-year operation; and scram function during earthquake.

1.2.3. Reinforced concrete containment vessel (RCCV)

Adopted for the primary containment vessel of the ABWR, the RCCV is less limited in terms of shape than a steel structure. A highly safe and economical RCCV is used for the reactor containment. Employment of reactor internal pumps has eliminated piping for the reactor coolant recircurating system. This, in turn, has enabled the center of gravity of the reactor building to be lowered, making the building more resistant against earthquakes. And integration of the RCCV and the reactor building has increased resistance against deformation resulting from internal pressure in the event of an accident. Local loads applied to the reactor containment in an accident have also been successfully reduced. Taking advantage of this feature enabled greater aseismic strength and more rational equipment layout inside the RCCV, which improves maintenance work efficiency.

1.2.4. ABWR type main control panels

ABWR control panels have been developed for improved operability at nuclear power plants. The new control panels use optical multiplex transmission systems and an overall digital control system for improved controllability and reliability. They also offer improved supervisory and operating functions through application of state-of-the-art data processing technologies and human engineering knowledge. Full-scale mock-ups of the control panels were built and reviewed by operators for monitoring capability and operability before putting the panels into use. So, ABWR type main control panels have advanced man-machine interface. Expanded use of automated systems reduced operator's work and advanced display technology enable accurate monitoring of entire plant status. Expansion of digital control systems and optical multiplexing signal transmission network make instrumentation and control system to high reliability and easy maintenance.

1.2.5. Turbine

The ABWR uses a large turbine with last stage buckets of 52 inches. The exhaust loss of the turbine is lower than that of the turbines with 41-inch buckets used in 1,100MW(e) plants, resulting in an improvement of more than 25MW in electrical output. Use of the longer buckets, together with moisture separator reheaters and heater drain pump forward systems, has improved plant thermal efficiency from 33.4% to 34.5%. A single piece of forged material fabricated from an ingot of 630 tons is used for the rotor of the turbine to assure its reliability.

After about 10 years R&D, we succeeded the development of the compact and economic plant, ABWR (1356MW(e)) whose building volume had been reduced to 76%, compared with previous BWR (1100MW(e)).

2. CONSTRUCTION EXPERIENCE OF KASHIWAZAKI-KARIWA UNIT NO. 6 & NO. 7

The construction of Kashiwazaki-Kariwa Nuclear Power Plant unit No. 6 that is the first ABWR plant in the world was started September 1991, and that of unit No. 7 was started February 1992 (Table I). We used effective construction methods, such as large block construction method and all weather construction method, to shorten the construction duration from first concrete to commercial operation start to 51.5 months (Table II).

TABLE I. CONSTRUCTION RECORD OF KASHIWAZAKI-KARIWA UNIT NO.6&7

Unit No.	6	7
Start of Construction	9/17/1991	2/3/1992
Inspection of Founding	7/29/1992	3/16/1993
Center Mat Finish	1/29/1993	9/30/1993
RPV on	8/23/1994	5/19/1995
RPV First Hydrostatic Test	5/18/1995	4/26/1996
Fuel Loading	11/30/1995	10/10/1996
Start of Commercial Operation	11/7/1996	7/2/1997

TABLE II. ABWR PLANT DATA

	BWR-5 (KK-Unit 1,2,3,4,5)	ABWR (KK-Unit 6,7)
Generator output Generator frequency Reactor type Reactor thermal output Main steam flow Feed water temperature Reactor Pressure Vessel • Design pressure • Operating pressure • Operating temperature • Height • Internal diameter Primary Containment Vessel Type	1,100 MW(e) 50 Hz BWR5 3,293 MWt 6,410 t/h 216 °C 8.62 MPa [87.9 kg/cm 2] 6.93 MPa [70.7 kg/cm 2] 286 °C 23 m 6.4 m • Self-Standing Steel • Containment • (Unit 1 : Mark-II • Unit 2, 3, 4, 5 • : Improved Mark-II)	1,356 MW(e) 50 Hz ABWR 3,926 MWt 7,640 t/h 216°C 8.62 MPa [87.9 kg/cm 2] 7.07 MPa [72.1 kg/cm 2] 287 °C 22 m 7.1 m • RCCV (Reinforced Concrete Containment Vessel with Liner)
Turbine • Capacity • Steam pressure • Steam temperature • Last stage turbine blades • Turbine bypass capacity	 1,100MW 6.55 MPa [66.8 kg/cm 2] 282°C 41 inch • Unit 1 : 25 % • Unit 2, 5 : 100 % • Unit 3, 4 : 37.5 %	 1,356MW 6.69 MPa [68.2 kg/cm 2] 284°C 52 inch 33 %
Generator • Capacity • Voltage	 1,300 MVA 19 kV	 1,540 MVA 27 kV
Core • Average power density • Core flow • Number of fuel assemblies • Type of assembly Number of Control Rods	 50.0 kW/liter 48,300 t/h 764 High burn-up 8×8 fuel 9×9 fuel (Type A, Type B) 185	 50.6 kW/liter 52,200 t/h 872 High burn-up 8×8 fuel 9×9 fuel (Type A, Type B) 205
Coolant Recirculation (Number of Pumps) Control Rod Drive Normal Scram Emergency Core Cooling System (ECCS) Residual Heat Removal System Turbine Cycle	External Recirculation Pump (2) Jet Pump (20) Hydraulic Locking Piston Drive Hydraulic Piston Drive Division I:LPCI + LPCS Division II:LPCI + LPCI Division III:HPCS ADS 2 Divisions TC 6F-41" (Non-Reheat)	Internal Pump (10) Fine Motion Electric Motor Drive Hydraulic Piston Drive Division I: RCIC + LPFL(RHR) Division II: HPCF + LPFL(RHR) Division III: HPCF + LPFL(RHR) ADS 3 Divisions TC 6F-52" (2-Stage Reheaters)

3. OPERATING EXPERIENCE OF KASHIWAZAKI-KARIWA UNIT NO. 6 & NO. 7

The commercial operation start of Kashiwazaki-Kariwa Nuclear Power Plant unit No. 6 was November 1996 and that of unit No. 7 was July 1997. After that, unit No. 6 had three times of annual outage and unit No. 7 had two times of annual outage. Both plants have good operating experiences. From commercial operation start to end of August 2000, the capacity factor of unit No. 6 is 86.4% and that of unit No. 7 is 86.9%. Records at annual outage of Kashiwazaki-Kariwa unit No. 6 & No. 7 are shown in Table III. Major inspection items of annual outage are listed in Table IV. Operating performance of ABWR is shown in Table V.

TABLE III. ANNUAL OUTAGE

Unit	Time	Date	Period (Days)	RadiationExposure (Man Sv)	Radioactive waste drums
6	1st	11/20/1997 -1/19/1998	55+6 (6 holidays at new year)	0.30	54
	2nd	3/13/1999 -4/25/1999	44	0.33	53
	3rd	6/12/2000 -8/3/2000	53	0.34	tallying now
7	1st	5/27/1998 -7/20/1998	55	0.15	62
	2nd	9/18/1999 -11/1/1999	45	0.30	53

TABLE IV. MAJOR INSPECTION ITEMS OF ANNUAL OUTAGE

Component			Total Num-ber	Mainte-nance Cycle	Unit 6			Unit 7	
					1st Outage	2nd Outage	3rd Outage	1st Outage	2nd Outage
Reactor System Compo-nent	Reactor Internal Pump		10	5 Outage	2	2	2	2	2
	FM CRD	Main Body	205	25%/10 Outage	3	5	3	11	3
		Spool Piece	205	10 Outage	21	21	20	21	21
	Replacement of LPRM		52	10 Outage	7	7	7	7	7
	Safety Relief Valve		18	Every Outage	18	18	18	18	18
	Main Steam Isolation Valve		8	Disassemble: 4 Outage	2	2	2	2	2
				Leak Test: Every Outage	8	8	8	8	8
	New Fuel Loading		872	-	180	240	204	156	192
Turbine System Component	Main Turbine		4 casing	2 Outage	4	2	2	4	2
	Moisture Separator Reheater		2	2 Outage	2	1	1	2	1
	Main Valve	Main Stop Valve	4	Every Outage	4	4	4	4	4
		Control Valve	4	Every Outage	4	4	4	4	4
		Combination Inter-cept Valve	6	Every Outage	6	6	6	6	6

TABLE V. OPERATING PERFORMANCE OF ABWR

Unit	FY	Capacity factor	Scram (Unplanned automatic shut down)	Unplanned manual shut down	planned shut down	Radioactive waste drums	remarks
6	1996	100.0	0	0	0	25	
	1997	83.0	0	0	1	69	
	1998	93.5	1	0	1	26	
	1999	90.1	1	0	0	106	
	2000	(56.2)	0	1	1	(72)	-Aug.31
	total	86.4	2	1	3	298	
7	1997	100.0	0	0	0	37	
	1998	84.5	0	1	1	62	
	1999	73.9	0	1	1	71	
	2000	(100.0)	0	0	0	(5)	-Aug.31
	total	86.9	0	2	2	175	

4. SITUATION OF ABWR

Now, in Taiwan, ABWR plants are under construction at Lungmen unit No. 1 and 2. In Japan, ABWR plants are under construction at Hamaoka unit No. 5 and Shika unit No. 2, and under licensing at Ohma unit No. 1. Other 8 ABWR plants are at planing stage. Subsequent ABWR Plans are listed in Table VI.

TABLE VI. SUBSEQUENT ABWR PLANS

	Country	Plant	Utility	Out put	CS	OS	Stage
	Taiwan	Lungmen unit No. 1	Taiwan Power Co.	1371 MW	1999/3	2004/7	under construction
	Taiwan	Lungmen unit No. 2	Taiwan Power Co.	1371 MW	1999/3	2005/7	under construction
	Japan	Hamaoka 5	Chubu EPCO	1358 MW	1999/3	2005/1	under construction
	Japan	Shika 2	Hokuriku EPCO	1358 MW	1999/9	2006/3	under construction
	Japan	Ohma 1	Electric Power Development Co	1383 MW	2002/3	2007/7	under licensing
	Japan	Shimane 3	Chugoku EPCO	1373 MW	2002/3	2009	at planing
	Japan	Fukushima Daiichi 7	Tokyo EPCO	1380 MW	-	-	at planing
	Japan	Fukushima Daiichi 8	Tokyo EPCO	1380 MW	-	-	at planing
	Japan	Higashidori 1 (Tokyo)	Tokyo EPCO	1385MW	-	-	at planing
	Japan	Higashidori 2 (Tokyo)	Tokyo EPCO	1385MW	-	-	at planing
	Japan	Kaminoseki 1	Chugoku EPCO	1373 MW	-	-	at planing
	Japan	Kaminoseki 2	Chugoku EPCO	1373 MW	-	-	at planing
	Japan	Higashidori 2 (Tohoku)	Tohoku EPCO	1385MW	-	-	at planing

CS: Construction Start OS: Operation Start

5. NEXT PLAN OF ABWR

Under the deregulatory circumstances of power market, nuclear power is required to be economically competitive for other power source. TEPCO is now progressing cost reduction of ABWR with other Japanese utilities. We adopt three approaches. The first approach is standardization, the second is design improvement that is useful for cost reduction, and the third is intelligent management. By using these measures, TEPCO intends 30% cost reduction of our next ABWR plants, compared with Kashiwazaki-Kariwa No. 6 & 7 of ABWR first plant.

5.1. Standardization

The most significant point is standardization. Standardization is consist of design standardization, document standardization and quality management standardization.

The scope of design standardization is standardization of system design, layout design in the reactor building and the turbine building, and embedded level of the reactor building and the turbine building. That standardization enables to omit most engineering. Our base is expansion of standardization except of items concerning with sight unique landform and seismic conditions. We only need to check sight unique engineering.

Document standardization is concerning to kinds and contents of documents, which are required to plant manufacturers by utilities. System design specification, equipment design specification, process flow diagram, equipment drawing, piping drawing and so on are covered by documents. Document standardization is connected with design standardization. When the system design is same, the system document also is same. So we only need to check sight unique documents.

5.2. Design improvement

We plan to adopt some design improvements. They are simplification of reactor internal pump power supply system, change of FMCRD motor from stepping motor to induction motor, change of process computer from special ordered computer to work station computer, down sizing of radioactive dispersal system, change of flammability control system (FCS) from thermal recombiner to passive autocatalytic recombiner, size up safety relief valve and so on.

5.2.1. Simplification of reactor internal pump power supply system

Ten reactor internal pumps are located at the bottom of the reactor pressure vessel. In order to simplify the power supply system for the reactor internal pumps, a new inertia-increased reactor internal pump was developed, which allows eliminating the Motor-Generator (M-G) sets. The rotating inertia was increased approximately 2.5 times of current reactor internal pump inertia by addition of flywheel on its main shaft.

5.2.2. Change of FMCRD

Shaft sealing type of usual FMCRD was gland-packing type. To eliminate leak potential of FMCRD, We invent seal-less type FMCRD. That has magnetic coupling. And at same time we adopt induction motor instead of stepping motor for FMCRD.

5.2.3. Change of process computer

The development of workstation computer is remarkable. So, we can use workstation computer instead of special ordered computer as process computer.

5.2.4. Down sizing of radioactive waste treatment system

We made a systematic analysis on the quantity of waste actually produced and conducted a series of surveys on the chemical and physical properties of liquid waste. The survey on the quantity and quality of waste found that RW facility has a sufficient reserve margin of capacity for the quantity of wastes actually produced at the plants and that liquid waste from the plant is cleaner than the design water quality of such waste from the plant.

So we reduce the design treatment volume of high conductivity liquid waste treatment system.

5.2.5. Change of flammability control system (FCS)

We are planning to change FCS from thermal recombiner to passive autocatalytic recombiner. Passive autocatalytic recombiner doesn't need blower, electric power supply and heating sources. So, that has cost performance.

5.2.6. Size up safety relief valve

Capacity of current safety relief valve was 395T/H. Large-scale safety relief valve (460T/H) was developed through R &D. Adopting that valve we can reduce number of safety relief valves from 18 to 16.

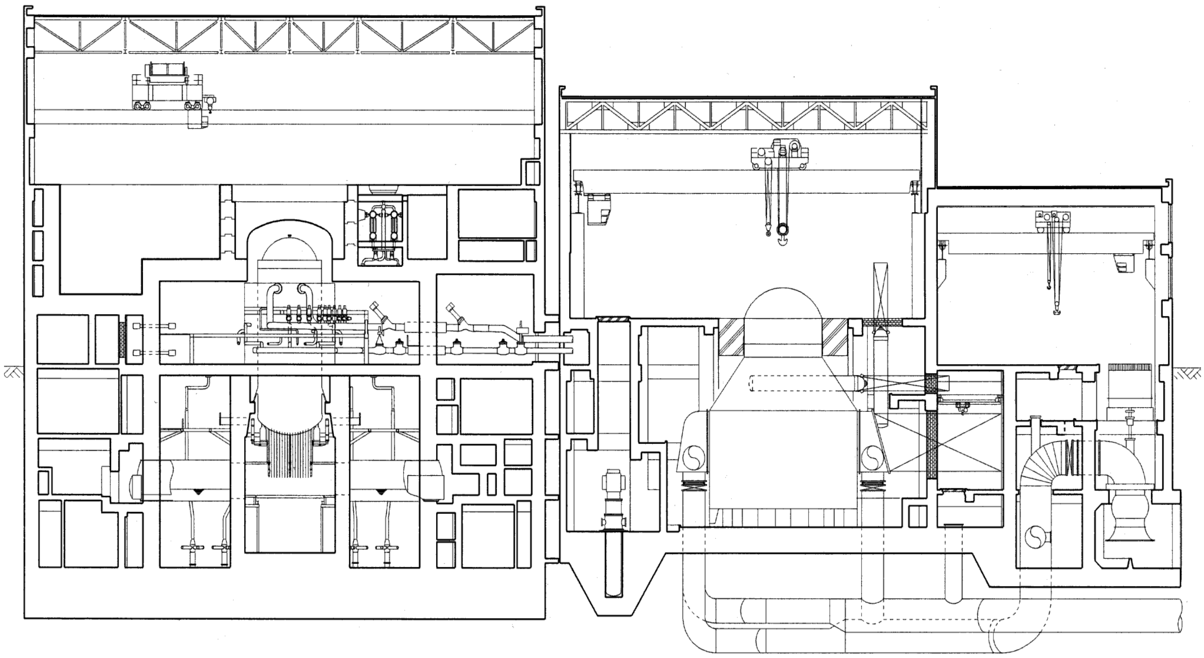

FIG. 3. ABWR-II design features.

5.3. Intelligent management

When we construct a nuclear power plant, we must manage many kind of works these are licensing works management, engineering management, procurement management, construction schedule management and so on. We analyze construction works of Kashiwazaki-Kariwa Unit No. 6&7, and develop intelligent project management program. That program is effective for cost reduction of next plant. And making use of these electrical data in operation and maintenance will be effective for plant management during plant life cycle.

6. ABWR- II

A preliminary concept design of ABWR- II has been developed in aim to commence plant operation in the late 2010s. The designs developed so far are as follows: higher level of safety and reliability by adopting passive systems and design considering severe accidents, improvement in economy by up-rating the power to 1700 MW(e), rationalizing components and reducing the amount of materials, and improvement in operability and maintainability by implementing on- line maintenance. ABWR- II design features are shown in Figure 3.

7. CONCLUSION

We developed ABWR for realizing the ideal light water reactor. The world's first ABWRs were constructed at Kashiwazaki-Kariwa nuclear power station units No. 6&7. The commercial operation and the annual outage experiences of them were very satisfactory. During this period, we have accumulated valuable know-how and data for the operation and maintenance of ABWRs. We would like to further develop ABWR for moreover cost reduction. And we start to develop ABWR- II as a large-scale (1700 MW(e)) next generation reactor in aim to commence plant operation in the late 2010s.

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VALIDATION OF THERMAL HYDRAULIC COMPUTER CODES FOR ADVANCED LIGHT WATER REACTOR

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Abstract

The Czech Republic operates 4 WWER-440 units, two WWER-1000 units are being finalised (one of them is undergoing commissioning). Thermal-hydraulics Department of the Nuclear Research Institute Rež performs accident analyses for these plants using a number of computer codes. To model the primary and secondary circuits behaviour the system codes ATHLET, CATHARE, RELAP, TRAC are applied. Containment and pressure-suppressure system are modelled with RALOC and MELCOR codes, the reactor power calculations (point and space-neutron kinetics) are made with DYN3D, NESTLE and CDF codes (FLUENT, TRIO) are used for some specific problems. An integral part of the current Czech project „New Energy Sources“ is selection of a new nuclear source. Within this and the preceding projects financed by the Czech Ministry of Industry and Trade and the EU PHARE, the Department carries and has carried out the systematic validation of thermal-hydraulic and reactor physics computer codes applying data obtained on several experimental facilities as well as the real operational data. The paper provides a concise information on these activities of the NRI and its Thermal-hydraulics Department. A detailed example of the system code validation and the consequent utilisation of the results for a real NPP purposes is included.

1. INTRODUCTION

Until 1989, either home or modified Russian computer codes were used in the NRI for analysis of NPP safety,. In 1989, as first Western code, we obtained RELAP5 different Mods and in 1992, ATHLET and CATHARE, in 1995 — computer code TRAC.

Later on, containment codes MELCOR, DRASYS and RALOC, and advanced CFD computer codes (FLUENT, TRIO).

So, at the beginning, the fairly important part of our work consisted in learning how to use these codes and in modifying them for the hardware available at the NRI. The next stage consisted in the preparation of input data for the NPPs with WWER reactors and QA of input data decks.

Since all presented system codes have been developed and verified for western NPPs, their verification on Czech NPPs was an important stage. This verification was performed mainly on experimental facilities with parameters similar to those of WWER reactors, and on measurements performed during start-up of the NPPs. For this verification served, especially the ATHLET, CATHARE, RELAP, MELCOR and RALOC codes. We also participated in different computations of experiments organized under the auspices of OECD (ISP), EU (PHARE projects, 5^{th} framework projects) and national project (sponsored Ministry of Trade and Industry).

Our activity in the field of Advanced Reactor started by participation in the IAEA project Thermohydraulic Relationships for Advanced Water-Cooled Reactors. This activity, which was focused to determination of CHF, followed former projects taking place in the Czech Republic, both experimental and methodological.

Text cont. on page 137.

TABLE I. LIST OF COMPUTER CODES USED FOR WWER 1000 IN NRI

	CODE title	used by organisation country of origin developed for	type of code what is modelled	State of utilisation types of processes	state of code validation
1	RELAP5	NRI,USA,PWR	system code primary and secondary circuits	LOCA, LOCA secondary. primary-to-secondary, transients, ATWS SGTR	experimental facility measurements on NPP
2	CATHARE	NRI,France, PWR	system code primary and secondary circuits	Loss of Flow	experimental facility measurements on NPP
3	ATHLET 1.2 A		system code primary and secondary circuits	LBLOCA,SBLOCA, ATWS	experimental facility measurements on NPP
	ATHLET CD	NRI,Germany, PWR, WWER	System code, include core degradation model	LBLOCA	
4	DYNAMIKA-ÚJV	NRI,Russian, modified in Czech Republic WWER	primary and secondary circuits transients	Transients of WWER 440/213	measurement on NPP loss of flow
5	MELCOR	NRI,USA,PWR	primary and secondary circuits,	DBA, BDBA Containment, bubble condenser	against experimental test facility
6	DYN3D	NRI, Germany WWER	reactor, 3D neutron kinetics, simple models of coolant mixing	RIA	against experimental test facility
7	DRASYS	NRI, Germany PWR, WWER	containment, hermetic boxes	Containment, bubble condenser	against experimental facility
	RALOC FLUENT	NRI, US,	general fluid dynamics	Selected local problems	against experimental facility
	TRIO	NRI, France	general fluid dynamics	Starting implementation	

TABLE II. STATE OF ASSESSMENT OF COMPUTER CODES

CODE title	state of code verification for WWER in Czech Republic	type of experimental facility	process
RELAP 5	performed continuously	PMK- Hungary PACTEL-Finland NPP-Dukovany RVS-Řež Russien-NPP ISB-2	Small LOCA primary-secondary natural circulation loss of flow
CATHARE	performed continuously	ISP-38 ISP-42PANDA	Shut-down state
ATHLET 1.2A	performed continuously	NPP PMK- Hungary ISB	Loss of flow SBLOCA
DYN3D	performed continuously	LR0-NRI Řež	RIA, Turbine trip
DRASYS	performed continuously	PHARE 2.13, Bubble condenser qualification	LBLOCA
RALOC	performed continuously	ISP-42 PHARE 2.13, Bubble condenser qualification	LBLOCA
MELCOR	performed continuously	PHARE 2.13, Bubble condenser qualification	LBLOCA
FLUENT	performed continuously	ISP-43	Boron dilution
TRIO			

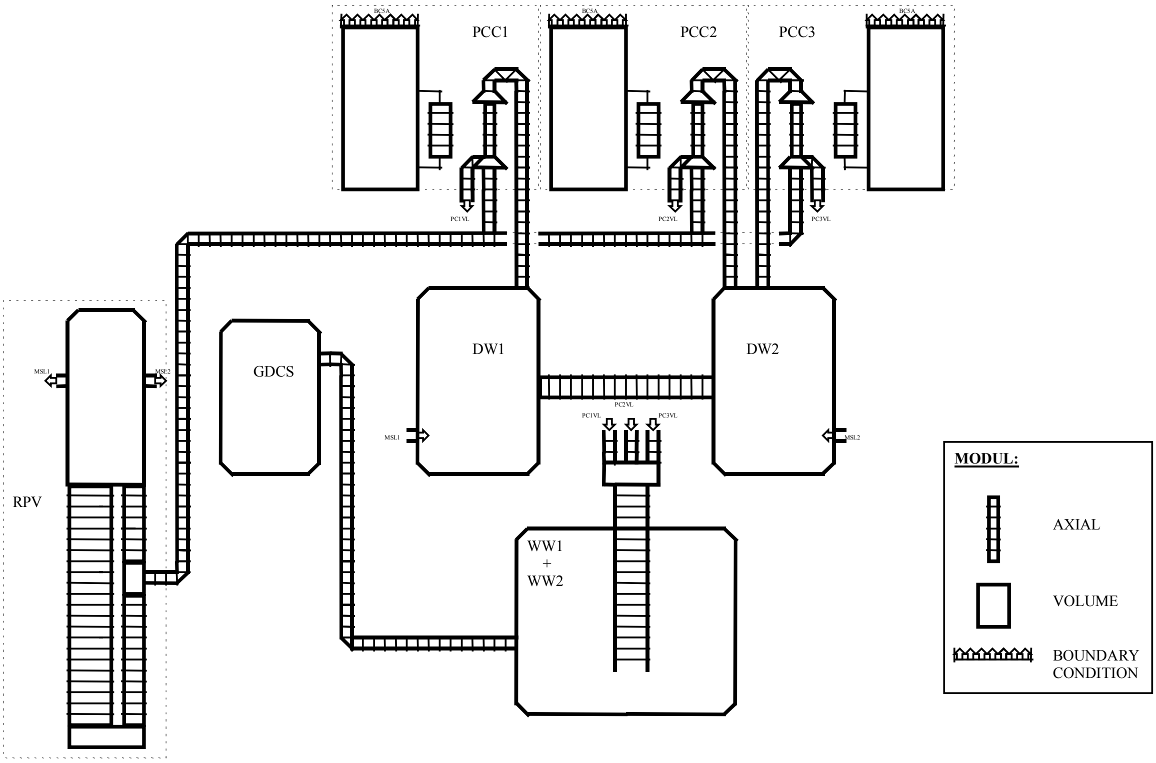

FIG.1. Nodalization of phase A (PANDA — CATHARE nodalization).

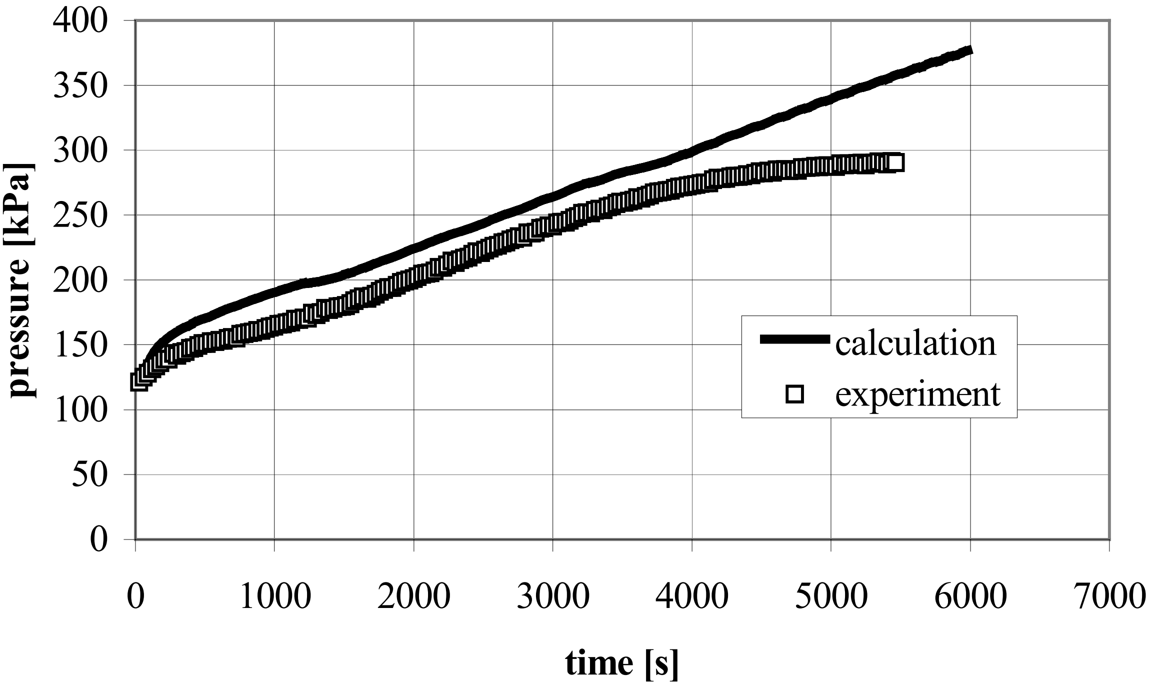

FIG. 2. PANDA — CATHARE reactor pressure, phase A.

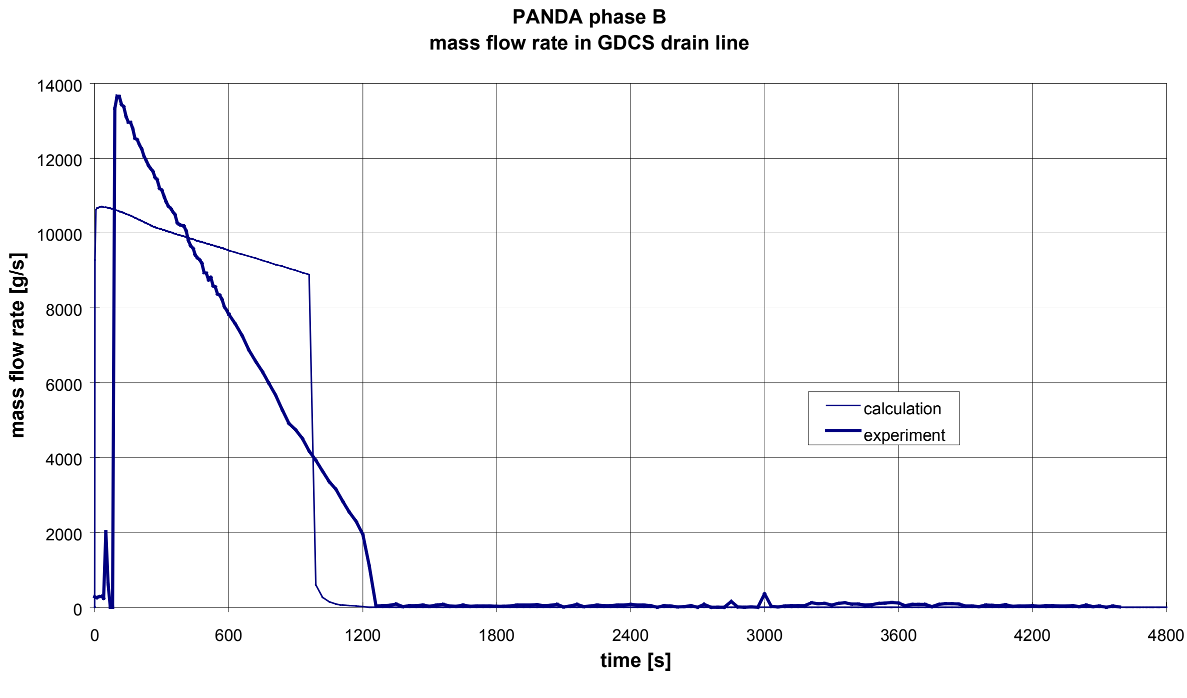

FIG. 3. PANDA-RALOC phase B, mass flow rate in GDCS drain line.

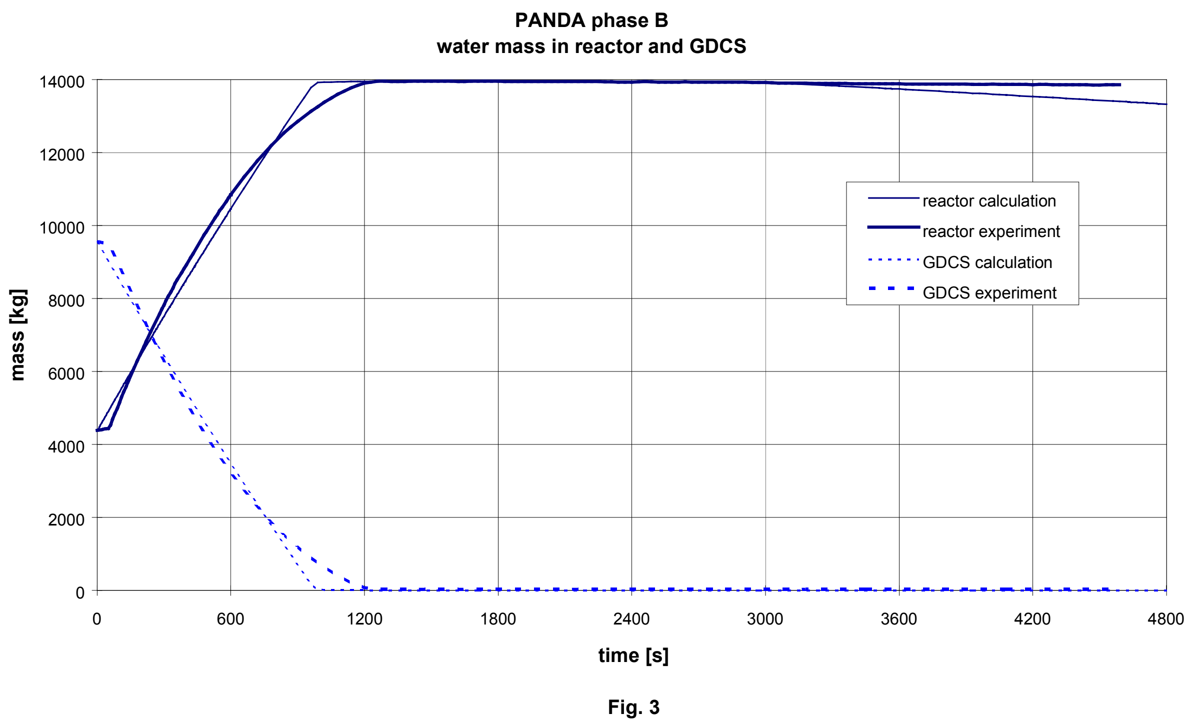

FIG. 4. PANDA-RALOC phase B, water mass in reactor and GDCS.

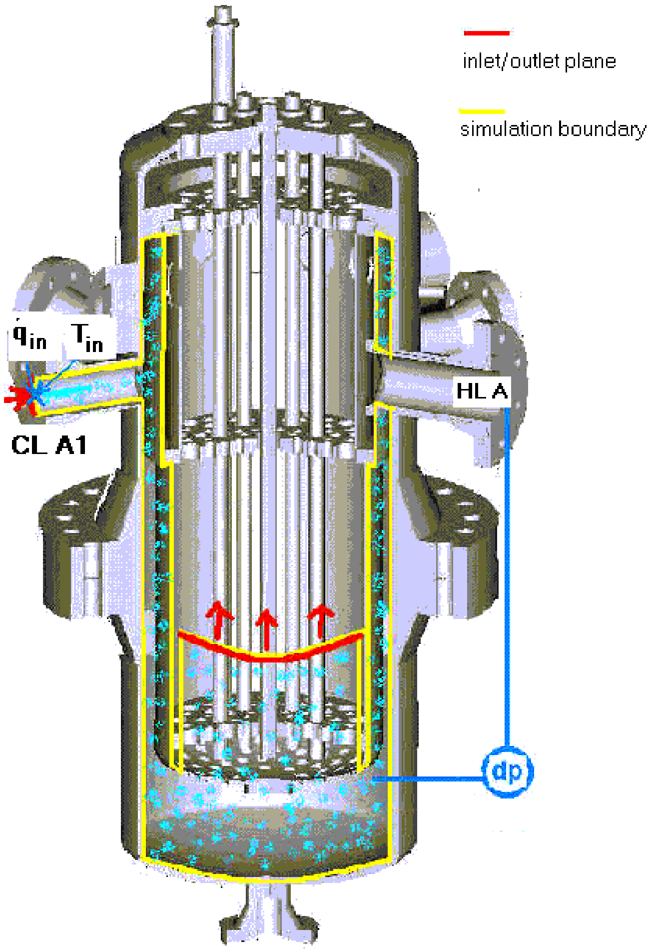

FIG. 5. ISP-43 solution region and boundary conditions.

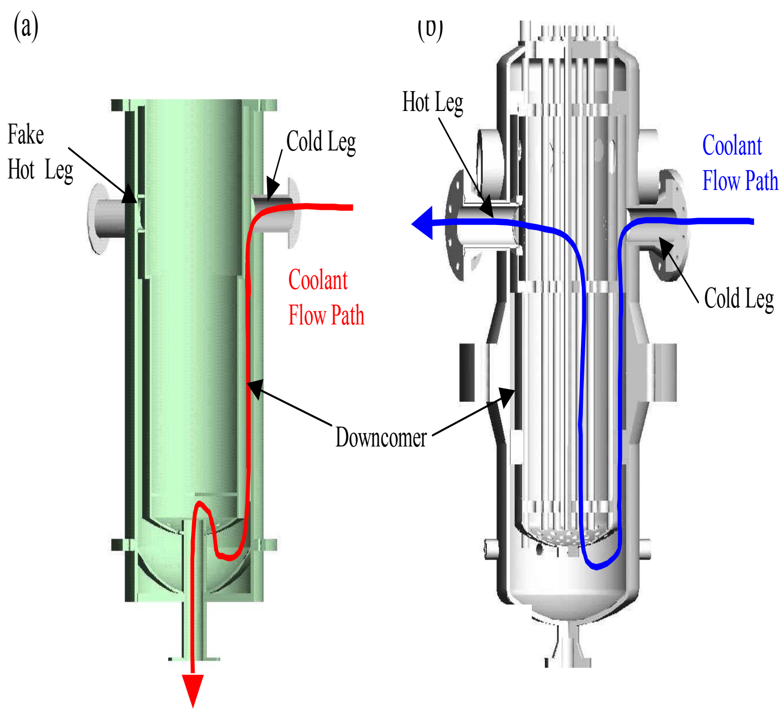

FIG. 6. ISP43- Cross-sectional view of the (a) visualization facility and (b) UM 2 × 4 loop facility vessels.

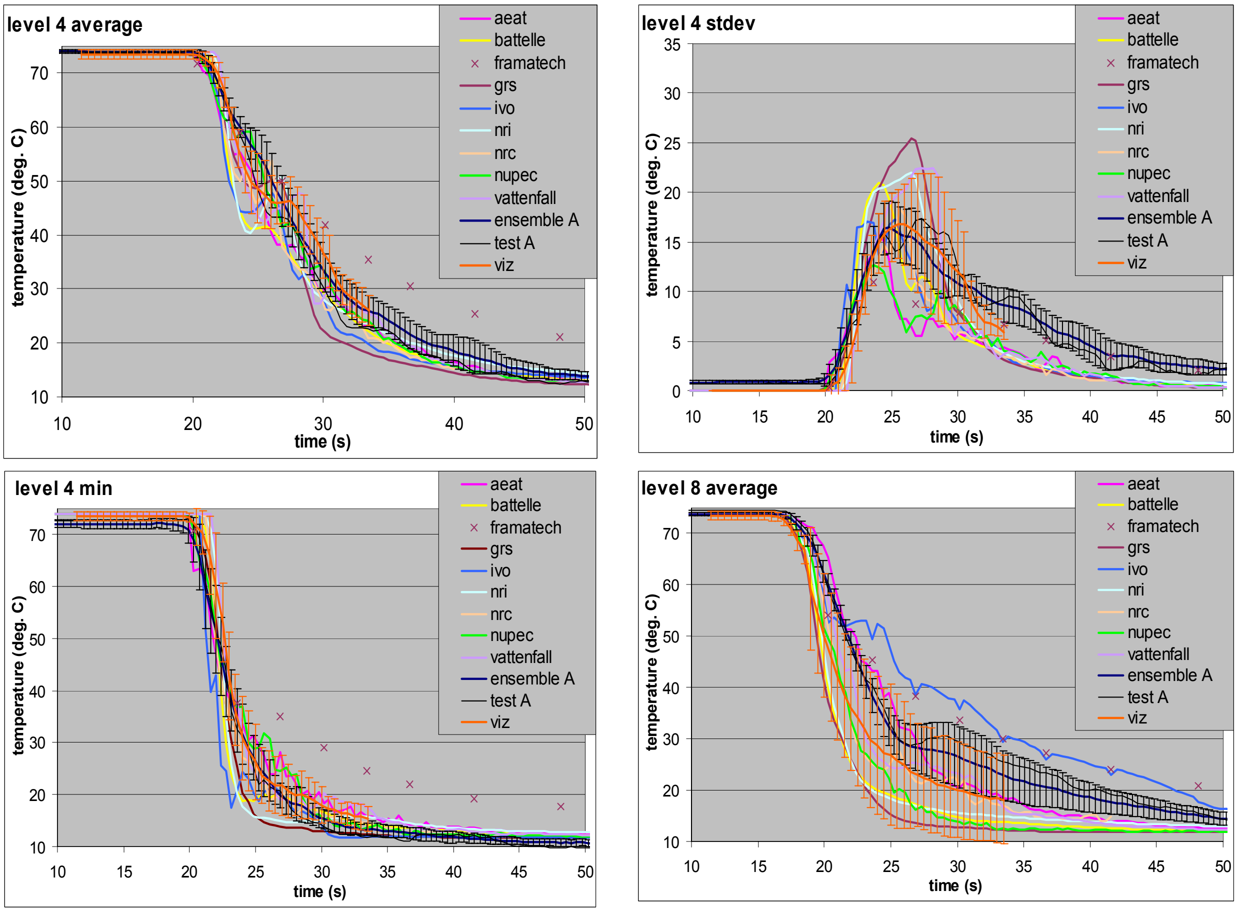

FIG. 7. ISP-43 Comparison of the results.

An experimental program was carried out in the Czech Republic, where Boiling crisis and Critical Heat Flux (CHF) were measured on the facilities that simulated the fuel assemblies of the former Soviet's Pressurised Water Reactors (PWR) WWER-440 and WWER-1000. The large part of experiments related to the CHF was performed at Skoda Plzeň Ltd, Nuclear Machinery Plant. The NRI started a complex of research activities in this field at the end of seventies.

The following topics have been developed systematically:

- Critical heat flux data bank for tubes, annuli and rod bundles
- Computational system based on the CHF data bank for testing critical heat flux correlations
- Subchannel analysis code for the calculation of coolant local thermo hydraulic conditions based on the rod bundles data obtained from the CHF data bank (subchannel CHF data bank)
- Computer code CALPER — a thermal hydraulic subchannel analysis code for the assessment of coolant local conditions in the fuel assemblies and in the core of PWR/WWER-type nuclear reactors
- System code calculations with CHF correlation for the safety analyses.

During the last years, a number of methodologies of improving NPP safety is under development within Czech Ministry of Industry and Trade, and computer codes are validated on experimental facilities. One of the important projects consists in searching new power sources and is also focused on the development of new methodologies including validation of computer codes.

2. BRIEF DESCRIPTION OF THE MAIN COMPUTER CODES

2.1. Computer code ATHLET

To perform analyses described above the advanced thermal hydraulic code ATHLET was applied, this code is used at the NRI since 1993. The code has been verified for the WWER accident analyses, in the first place by computations performed by the code-developer organization — GRS.

Significant advantage of the ATHLET code is the possibility to couple it with some other computer codes:

- 3D neutron kinetics codes (DYN 3D, BIPR 8)
- simple containment computer code CONDRU
- possibility of computer code for accurate calculation of reflood during LB LOCA

2.2. Computer code RELAP

This code was kindly transmitted to the Czech Republic by the US NRC. RELAP5 code is used for thermal hydraulic analyses of NPPs accidents, especially for loss of coolant accidents "small LOCA" and "medium LOCA", LB LOCA, primary-to-secondary leakages, and secondary side leakages. The RELAP5 (MOD2/RMA and later — MOD2.5 up to MOD3.2) is used at the NRI since 1989 for both analyses of NPP accidents and modeling of experimental facilities (pre and post-tests) which is very important for the code verification and generally for getting used to the code capability. Basic characteristics of the RELAP5 versions presented above are as follows:

- 1D model of two-phase flow including
- two mass conservation equations
- two energy conservation equations
- two momentum conservation equations
- equation for noncondensable gases
- boron transport
- point model of neutron kinetics
- hydrodynamics of the system, modeled using basic NAP components.

General features of the code and the way how the input data are introduced enables its application for various types of facilities.

In 1995 the model was modified to take into care of NPP Temelín analyses with new Westinghouse-supplied fuel, as well as for the verification of that of Zaporozhska NPP.

3. THE EXAMPLES OF COMPUTER CODES VALIDATION

3.1. The ISP-42 (PANDA)

PANDA is a large scale facility, which has been constructed at the Paul Scherrer Institute for the investigations of both overall dynamic response and the key phenomena of passive containment systems during long term heat removal phase for Advanced Light Water Reactors.

The whole ISP consists of six phases representing different scenarios:

Phase A:	Passive Containment Cooling System Start-Up
Phase B:	Gravity-driven Cooling System Discharge
Phase C:	Long-term Passive Decay Heat Removal
Phase D:	Overload at Pure-Steam Conditions
Phase E:	Release of Hidden Air
Phase F:	Release of Light Gas into the Reactor Pressure Vessel

3.1.1. PRE-TEST CALCULATIONS OF PHASES A,B,C,D WITH CATHARE CODE (Malacka, paper on PANDA Workshop OECD PSI JULY 2000)

Pre-test calculations of phases A,B,C,D were performed with code CATHARE 2 v1.4U_r1.51. Standard version of code was used without any modification of released version.

The nodalizations for phases B,C,D are similar. The standard modules of CATHARE (axial module, volume module, boundary condition module) were used. There are several common features in all nodalizations:

(1) The wet well volumes were joined into one volume.

(2) Instead of three PCC vent lines (or two in case of phase D) ends in the wet well the one common line is used ending in the wet well. For this, one auxiliary volume is used.

(3) In all axial and volume modules the walls are also modelled.

(4) Secondary sides of heat exchangers (PCC pools) are modelled as one volume module connected with one axial module.

(5) The standard CATHARE boundary condition BC5A module is used for open level at the PCC pools.

Problems, which were encoutered:

a) Initialisation

Problems with initialisation lead to a necessity to simplify input model. There were also problems to reach initial states of each transients. The auxiliary transients were used including sinks or injections of water, vapor and air. Those auxiliary transients could influence beginning of real calculated transients.

b) Heat exchange at passive containment cooler (PCC)

It was not possible to reach the right value of heat exchange at PCC. Mainly it is caused by used version of CATHARE code. The heat exchange in the calculations is very underestimated which influenced the courses of transients.

c) During calculations of phase C and D, when the water temperature at PCC pools reaches the saturated value, sudden unrealistic flow out of water from the pools follows. To prevent this event the pressure above the water level at pools was increased to the value of 130 kPa. The saturated temperature is then not reached, but there is also no evaporation of water from the pools which again influenced the results of pre-test calculations.

An example of calculation is on the figure, where there is a comparison of calculated and experimental reactor pressure during phase A. On this figure the problems, mentioned above, are illustrated. At the beginning of transient the calculated pressure is going above the experimental one due to auxiliary transient (see problem of initialisation). At the end of transient the experimental pressure decreases but calculated pressure goes up due to underestimation of heat exchange at PCC.

3.1.2. RALOC Model of ISP-42 in PANDA Facility (Simonkova, Paper on PANDA Workshop OECD July 2000 PSI)

The chosen phases A, B and F were calculated with code RALOC MOD 4.0 cycl AG. The basic objects simulating the individual parts of the PANDA facility represent **zones**, *juntions* and heat structures marked on diagram by different font style; structures are crosshatched. Each of the six vessels is simulated by one zone and the connection pipes between two drywells and two wetwells represents also one volume. Each of the three passive containment coolers is simulated by four volumes. The pipelines between vessels and coolers are modelled by various types of junctions. The action of valves is simulated by external control conditions. To all considered zones there are coupled heat structures representing the walls of vessels and modeling the heat transfer and conduction via walls. The heating power of reactor vessel and helium supply into reactor are modelled as a time dependent heat and He injection respectively.

During the pre-test calculations we met some problems which are listed as follows:

- The steam/air mixture from drywells entering in upper collector drum of PCC unit is not condensed in the tube bundle without definition of fluid parts with zero initial water mass in all PCC zones.
- Because it is not possible to connect the vent pipes to different zones (in case of wetwell 2) it was necessary to join together the units PCC2 and PCC3 into one unit PCC23.
- Some discrepancies were found in input values of cross section area of junctions without and with valves.

On the figures there is comparison of calculated and experimental results for phase B. There is compared the mass flow rate in GDCS drain line and its effect on the increase of the liquid level in reactor and the decrease of water mass in GDCS. In spite of different calculated and measured GDCS drain line mass flow the differencies between corresponding water masses in reactor and GDCS are very small.

3.2. Computer code FLUENT, ISP-43 (Muhlbauer: International Standard Problem ISP-43. Comparison of Pretest Calculations with Experimental Results. Report NRI 11 464, November 2000)

Increased power of present computers and progress in numerical methods and programming enables application of more sophisticated computer codes to some industrial problems. Before such aplication is made, the computer codes must be validated, especially when solving the problems of nuclear safety. Also the NRI therefore started validation and application of Computational Fluid Dynamics (CFD) codes to some selected problems encountered in NPP safety analyses. The commercial code FLUENT 5 was the first code undergoing such validation.

In the period of 1998-99, two sets of experiments focused on problems of rapid decrease of concentration of boric acid in reactor coolant at nuclear reactor core inlet were performed at the University of Maryland, US, under the auspices of OECD. The situation, when there is an inadvertent supply of boron-deficient water into the reactor vessel, could lead to a rapid (very probably local) increase of reactor core power in reactor, operated at nominal power, or to a start of fission reaction in shut-down reactor (secondary criticality). In the above mentioned experiments the transport of boron-deficient coolant through reactor downcomer and lower plenum was simulated by flow of cold water into a model of reactor vessel. These experiments were selected as the International Standard Problem ISP-43 and organisations, involved in thermal — hydraulic calculations of nuclear reactors, were invited to participate in their computer simulation. Altogether 10 groups took part in this problem with various CFD codes. The participants obtained only data on geometry of the experimental facility, and initial and boundary conditions.

A scheme of the experimental facilities is in Figs.. The University of Maryland (UM), College Park 2 × 4 Thermal-Hydraulic Loop and its plexiglass replica are scaled down models of the Three Mile Island Unit 2 Babcock&Wilcox pressurized water reactor. Cold water enters the vessel through the cold leg CL A1 and leaves it via hot leg HL A. Two situations have been tested: front mixing (infinite volume of cold water), test A, and more realistic slug mixing (finite volume of cold water), test B. Average temperature at the downcomer outlet (24 thermocouples at the centre of the downcomer gap) was selected as the primary figure of merit for comparisons.

The NRI group selected very simple input model since the ISP-43 represented our first larger application of the FLUENT 5 code and mainly of the GAMBIT pre-processor. After several attempts we decided not to model the flaps in the lower part of the downcomer, the perforated bottom of the core barrel, lower support plate, and the heater rods, and spent the capacity of the computer on the rest of the domain. Also the outlet plane was situated at the position of the support plate, quite near the downcomer outlet. Hexahedral control volumes were used throughout the domain with the exception of the region of cold leg nozzles, where unstructured tetrahedral mesh was generated.

In Figures main results of calculations of all participants of the ISP-43 together with the experimental data are presented for the test A. Despite the very simplified input model, our results are well within the range of results of the other participants, and follow some significant features of the experiment. In the post-test phase, reasons of discrepancies should be identified, and corresponding lessons learned.

3.3. Conclusions

Experience accumulated in the process of the advanced codes application for various simulated accidents and transients allows us to believe that these codes can be used, with a great degree of confidence, not only for the Safety Reports purposes, but also to support simulators which will be in operation at both Czech power plants and also for new type of reactors .

IMPROVEMENT OF OPERATIONAL PERFORMANCE AND INCREASE OF SAFETY OF WWER-1000/V-392

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Abstract

The national program of nuclear power development approved by the Russian Federation Government in 1998 considers the design of WWER-1000/V-392 power unit as a priority project of the new generation NPP with improved operational performances and increased safety. The pilot unit of this design (NVAES-2) is licensed for construction at the Novovoronezh NPP site. The NVAES-2 design is developed on the basis of standard power unit with reactor plant V-320. Twenty units of this type are in operation at the nuclear power plants in Russia, Ukraine and Bulgaria having totally about 270 reactor-years of operation. Two more V-320 units are being commissioned this year at Rostov NPP and Temelin NPP. So, the WWER-1000/V-392 design is as a whole an evolutionary development of the operating standard unit WWER-1000/V-320. Many technical solutions aimed at increase of safety and improvement of operational performance of the plant are implemented in the NVAES-2 design, such as advanced reactor WWER-1000, passive system of residual power removal, passive system of the core flooding under loss-of-coolant accidents, and others. NVAES-2 design refers to a class of advanced light water reactors and corresponds to the international requirements imposed to the nuclear power plants to be put into operation after the year 2000. New V-392 power unit has a good perspective from the view point of extensive implementation in the framework of the nuclear electricity production in Russia. Design decisions on NVAES-2 power unit with WWER-1000/V-392 reactor plant which assure significantly higher safety level and improve economical performance as compared to the operating WWER-1000 units are briefly considered in the present paper.

1. INTRODUCTION

Nowadays in eight countries 49 commercial reactors of WWER-type are in operation with total power exceeding 32 GW, including 20 reactors WWER-1000 and 29 reactors WWER-440. More than 800 reactor-years of operation without serious incidents with radioactivity release outside NPP site have demonstrated high level of WWER type reactors safety. Economic operational indices of these reactors point out the competitiveness of WWER reactors as electric power producers.

Nevertheless, operation experience of WWER nuclear power plants, new national and international safety standards and changes in economy of Russia govern the necessity of development of advanced nuclear power plants. New reactors of WWER type shall possess the increased characteristics of safety and economical efficiency, be more convenient for operation and maintenance.

The national program of nuclear power development for the period until 2010 was approved by the Russian Government in July 1998. The program is aimed at development of nuclear power with NPP siting in those regions, for which appropriate permission for construction of nuclear power plants is received. In particular, development of perspective NPP designs on the basis of the advanced safety technologies and construction of advanced WWER units of increased safety are being planned. The program is also intended to expand nuclear technologies export including WWER nuclear power plant construction abroad (in particular, in Iran, China and India in accordance with the existing inter-governmental agreements and contracts for construction of NPP with the advanced WWER-1000 reactors).

Within the context of these tasks, Russian design and scientific organisations are performing a complex of R&D work on development of power units of new generation with advanced reactor plants of 1000 MW(e) power which meet new requirements for safety and economical efficiency of electric power production. Design decisions on Novovoronezh NPP-2 with WWER-1000/V-392 reactor plant which assure significantly higher safety level and improve economical performance are briefly considered in the present paper.

2. NVAES-2 POWER UNIT

Power unit 1000 MW(e) design with reactor plant WWER-1000/V-392 [1] is developed for construction in Russia and elsewhere; the power unit forerunner is planned for commissioning at the Novovoronezh NPP site (NVAES-2). The design is based on the application of equipment and processes that have proved their operability at existing nuclear power plants. At the same time, this design is aimed at achieving higher safety level and essentially better economical indices as compared to operating WWER-1000/V-320 units.

Power unit V-392 presents itself a mono-unit with four-loop reactor plant, working pressure in the primary circuit 15,7MPa, reactor outlet coolant temperature 593K, steam pressure 6,3MPa, steam capacity about 6000 t/h.. Layout of the reactor building, turbine hall, safety and auxiliary system buildings ensure minimum routing of communications and high reliability of normal operation and safety functions. Reactor plant is placed in the double containment which prevent from radioactivity release to the environment and protect reactor plant against of external impacts. The design is intended to take up the external impacts from natural and man-induced phenomena such as earthquake of magnitude of 8 as per MSK-64 scale, tornado, hurricane, shock wave with front pressure to 30kPa, etc.

The advanced design of NVAES-2 has been developed on the basis of standard power unit with reactor plant V-320. These units are in operation for a long time at nuclear power stations in Russia, Ukraine and Bulgaria. Today 20 power units with WWER-1000 reactor operate, and total time of their work is 270 reactor-years. Power units of this type are also being constructed or planned in Czechia, Russia, Ukraine. In particular, unit 1 of Rostov NPP in Russia and unit 1 of Temelin NPP in Chechia are being commissioned this year.

Many technical solutions aimed at increase of safety and improvement of operational performance of the plant are implemented in the NVAES-2 design. Some examples of such advancements are as follows:

- advanced reactor WWER-1000;
- passive system of residual power removal (under certain conditions, this system can also facilitate to the emergency containment cooling);
- passive system of the core flooding under loss-of-coolant accidents;
- passive system of rapid boron injection for the reactor shutdown;
- primary coolant pump preventing coolant leak under long-term station blackout;
- passive system to create the rarefied atmosphere in the inter-containment gap and to clean the emergency leaks under accident conditions including severe ones.

So, the WWER-1000/V-392 design is as a whole an evolutionary development of the operating standard unit WWER-1000/V-320. The design refers to a class of advanced light water reactors and corresponds to the international requirements imposed to the nuclear power plants to be put into operation after the year 2000. This conclusion was confirmed by a few peer reviews of NVAES-2 design including the reviews performed by Western experts.

3. IMPROVEMENT OF OPERATIONAL PERFORMANCE

High operational availability of the power unit with WWER-1000/V-392 reactor plant is assured owing to:

- application of the equipment and technical solutions proven by operation;
- using of proven materials and technologies for the equipment manufacturing;
- optimal water chemistry of primary and secondary circuits;
- reducing the number of process systems and simplification of their schemes.

The design provides for on-line monitoring the state of equipment and components with the help of special diagnostic systems (noise diagnostics, monitoring the equipment vibration, detection of loose and poor fixed objects, primary-to-secondary leak monitoring). Special systems are provided for information support of the operation (safety parameters display system, equipment residual life assessment system, operator's support system).

To improve the conditions of maintenance of the reactor plant systems and equipment the appropriate experience is taken into account as well as the practices applied by Western specialists, including the results of international assessment of technical solutions accepted in the V-392 design.

Power unit NVAES-2 designing was carried out with regard for the requirements related to inspection, maintenance and repair of equipment. In particular, good access is provided to the equipment requiring periodical examination and repair, the non-destructive inspection systems included. For the primary equipment the materials are used mainly with low content of cobalt and other elements with long half-life period. Together with designing of systems and equipment the process specifications for their maintenance and repair were developed. During the design process of WWER-1000/V-392 power plant, all the comments of Russian Gosatomnadzor related to correspondence of the previous V-320 unit to national safety norms have been solved. Also, the relevant recommendations of the IAEA missions to operating plants have been taken into account.

Design of the WWER-1000/V-392 power unit makes maximum use of technical solutions proven by operation experience of existing WWER-1000 power units. Such consistency improves technical characteristics of the reactor plant including also operational availability and maintenance. At the same time, a number of new technical decisions on the unit systems and equipment are applied in V-392 design to take into account operating experience of WWER-1000 units and modern requirements to nuclear power plant competitiveness. Some examples of advancements aimed to improve the operational performance and plant economical efficiency, to decrease the costs of construction, repair and maintenance of the systems and equipment are given below.

3.1. Reactor plant

The length of reactor pressure vessel V-392 is increased in comparison with reactor V-320 at the expense of larger length of the supporting shell, keeping the possibility to transport the vessel by railway. With this, the core top elevation is decreased in relation to the elevation of the reactor supporting structure that allows to reduce considerably the personnel dose commitment in maintenance of reactor, steam generator and electric drive of the reactor coolant pump. Thus, neutron flux in the region of support is reduced almost two times in direct passing from the core through the vessel, and from the streaming out of the gap between the reactor vessel and «dry» shielding - more than 10 times.

The vessel extension allows also to reduce the neutron irradiation intensity of critical weld between the supporting shell and the shell of nozzles zone. Owing to this, the margin is increased for the vessel integrity under pressurised thermal shock. The vessel extension allowed to increase the coolant inventory between the core top and the lower generant of the inlet nozzle, that is, to improve the core cooling conditions under loss-of-coolant accidents.

Containers with surveillance specimens in reactor V-392 are placed on the vessel inner wall, whereas in reactor V-320 the surveillance specimens are placed on the upper end of the core baffle. Such upgrading brings together the conditions of neutron flux effect on the surveillance specimens and on the vessel metal allowing to predict more exactly the variation of the vessel mechanical properties in the course of operation.

In reactor V-392 the stops are installed on the core barrel bottom, and arms are placed on the upper end of the core baffle (these components are not provided in the operating reactors V-320). The stops are installed with a small clearance in relation to the core barrel bottom. Therefore in case of hypothetical guillotine break of the core barrel a part, broken away, will move only slightly and the arms will keep the core baffle in engagement with the protective tube unit lower plate under such downward motion of the core barrel.

On the cylindrical part of the core barrel in the zone of flow separator the compensating plates are placed with the help of which the design value of mounting clearance is achieved. In heating-up the reactor vessel and internals this clearance is decreased and the core barrel is clamped to the flow separator over the whole perimeter that reduces the vibration loads on the core barrel. Design of the core baffle channels is changed to smooth temperature fields in the baffle and to decrease the resulted deformations of the baffle.

Distance between the middle and upper plates of the protective tube unit is increased. This allows to increase the bending radius of the guiding channels where the in-core instrumentation elements are arranged. Owing to this upgrading all channels are brought into periphery nozzles of the upper unit that improves the reliability of the in-core measurement system and simplifies its maintenance.

On the upper unit of V-392 reactor 121 nozzles are provided for the members of reactivity control system and reactor emergency protection (CPS) in comparison with 61 nozzles in V-320 reactor. This gives a possibility to vary the number and arrangement of CPS members and to optimise each fuel cycle for reaching the best characteristics of the core safety and efficiency.

In V-392 reactor the measurements of coolant temperature and core power are combined and brought through the common nozzles of in-core instrumentation (ICI), while in V-320 reactor there are separate nozzles for temperature monitoring and nozzles for core power monitoring. With this, all ICI nozzles are arranged on the periphery of the reactor upper head that facilitates the access to them when reactor assembling or removing the upper head unit, and reduces the repair personnel dose commitment.

In V-392 reactor the upgraded control rod drive is used with the improved maintainability and more simple procedure of the drive mounting-dismounting. The drive service life (including electrical part) is 30 years with the outlook of its further extension to the reactor service life. Position indicator, used in the drive, provides for monitoring the position of control rod in the core in each 20 mm (instead of 350 mm in the existing analogues). Monitoring of drop time and position of control rods in the core is also provided under the reactor scram, that is, the functions of diagnostics are also fulfilled.

Design of reactor plant V-392 is developed with application of «leak-before-break» concept that allows to give up the massive supports-restraints on the main coolant pipelines. Owing to this, all sections of pipelines become accessible for in-service inspection that improves their reliability. The personnel dose commitment during the inspection of pipelines bending is also reduced because the labour consuming procedures on removal of the upper parts of the emergency supports are excluded.

Many other components of reactor plant V-320 are upgraded in V-392 design with the aim to enhance the station safety and efficiency and improve the conditions for operation and maintenance. In particular, for reactor plant V-392 the reactor coolant pump MCP-1391 is used. This pump is the upgraded MCP-195M, which is used for the operating reactor plants V-320. In MCP-1391 pump water is applied as the lubricant and cooler of the main bearing; in combination with the improvements in the system of motor lubrication this allows to give up the outside oil system and exclude possible fire.

3.2. Core and fuel handling

The core of V-392 reactor uses practically all technical solutions on the advanced core of operating WWER-1000/V-320. The prototype of the advanced fuel is standard fuel assemblies (FA) with stainless spacing grids and guiding channels which have been in operation at WWER-1000 since 1982. Originally the standard fuel was operated in the two-year fuel cycle, then the transition was done to three-year fuel cycle with the corresponding increase of average burn-up.

Operation experience of standard fuel revealed certain drawbacks both concerning efficiency of fuel utilisation, and design of fuel assembly (highly absorbing material within the active part; boron-based burnable absorber; low design service life; one-piece structure). Therefore designers and manufacturers of Russian fuel for WWER-1000 have developed the advanced FA with zirconium structural materials and this FA is being implemented at present.

Advanced fuel assembly (AFA) has been developed both for replacement of standard fuel at the operating reactors, and for new nuclear power plants with advanced WWER. The main difference of AFA, being the most effective as to economy, from standard fuel is application of only zirconium structural materials in the assembly active part. This allowed (in combination with specially developed refuelling patterns) to reduce the specific consumption of uranium approximately by 13%. Application of gadolinium burnable absorber instead of boron absorber allows to reduce this index by approximately 5% more. Application of AFA allows also to reduce enrichment of makeup fuel. Using of uranium-gadolinium fuel allows to reduce neutron fluence to the reactor vessel, to improve flexibility of fuel cycle, to exclude expenses for operation and storage of burnable absorber rods.

Guiding channels (GC) for absorbing elements are optimised by outer diameter and wall thickness. The aim was to improve the conditions of insertion of absorbing elements under the mode of free drop, to keep sufficient DNBR in the surrounding fuel rods and to provide for the required strength of GC, as the load-carrying component. Tests are being performed of GC of zirconium alloy Э635 of decreased radiation creep that could be used at higher burn-up.

Difference in linear expansions coefficients of guiding channels of the assembly and reactor core barrel is compensated by increase of the working stroke of spring block of FA cap. For connection/disconnection of AFA stainless cap and GC a simple device is used not requiring replaceable fasteners or complicated fixtures with power nut drivers. As the inspection stand is available the procedure on dismounting or mounting the AFA cap takes a minimum time.

Absorbing elements are upgraded as well. The combined absorber is used in them comprising boron carbide and dysprosium titanate. This allowed for two times increase of absorbing element service life. For the absorbing element cladding the new alloy is applied with improved mechanical properties and radiation strength. This allows to decrease the cladding thickness and improve the efficiency of absorbing rod.

So, the advanced fuel provides for improvement of safety and economic efficiency of reactor plants of new generation. Nowadays all the mentioned improvements of the core are being checked and implemented at the operating NPP with WWER-1000, therefore the experience obtained is a reference one for the advanced reactor plant V-392.

In design of fuel handling system for the advanced reactor plant V-392 some changes are introduced into the process and structure of fuel handling equipment. These changes provide for improvement of safety, of maintenance conditions and simplification of fuel handling procedures. For example, in V-392 design a nuclear accident in case of drop of transport packing set is prevented by installation of shock absorbers at the places of packages lifting to the height exceeding the design one for these packages.

All the fuel handling equipment of V-392 reactor plant (in-plant transport packing set for fresh and spent fuel, leak-tight bottles, bottles of defective assembly detection system) have the cells for fuel assemblies made of hexahedral tubes. This measure provides for improvement of nuclear safety under accident situations and also prevents mechanical damage of fuel assembly during its installation and withdrawal from the fuel handling equipment.

3.3. Safety systems

Technical solutions on the safety systems of NVAES-2 design are aimed, in the first turn, at significant increase of the safety level of new unit as compared to the operating WWER-1000/V-320. Safety system structure and configuration have been also optimised from the viewpoint of economical characteristics (number of equipment, repair and maintenance costs, dose loads to personnel, etc). Task to enhance the safety level as compared to V-320 power unit has predetermined the implementation of the design solutions dealing with protection against common-cause and dependent failures in the safety systems including:

- application of functional and structural diversity in the systems that perform each
- critical safety function;
- protection of the safety-related systems and equipment against internal and external
- impacts;
- protection against erroneous operator’s actions.

Analysis of technological possibilities of implementation of the above issues has shown that the simplest and the most economical way is equipping of the modular safety system with active and passive principles of operation that duplicates each other as regards the critical safety functions. Functional and structural diversity in the safety systems permits to provide in-depth defence against the common cause failures and operator errors. The fulfilment of a number of safety functions is based on this principle in NVAES-2 design, for example:

- reactor shutdown and maintain the sub-criticality;
- decay heat removal;
- maintain the reactor coolant inventory;
- maintain the continuous rarefaction in inter-containment gap.

Application of the above principles and solutions in practice have come across with a problem of uncertainty with respect to achievable efficiency; the latter is connected with necessity of quantification of the safety level. To evaluate and assess the achievable safety level, one has to perform the full-scale PSA in addition to deterministic principles. Common realisation of the deterministic and probabilistic analyses during the NVAES-2 design process has permitted to obtain optimal solution with respect to balance of the active and passive trains of the modular safety systems. As a result of this optimisation, total core melt frequency for NVAES-2 is about three orders of magnitude less than for unit 1 of Balakovo NPP with V-320 reactor plant.

In WWER-1000/V-392 design it is supposed to use some passive systems intended for fulfilment of the main safety functions (reactor shutdown, decay heat removal, core cooling). These systems, in the first turn, are intended to improve considerably the plant safety. Alongside with this, passive systems, as a rule, are simpler in operation and maintenance, and therefore improve also the plant economical characteristics.

From the reliability and economical efficiency viewpoint, so-called “principle of adjustment” of safety and normal operation functions has a great importance, and in the NVAES-2 design a number of safety systems are based on this principle. Following to this principle, a safety system performs the designated normal operation function, and at the accident signal appearance the system begins to fulfil the required safety function. At this, no (or minimum number of) changes in the status of the system’s elements (valves, pumps, etc) are required. Such solution permits, as compared to the traditional on-duty safety system, to increase in 5-6 times the reliability of the safety function fulfilment due to small sensitivity to latent failures. From the economic viewpoint, the quantity of devices, valves, pumps, cables, I&C and automation is decreased essentially. For example, the NVAES-2 active system for emergency heat removal via primary circuit has 4 pumps, whereas the same functions with the traditional approach would require 12 pumps.

3.4. Instrumentation and control system

The existing nuclear power plants were constructed by designs of 60-70-ties with the use of automatics, equipment, cables and actuating mechanisms manufactured mainly at the enterprises of the Soviet Union at that time. Monitoring and control system (MCS) of Russian plants may be conventionally divided into three generations.

The first generation includes MCS of the reactors commissioned before 1975. Specific feature of these systems is wide application of remote control from the operator’s panels, remote control of process parameters and relatively simple automatic devices (process protections, automatic control, interlocks, signalling). MCS of the second generation is characterised by wide application of measuring and control instruments with the unified electric signal, logic control devices, aggregated monitoring systems for the plant process systems. In the systems of the second generation the links between the control devices for the reactor, turbine and other system were considerably extended, the scope of monitoring and automation of processes was increased. In the devices of automatic control and protection of the reactor the elements of microelectronic technique are applied. The specific feature of MCS of the third generation is wide application of micro-processing and computing technique for control of processes. For representation of information to operator both the mimic panels, and alphanumeric and graphic displays are applied.

For the advanced V-392 design, new systems of monitoring and control are developed. They apply widely the micro-processing technique for implementation of all MCS functions including safety functions. The requirements of new regulatory documents are taken into account as well as recommendations of international standards, up-to-date principles of system construction, such as:

- high automation level of processes;
- regard for operation experience of the existing systems and the latest achievements in the world practice in creation of control rooms;
- developed information support to operator, high functional reliability and self-diagnostics of hardware;
- redundancy, independence, diversity, resistance to common cause failures;
- assurance of serviceability under internal and external impacts including accident
- conditions;
- reducing the maintenance work scope and the number of personnel engaged.

Development of hardware for new MCS is performed according to complex programme of Minatom of Russia prescribing the development of software-hardware for MCS engineering for the nuclear power plants being reconstructed, under construction and under design, including advanced power unit WWER-1000/V-392.

4. SAFETY INCREASE

The NVAES-2 design is developed on the modern level of the national and international requirements which envisage:

- the probability of limiting release and serious core damage at beyond-design accidents less than 10^{-7} and 10^{-5} per reactor-year, respectively;
- reduction of urgent evacuation area to 300-500 meters and emergency planning area to protect the population in case of beyond-design accidents to 700-3000 meters.

This power unit is designed so, that its safety level exceeds the above requirements, and radiation impact of NVAES-2 on the population and environment is essentially below the allowed limits established by up-to-date safety standards.

Significant safety increase in V-392 design as compared to the operating V-320 units is achieved due to extensive application of passive safety means, using natural physical processes, along with the traditional active systems. The IAEA Conference on “The Safety of

Nuclear Power: Strategies for the Future” [2] has noted that the use of passive safety features is a desirable method of achieving simplification and increasing the reliability of the performance of essential safety functions, and should be used wherever appropriate. However, the application of passive means is connected with some problems, which have to be solved by each plant designer. The passive systems have their own advantages and drawbacks in comparison with the active systems both in the area of plant safety and economics. Therefore a reasonable balance of active systems and new passive means is adopted in V-392 design to improve safety and public acceptability of nuclear energy.

One important problem related to the implementation of the passive means is that, in the most cases, sufficient operating experience of the passive systems/components under real plant conditions does not exist. Besides, the existing computer codes for transient and accident analysis are not sufficiently validated for the conditions and phenomena which are relevant to the passive system functioning. Therefore, the extensive experimental investigations and tests have been already performed and are being planned to substantiate the design of the safety features proposed for the NVAES-2 power unit. On this basis, a number of relatively innovative passive safety means are implemented in V-392 design to ensure or to back up the fundamental safety functions: reactivity control, fuel cooling and confinement of radioactivity.

4.1. Reactivity control

Traditional gravity-driven control rods are the main system to ensure reactor scram both in currently operating and new WWERs. For existing pressurized water reactors, this system is not sufficient to bring the reactor to a cold shutdown state; therefore the control rod system of existing WWERs is supported by pumped emergency supply of the borated water to the primary circuit. New WWER designs V-407 and V-392 have an increased number of gravity-driven scram rods to maintain shutdown margin even in the absence of boron supply during the reactor cooling down.

Although very good reliability records exist for scram excitation, some failures of the gravity-driven control rod insertion have been recognized. The failures occurred for the different reasons; in particular, the cases of insertion speed reduction and incomplete insertion due to fuel assembly deformation have been reported during last ten years (see for example [3]). Besides, some failure modes may be considered which could prevent all the control rods to insert, and it was the basis for designers to analyze Anticipated Transient Without Scram events.

Keeping that in mind, for WWER-1000/V-392 a special quick boron supply system has been designed as a diverse system to the gravity-driven scram system. A concentrated boron solution tank is connected to the suction and discharge pipes of each main coolant pump. The valves in the connecting pipes will automatically open if there is a demand for reactor trip but the reactor power after some time is higher than its value after scram should be. The concentrated boron solution is supplied to the reactor due to pressure difference between discharge and suction of the main coolant pump (pump head). Inventory and concentration of the boron solution is selected to ensure compliance with safety criteria in the design events accompanied by control rod system failure to trip the reactor. The operability of the quick boron supply system has been confirmed by extensive experimental investigation using a scaled model.

4.2. Fuel cooling

The safety function «fuel cooling during transients and accidents» is ensured by provision of sufficient coolant inventory, by coolant injection, sufficient heat transfer, by circulation of the coolant, and by provision of an ultimate heat sink. Depending on the type of transient/accident, a subset of these function or all of them may be required. Various passive systems and components are proposed in V-392 design to fulfill these functions.

The passive residual heat removal system (PHRS) is included in the V-392 design to remove heat from the reactor plant. The design basis of this system is that in case of station blackout, including loss of emergency power supply, the removal of residual heat should be provided without damage of the fuel and of the reactor coolant system boundary for a long time period. The PHRS consist of four independent trains; each of them is connected to the respective loop of the reactor plant via the secondary side of the steam generator. Each train has pipes for steam and condensate, valves and modular air-cooled heat exchanger installed outside of the containment. The steam that is generated in the steam generator due to the heat released in the core condenses in the air-cooled heat exchanger, and condensate is returned back to the steam generator. The motion of the cooling media (steam, condensate and air) takes place in natural circulation.

The passive system for reactor flooding during LOCA in V-392 design comprises two groups of hydro-accumulators. First group (so called first stage accumulators) consists of four traditional ECCS accumulators being used at operating WWER-1000 reactors; these accumulators are pressurized by nitrogen to 6MPa and connected in pairs to the upper and lower plenums through special nozzles in the reactor pressure vessel. Second stage accumulators are 8 tanks connected to the reactor coolant system through the check valves and special spring-type valves. These valves are kept closed by the primary pressure; when the primary pressure drops below 1,5MPa, the spring open the valve. Such a connection configuration and valve design ensures continuity of hydrostatic head irrespective of the primary pressure change during an accident. Installation of hydraulic profiling of the outlet route ensures a step-wise limitation of the water flow rate from the tank when the water level in the tank is decreasing. The water inventory in the second stage accumulators (about 1000t) ensures the core cooling for 24 hours during a LOCA even if all active ECCS mechanisms are inoperable. Joint operation of the second stage accumulators and SPOT gives a possibility to increase this period by natural way and/or by a simple means of the accident management.

4.3. Confinement of radioactivity

This safety function is ensured by protecting and maintaining the integrity of the potential radioactivity release barriers (fuel, reactor system boundary and containment). These barriers are passive components as themselves; in addition, several passive means are proposed in V-392 design for the protection of these barriers (some of them are reflected above). The V-392 design implies substantial improvement of the containment protection against different loads related to design basis and severe accidents, and various passive systems are important part of this protection.

To limit considerably the release of fission products beyond the containment, a permanent under-pressure is maintained in the inter-containment gap of the V-392 design. This safety function, one of the most important, is fulfilled by two systems: (1) an exhaust ventilation system equipped with a filtering plant with suction from the inter-containment gap and outlet into the stack; (2) a passive system of suction from the inter-containment gap. The first system is intended to control removal of steam-gas mixture from the inter-containment gap under design basis accidents with loss of external power. The system is capable to remove and clean the leaks from the inner containment 1.5% of containment volume per 24 hours. The second system consists of lines connecting the inter-containment gap with the PHRS exhaust ducts, which are always in the hot state. This solution enables permanent removal and purification of inner containment leaks irrespective of the electricity supply and operator actions. According to estimations, the under-pressure is maintained at any point of the inter-containment gap with inner containment leaks up to 1.5% of containment volume per day (the design basis for the containment is 0.3%). The technical solution described above in combination with the systems for the containment pressure decrease (spray system and passive heat removal system) allows to give up the filtered venting system designed for NVAES-2 in spite of this system follows the current requirements that filtered venting should not increase the risk of loosing the containment function and filtered venting is not required in the short term of a core melt accident.

Special systems and components are implemented in NVAES-2 design to prevent hydrogen burning or explosion. The hydrogen suppression system comprises passive catalytic hydrogen igniters based on an efficient high porosity cellular material. Each of 50 elements of this system is capable to oxidize about 30 grams of hydrogen per hour at its volumetric concentration 4%. This system prevents the explosive concentration of hydrogen even if 100% of the core Zr will be oxidized during an accident.

5. CONCLUSIONS

In accordance with the national program of nuclear power development, a complex of R&D work is being carried out by the Russian design and scientific organisations on development of advanced power units of new generation with reactor plant WWER-1000/V-392. These units meet up-to-date national and international requirements for safety and economic efficiency of electric power production. Permission of the Gosatomnadzor RF has been issued for the construction of the forerunner unit of this design at Novovoronezh NPP site.

In the advanced WWER-1000/V-392 power unit design many new technical solutions are applied with a view to improve safety, to optimise economical indices and to minimise the expenses on maintenance of the station equipment and systems.

The power demand prognosis [4] shows that even based on a conservative scenario, power use in Russia will increase by about 30% by the year 2010, and by the year 2030 — in 2 times as compared 1995 level. The tendency during the last few years in Russia is to double-triple lead the rate of increase in fossil fuel cost compared to the rate of increase in nuclear fuel cost. So, new WWER-1000/V-392 power unit has a good perspective from the view point of extensive implementation in the framework of the electricity production in Russia.

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DESIGN FEATURES IN KOREAN NEXT GENERATION REACTOR FOCUSED ON PERFORMANCE AND ECONOMIC VIABILITY

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Abstract

As of the end of Dec. 1999, Korea's total nuclear power capacity reached 13,716 MWe with 16 units in operation and 4 units under construction. In addition, as part of the national long-term R&D program launched in 1992, the Korean Next Generation Reactor (KNGR) is being developed to meet the electricity demands in the years to come which is expected to be safer and more economically competitive than any other conventional electric power sources in Korea. The KNGR project had successfully completed its second phase and is now on the third phase. In Phase III of the KNGR design development project, KNGR aims at reinforcing the economic competitiveness while maintaining safety goals. To achieve these objectives, the design options studied and the design requirements set up in the first phase and pursued while the second phase are being reviewed. This paper summarizes such efforts for design improvement in terms of performance and economic viability along with the status of nuclear power generation in Korea, focusing on KNGR in current.

1. STATUS OF NUCLEAR POWER GENERATION IN KOREA

The first nuclear power to Korea was introduced in early 1970s with the construction of Kori unit 1, a 600MWe Westinghouse type PWR. Two additional 600MWe plants were built, one by AECL of Canada, and the other by Westinghouse. These three units are called first generation NPPs in Korea, and were constructed through turn-key contracts. Afterward, six additional PWRs of 950MWe, Kori 3&4, Yonggwang 1&2 and Ulchin 1&2, were put on the grid during relatively short period (1978-1988), the construction of six plants in 10 years was due to the urgency of energy security caused by the unstable oil market in 1973 and 1979. These power plants was adopted non-turn key contract of which Korea Electric Power Corporation (KEPCO) took responsibility for the projects including construction management and the startup of the plants based on the experience from the construction and operation of the previous units.

To meet the electricity demand and overcome the foreign dependency of critical technology, the Yonggwang 3&4 project was started in 1987. In this project, domestic entities became the primary contractors along with foreign sub-contractors in order to absorb the technology necessary for design and construction.

In succession, Korean Standard Nuclear Power Plant (KSNP), a 1,000 MWe PWR, had been developed by improving Yonggwang # 3 and # 4. Two KSNP series, Ulchin 3&4 started the commercial operation successfully in 1998 and 1999 respectively. At this time, 4 additional units are under construction in Yonggwang and Ulchin sites and 2 KSNP+ have recently been decided to be constructed near Kori site. The plants scheduled in the Shinpo site of North Korea are also KSNP series.

As of December 1999, installed capacity consists of 6.7% hydro, 26.3% coal, 10.0% oil, 26.3% gas and 29.2% nuclear. The emergence of LNG as one of the major electric power sources in Korea implies that the portfolio of energy sources will depend not only on the cost of power generation but also on the public preference for environment protection and the manageability of power supply.

Figure 1 shows the long-term view of the electric power sources which shows the installed capacity of 68% increase in 2015 year. Based on the consideration mentioned above, it can be interpreted as follows:

1) Nuclear power plants are expected to be continuously constructed due to energy security and cost effectiveness.
2) Using coal is to be maintained at the same level in terms of composition as in the present since it has a strong merit in cost despite its impact on the environment.
3) The portion of power generation from LNG anticipated to be reduced slightly. However, LNG will remain as one of the major energy sources for power generation for the time being despite its high price in Korea.
4) The power generation from oil and hydro will vary also. However their role will be limited as in the present.

From the above prospect for the various power generation sources, it is clear that the nuclear power is anticipated to increase gradually and remain as a major power source in Korea. Its competitors are coal and LNG power generations of which merits are low cost for coal and less environmental impact and flexibility of energy management for LNG.

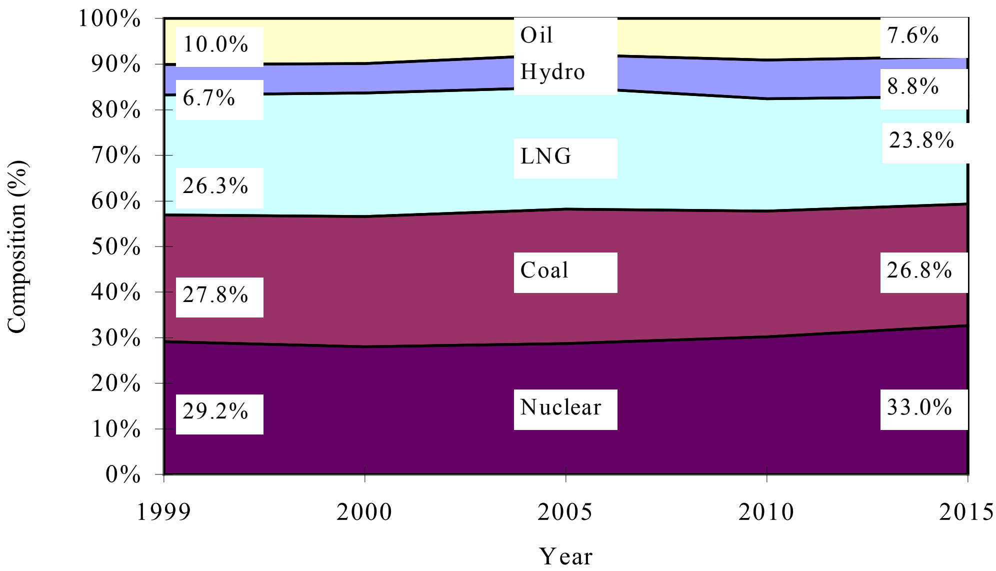

FIG.1. Future outlook for the composition of the electric power source (installed capacity).

2. KOREAN NEXT GENERATION REACTOR PROJECT STATUS AND DESIGN CHARACTERISTICS

In order to further enhance safety and economic competitiveness, a new project to develop an ALWR called KNGR, 1400 MWe PWR, was launched in 1992. Like other ALWRs being developed worldwide, KNGR reflects operating experiences as well as the technology accumulated through the KSNP design. Also, the development of KNGR is closely linked to the construction plan so that the design can be materialized in due time. This project consists of three stages in whole. The first stage, conceptual design stage, was completed on Dec. 1994, and the second stage, basic design stage, was finished also on Feb. 1999. Now the third stage is in progression to the goal of design certification of standard design by the end of 2001.

The KNGR is an evolutionary ALWR based on the current Korean Standard Nuclear Power Plant (KSNP) design with capacity increment. It also incorporates a number of design modifications and improvements to meet the utility's needs for enhanced safety and economic goals and to address the new licensing issues such as mitigation of severe accidents.

The major evolution concept of the KNGR was developed as the result of two years research of the Phase I. During this period, The design concepts had been set up to meet domestic needs and capabilities through reviewing ALWR designs being developed by leading countries in nuclear industry. To establish the safety and economic goals for the KNGR, the ALWRs were also compared quantitatively through safety and economic evaluation. The 42 top-tier design requirements had been established through this comparative study. The major requirements for KNGR are as follows;

- General Requirements
  - Type and capacity: PWR, 4000 MWth;
  - Plant lifetime: 60 years;
  - Seismic design: SSE 0.3g;
  - Safety goals: core damage frequency lower than 10^{-5}/RY and containment failure frequency lower than 10^{-6}/RY.
- Performance Requirements and Economic Goals
  - Plant availability: 90%;
  - Occupational radiation exposure: less than 1 manSv/RY;
  - Construction period: 48 months(Nth plant);
  - Economic goal: 20% cost advantages over coal power plant.

The nuclear steam supply system of the KNGR is designed to operate at rated output of 4000 MWth to produce an electric power output of around 1450 MWe. The major components of the primary circuit are the reactor vessel, two coolant loops, each containing one hot leg, two cold legs, one steam generator (SG), and two reactor coolant pumps (RCPs), and one pressurizer (PZR) connected to one of the hot legs. Two SGs and four RCPs are arranged symmetrically.

The active safety systems consist of the safety injection system (SIS), safety depressurization and vent system (SDVS), in-containment refueling water storage system (IRWST), auxiliary feedwater system (AFWS), and containment spray system (CSS).

The main design concept of the SIS is simplification and redundancy to achieve higher reliability and better performance. The safety injection lines are mechanically 4 trains and electrically 2 divisions without a tie branch between the injection lines for simplicity and independence. Each train has one safety injection pump and one safety injection tank. The common header currently installed in the SIS trains is eliminated and, finally, functions for safety injection and shutdown cooling are separated. Through the IRWST the current operation modes of high pressure, low pressure, and re-circulation can be merged into only one operation mode (i.e. safety injection). The emergency cooling water is designed to inject directly into the reactor vessel so that the possibility of spill of the injected flow through the broken cold leg is eliminated.

The refueling water storage tank is located at the inside of the containment and the arrangement is made in such a way that the injected emergency cooling water can return to the IRWST. The susceptibility of the current refueling water storage tank to external hazard is lessened by locating it at the inside of the containment. The functions of IRWST are as follows; the storage of refueling water, a single source of water for the safety injection, shutdown cooling, and containment spray pumps, a heat sink to condensing steam discharged from the pressurizer for rapid depressurization if necessary to prevent high pressure core melt or to enable feed and bleed operation, and coolant supply to the cavity flooding system in case of severe accidents to protect core melt.

The AFWS is a 2 division and 4 train system. The AFWS is designed to supply feedwater to the SGs for RCS heat removal in case of loss of main/startup feedwater systems. The reliability of the AFWS has been increased by use of two 100% motor-driven pumps, two 100% turbine-driven pumps and two independent safety-related auxiliary feedwater storage tanks as a water source instead of condensate storage tank.

In addition to the added protection and prevention system, severe accident mitigation features and strategy are incorporated into KNGR as follows; 1) hydrogen mitigation system such as passive auto recombiner and hydrogen ignitor, 2) wide reactor cavity area and cavity flooding system, 3) POSRV(Pilot Operated Safety and Relief System) and IRWST, 4) in-vessel retention of molten core by external reactor vessel cooling.

3. KNGR DESIGN FEATURES ENHANCING PERFORMANCE AND AVAILABILITY

Since TMI-2 incident, the added regulatory burden and complexity to the plants have decreased the economic competitiveness of nuclear power. To remedy the situation, nuclear utilities worldwide have examined the requirement for future light water reactors, specially focusing on simplification and increased operational margin, standard design with repeated construction, and integration of operating plant insights and the consideration of safety, operability and constructibility during the design stage.

The recent trend of deregulation of electricity market also emphasizes further plant economy. In a deregulated environment, the large capital cost of nuclear compared to alternate energy sources could be a handicap if not compensated with substantial decrease in generation costs. Shorter construction schedules, simplification of designs together with regulatory stability are thus becoming increasingly important to control costs. In this section, the design improvement consideration for economic viability and performance improvement embodied in KNGR is reviewed.

3.1. Availability improvement

The sensitivity study of economics assessment indicates that the availability of the plant is the most sensitive parameter in ensuring the validity of the cost estimate of electricity. The availability improvement can be achieved by the reduction of outages: both forced outages and planned outages (normally for refueling purpose). Fault-tree model was developed to analyze each system to reduce the number of the forced outage by enhancing the system reliability. To reduce the planned outage duration, KNGR standard outage schedule was generated by reviewing the experiences from KSNP.

3.1.1. *Forced outage evaluation*

KNGR forced outage was preliminarily evaluated based on the operating experience study. The forced outage data of the existing plants are important to predict the forced outage characteristic for the KNGR. For this, trip records were investigated for the domestic PWRs (Kori Unit 1,2,3,4, YGN Unit 1,2,3,4, and UCN Unit 1,2). Trip causes and failure modes, time to restore, and the countermeasures against recurrence are identified for each trip case occurred during the period from 1978 to 1996 based on the annual trip report published by KEPCO. The trip cases occurred during the first one year of commercial operation were excluded in the analysis to get insights of plant lifetime considering the bathtub curve effect.

The focus is to estimate the number of the plant trip by the same root cause of the each trip case. Since the KNGR is at the basic design phase, system designers, equipment vendors, and system engineers of the plants were consulted to predict detailed design and to confirm design modifications and equipment refurbishment in each plant. Some failures were evaluated not to cause plant trip in the KNGR. For example, 5 trip cases occurred due to the failure of RTD (Resistance Temperature Detector) bypass line of the Reactor Coolant System. In the KNGR as well as some of the current plants, the thermo-well type rather than RTD is adopted for measuring temperature of the coolant, by which trip will not occur in the KNGR.

With assessment of the failure causes for the KNGR design, the trip frequency is estimated to be 0.8 per year and 2 days per year for forced outage duration which are considered achievable as a trend of today's domestic operating records.

3.1.2. *Planned outage evaluation*

Planned outages, during which refueling and maintenance (R/M) are performed, are the most dominant factor to the plant unavailability. The operating experiences show that the planned outage duration is characterized by normal outage and extended outage according to plant types and maintenance activity levels. Therefore, a typical 45-day normal outage schedule for a 1,000 MWe PWR was chosen as a basis to assess the planned outage of the KNGR. The RAM(Reliability, Availability, Maintainability) analysis has established the normal outage duration of the KNGR as 42 days by reducing the reactor vessel head handling work loads with little impact on the maintenance activities of the other major components like Steam Generators. Followings are some examples employed in KNGR to reduce the planned outage and operational conveniences.

KNGR has simplified reactor vessel head area by utilizing an integrated reactor vessel head package. An Integrated Head Assembly (IHA) is designed to incorporate all of the reactor vessel head components into one module(Figure 2). The IHA casing is designed so that one can use the multiple stud tensioner during refueling outage. The multiple stud tensioner allows simultaneously detensioning and tensioning of all the reactor vessel studs. This enables the removal of the head area components and reactor vessel head at once. The use of IHA is estimated to save almost 2 days in comparison to the typical seven-day schedule of existing plants. Prior to refueling in existing PWRs, the equipment on reactor vessel area has to be removed and temporarily stored before removing the reactor vessel head.This process has resulted in increasing the overall refueling duration as well as the personal radiation exposure. The IHA is being designed to consolidate the following into an one-package component design: the head lifting rig; lift columns; missile shield; CEDM forced air cooling system; electrical and instrumentation cabling; insulation and reactor vessel head. The IHA lifts the reactor vessel closure head and the head area equipment at one time. Therefore, the amount of critical path required to reach the reactor vessel internals can be reduced.

Furthermore, the KNGR have advanced design features such as a permanent pool seal (PPS) and a quick opening fuel transfer tube blind flange (QOBF) to reduce refueling work load. The PPS is installed between the reactor pressure vessel and the surrounding refueling canal floor to permit flooding above the vessel during refueling. Since the leak-before-break concept is applied to the reactor coolant piping, the possibility of the local pressurization in the reactor cavity is eliminated. It becomes possible to permanently install the refueling pool cavity seal (termed as pool seal). Thus the need for assembling and disassembling temporary pool seals is eliminated. The current blind flanges are installed with bolts to seals off the containment building transfer tube penetration sleeve during reactor operation. Therefore, it takes long time to lift, and to temporarily store it. The QOBF, however, can reduce workload need for lifting it during refueling.

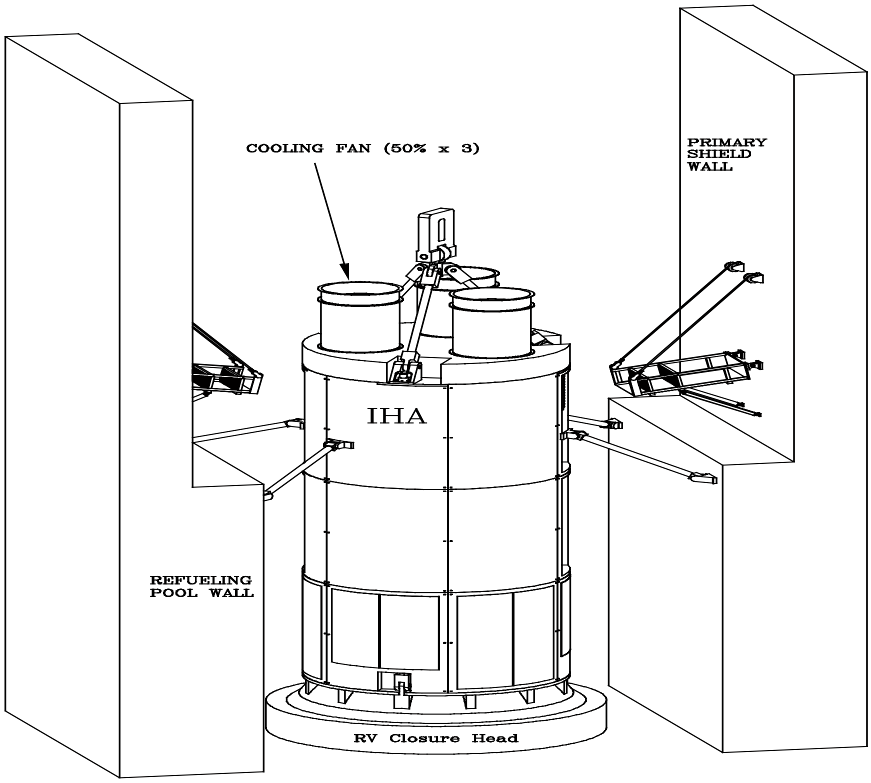

FIG. 2. Integrated head assembly (IHA) for KNGR.

While current domestic nuclear power plants are operated with 12 × 18 month refueling/maintenance(R/M) cycle, KNGR is being designed to have a capability of operating over 18 month fuel cycle, from post refueling startup to the subsequent post refueling startup as per EPRI URD. So for the KNGR, it would take 42 days for R/M cycle per 18 months (i.e. 28 days/a).

An extended outage is another type of outage. It is extended by the works such as ISI (In Service Inspection) and Turbine/Generator overhaul. Ultrasonic Test of the reactor vessel is performed according to the 10-year ISI program. Containment ILRT and turbine/generator overhaul are assumed to perform every 5 years and Steam Generators replacement is assumed once during a 60-year plant lifetime. Although the quantitative estimation of their effects are difficult, the extended outage is estimated to be 6 days per year based on the operating experience of YGN 3 and Kori Unit 1.

One of design improvements in this case is in steam generator replacement. Even if KNGR is incorporated the latest advances in materials and design to maximize the SG integrity, the overall industry experience record combined with the need to extend plant lifetime dictates that the design incorporate provisions for steam generator replacement. The containment polar crane is designed in such a way it can be utilized to remove the steam generators from their cubicles to the area of the equipment hatch. The generators can then be skidded out of containment through the equipment hatch to the auxiliary building where it will be loaded onto an awaiting transport vehicle. Containment layout provides access to piping to facilitate cutting and welding operations that will be required for this evolution

3.2. Constructibility assessment

Shortened construction period reduces the investment uncertainty and reduces the time for recovering investment. In this regard, KNGR is encouraged by the condensed construction schedule set by Tokyo electric Kashiwazaki-Kariwa-6 and 7 units. Following the practice of the Kashiwazaki-Kariwa construction as well as the AP600 construction plan, the way to adopt the advanced construction techniques is under review.

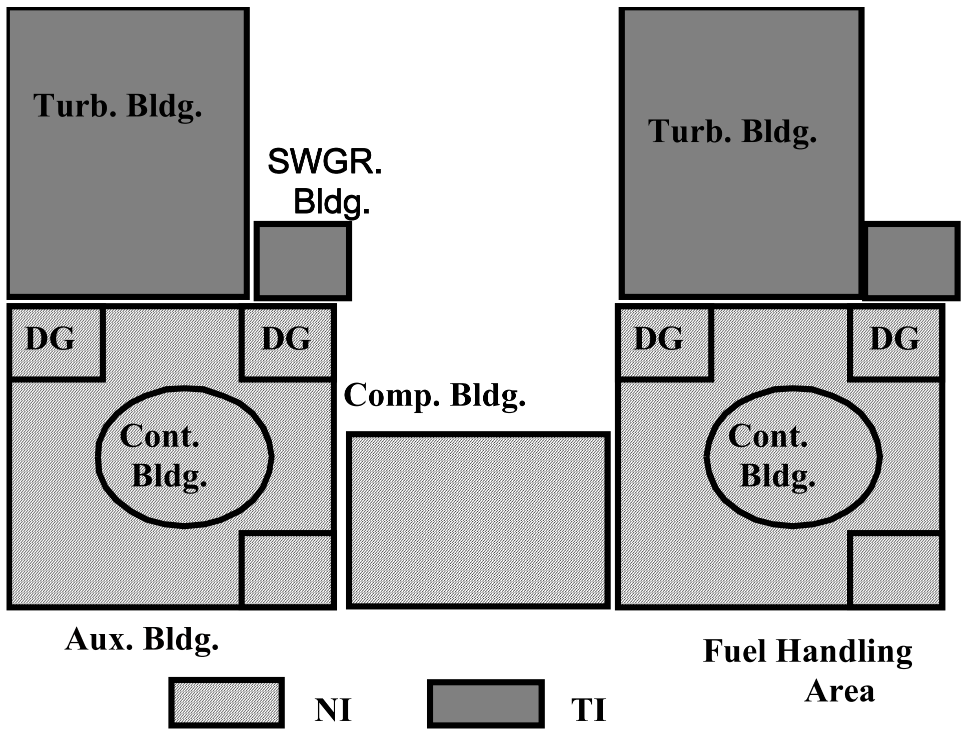

FIG. 3. The general arrangement with complex building of KNGR.

In KNGR design, the auxiliary building surrounds the containment (Fig. 3). Unlike arrangement of KSNP the wrap-around design of KNGR Aux. Building prevents direct access to the containment during the construction. It requires special consideration on the transport and installation of equipment in the containment. Furthermore, the reduction in the Aux. Building construction schedule becomes important since it is tied to the containment construction schedule. To alleviate the problem, the applicability of the over-the-top method is taken into account for NSSS main equipment installation using large cranes. To cut down the Auxiliary Building construction schedule, it is reviewed and decided to adopt the deck-plate method.

In conventional approach, after completion of structural construction by the high-strength shoring and placement of concrete, the installation of mechanical and electrical components commences. The deck plate construction method enables structural construction without the high-strength shoring in the slab concrete pouring. For existing plants, deck plate construction is applied partly as necessary. In KNGR, the deck-plate method will be applied to the entire auxiliary building. It is being integrated to the construction scheduling. The auxiliary building below grade is crowded with mechanical and electrical components. The construction of this area is on the critical path. To adopt the deck-plate construction method, the floor elevation of the Aux. Building has been adjusted. The recent analysis indicates that the use of the deck-plate method costs about the same as the conventional method while reduces the construction period by 15%.

To maximize the benefit of the deck-plate method, one needs to install the equipment modules by the over-the-top method at the same time. We are actively reviewing the equipment modularization. Currently, we have identified about 170 module candidates and are evaluating its effectiveness. Once chosen, the necessary design change will be made to use equipment modules.

3.3. Radiation protection

In Korea, the regulatory agency decided to apply ICRP 60, a major update of ICRP 26, for the ALWR licensing review in Occupational Radiation Exposure (ORE) and ALARA area. ICRP 60 set a lower limit for the occupational dose: an annual dose of 20mSv/a averaged over a period 5 years with an upper limit of 50 mSv in a year. ICRP 26 set the annual occupational dose limit at 50 mSV/a. Since the issuance of ICRP 60, Korean regulatory agency developed new regulatory requirement for radiation protection. The main features of this requirement are 1) for the simplicity of regulation, setting simply 20 mSv/a as annual occupational dose limit and 2) putting more emphasis on ALARA requirement than the compliance with annual dose limits.

The new regulation would increase the cost of electricity generation since it tightens the exposure limit. To minimize its impact, we paid extra attention to defining the hot zone and clean zone. The workers are not allowed to enter the clean zone from the restricted area (hot zone) except through the access building. In the process, radiation work and the worker's movement were carefully reviewed. Also, to reduce the impact of stringent requirement, we have taken the credit in the fuel failure rate based on the improved fuel performance(1% fuel failure rate to 0.25% fuel failure rate).

The reduction in the radiation exposure of workers is important not just for the ALARA but for the operating cost. The KNGR design team has sought to incorporate those lessons learned by the current generation of nuclear power plant to meet the exposure limit of 20 mSv/a set by ICRP60 and to limit the collective exposure to less than 1 person-Sv/a which is one of the EPRI URD goals. Radiation exposure from the maintenance work of KEPCO nuclear plants has been compiled and high exposure maintenance activities have been identified. Not surprisingly, the highest exposure maintenance work is the steam generator tube inspection. The improvement in the access to steam generators and the use of Inconel 690 for SG tubing are some of the consideration given to reduce occupational radiation exposure. Other means of achieving the exposure goals are summarized as follows:

1) The generation of crud is controlled by adopting the material with low cobalt impurities, and maintaining the pH of reactor coolant water in the range of 6.9 to 7.4.

2) Wider use of ion exchangers instead of evaporators in radwaste system.

3) Use of permanent and temporary shielding as an integral part of KNGR design.

3.4. Man-Machine Interfacing System(MMIS)

KNGR is equipped with digitialized Man-Machine Interface System(MMIS) which encompasses the control room systems and Instrumentation and Control(I&C) systems, reflecting the modern computer technology. The KNGR MCR design is characterized by 1) redundant compact workstations for operators, 2) seismically qualified Large Display Panel(LDP) for overall process monitoring of the plant to be shared among operating crew 3) multi-functional soft controls for discrete and modular control, 4) computerized procedure system to provide on one of the workstation CRTs with context sensitive operation guides, operational information, and navigation links to the soft controls for normal and emergency circumstances and 5) safety console for dedicated conventional miniature button type controls provided to control essential safety functions. CRTs and FPD(flat panel display) are extensively used for presentation of operational information.

The human factor engineering is an essential element of the control room facility design and Man-Machine Interface (MMI) design and its principles are systematically employed to ensure safe and convenient operation. Operating experience review analysis, function analysis, and task analysis are performed to provide systematic input to the MMI design.

One of the main features of the I&C system is the use of microprocessor-based multi-loop controllers for the safety including reactor protection and non-safety control systems. Engineering workstation computers and industrial personal computers are used for the two diverse data processing systems, respectively. To keep the plant safety against common mode failures in software due to the use of digital systems, controllers of diverse types and manufacturers will be employed in the control and protection systems. For data communication, a high speed fiber optic network is used. The remote signal multiplexer is also utilized for the safety and non-safety systems field signal transmission to save considerable amount of cables and cable trays. Since the S/W is heavily relied on in full digital MMIS, stringent S/W qualification process will be established and followed for the life cycle of KNGR. The MMIS concept to be implemented in the KNGR design is schematically depicted in Figure 4.

Partial dynamic mockup has been constructed based on the simulator of predecessor plant(KSNP) system models. This facility is used to perform initial verification of suitability of the MMI design. In the forthcoming design stage, the mockup will be expaned for intermediate validation of the design and I&C prototyping will be undertaken for smooth development of KNGR MMIS facility

4. KNGR DESIGN OPTIMIZATION

With the completion of the basic KNGR design at the end of Phase II, we have decided to perform an integrated review on the design and to perform an optimization when necessary. The integrated review of the design was conducted from the perspective of the safety, economics, constructibility, and operation and maintainability

Various issues and questions on the design options in KNGR have arisen during the KNGR development. Some of the issues were from the plant construction and operation department. Based on the construction and operation experiences of the Korean Standard Nuclear Plant (KSNP) series, reviews on design features were requested.

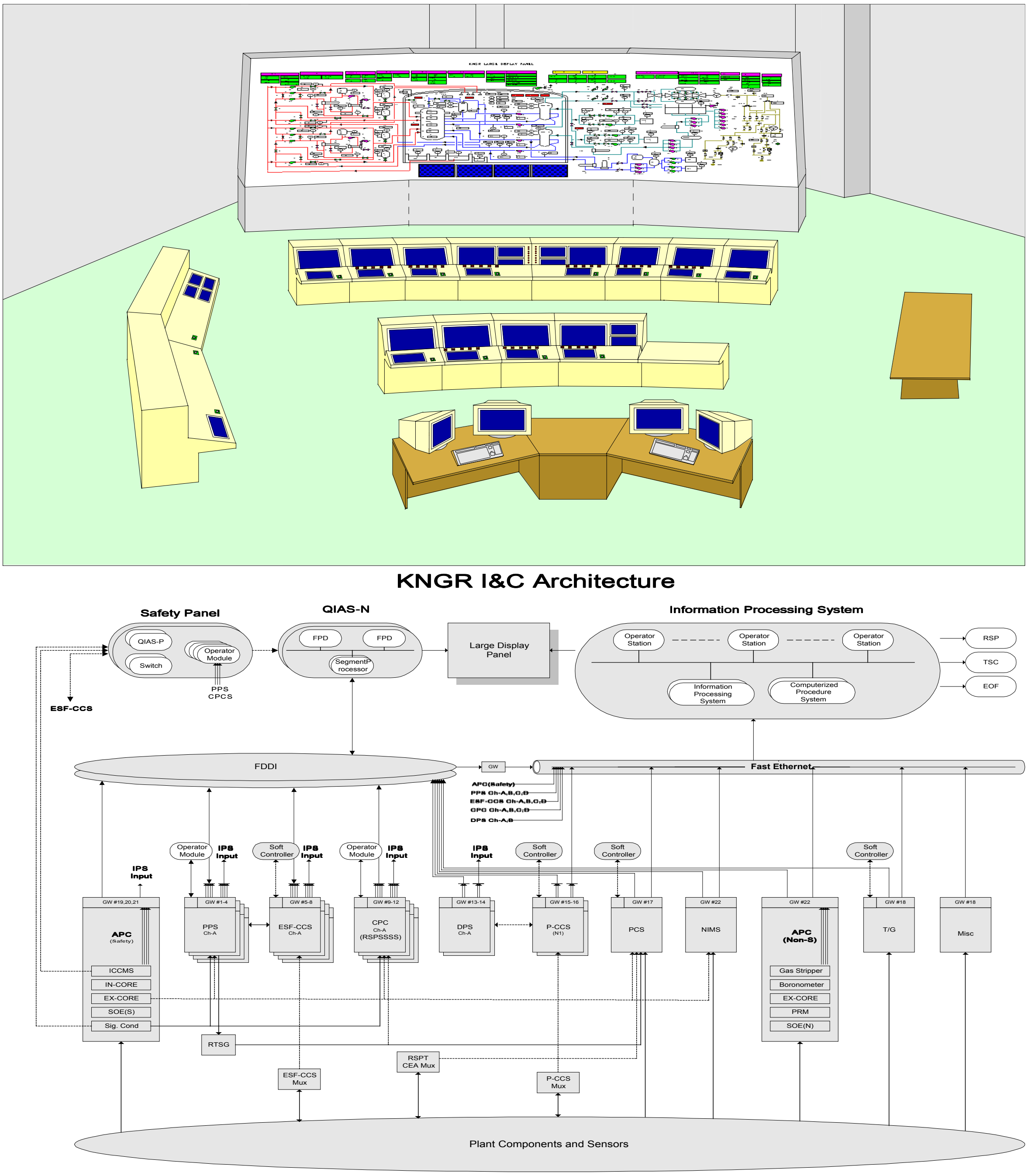

FIG. 4. KNGR man-machine interface system(MMIS).

Issues and questions were also raised regarding the cost-benefit of the safety enhancement features. During the optimization process, all these issues have been collected and grouped. More than twenty items went through the optimization study. Major items considered are 1) Electric power up-rating, 2) NSSS and BOP safety system optimization, 3) Fuel and core design optimization for thermal margin and fuel performance, 4) Containment and severe accident mitigation system optimization, 5) General arrangement (GA) and building structure optimization for convenient construction and maintenance.

Table I shows the summary of the optimization evaluation and its determining factors related to safety, operational or cost impacts. It was estimated the removal of passive secondary condensing system (PSCS) and double containment is cost-effective. The cost benefit for the removal of double containment is more than 10 millions dollars in direct cost savings without a large impact to safety. The final cost comparison after design optimization shows the cost is reduced by 60 million dollars per unit of original cost. The reduction in the construction duration is not credited at this cost assessment. The construction schedule experts estimated that the construction duration could be reduced by 1~3 months by elimination of outer containment.

It is predicted that efficient designs will be achieved in KNGR by incorporation of following factors; 1) Reflecting the operation and construction experiences of Korean nuclear power plants, 2) Gaining the economic competitiveness even though the energy environment will be changed much, 3) Obtaining the consensus for the stable detailed design through mutual consent between the experts inside and outside the industries.

TABLE I. OPTIMIZATION RESULTS BY EACH DESIGN ALTERNATIVES

Group	Items	Results	Remarks
Plant Power Level	Electrical power Up-rating (3,931→4,000MWth)	- 52" Last Stage Blade (LSB) adoption - Increase in Fuel Enrichment and new Fuel in refueling	Cost saving 23M$/unit-year including 13M$ by 52" LSB
NSSS Safety System	-Safety Injection System with DVI -POSRV -Fluidic Device in Safety Injection Tank	- SIS with Direct Vessel Injection - POSRV design - Fluidic Device(FD) Adoption	- No change from basic design
Fuel and Core Design	-24M Fuel Cycle -High Burn-up Fuel -MOX Core Design	- 18 Month fuel cycle - 30% MOX design cap.	- Change to 24 Month Cycle if necessary - Long term R&D item
Containment and Severe Accident	-Double Containment -Cavity Flooding System(CFS) -Hydrogen Mitigation System	- Single Containment & In-Vessel Retention - Replacement of Fusible Plug with MOV (Motor Operated Valve) Passive Auto-catalytic Recombiner + Igniter	Accident mitigation Measure such as IVR adopted
General Arrangement	-Structural Design Optimization	- Compound building - System, Building, Structure optimization	Reduction of 5~10% of volume & bulk material
PSCS	-PSCS Removal	- - Removal of PSCS	Cost-benefit analysis
Performance Requirement	- Load Follow Capability - SG Dryout Time	- Daily load follow - Relaxation of dryout time to 20 minutes	-Excluding frequency control - Related to PSCS removal

5. CONCLUSION

Intensive efforts were made to make the KNGR economically viable and some of the examples on the design optimization, reflection of operating plant insights are presented. In general, the evolutionary ALWRs do have higher investment costs due to more stringent safety requirement. However, the extra costs is counterbalanced by performance advances in plant availability and shorter construction schedule and attention to the cost from the beginning of the design phase. The economic viability could be achieved through several design features incorporated, so it is pretty difficult to say what the final economic competitiveness of KNGR will be. It is emphasized, however, that KNGR is expected to have balanced features in terms of economics and safety.

EPR DESIGN: A COMBINED APPROACH ON SAFETY AND ECONOMIC COMPETITIVENESS

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Abstract

Started in 1991 the French and German cooperation led to common work based on the experience of the two designers FRAMATOME and SIEMENS KWU with all their know how, the most important Utilities in France and Germany operating NPP and the technical supports of the Licensing Authorities GRS and IPSN. The conclusion of that work was the issue in November 1997 and February 1999 of two Basic Design reports for a European Pressurized Reactor (EPR) respectively with a power of 4250MW•th and 4900MW•th. The Basic Design approach was led under two key items: Enhancement of the overall safety level by implementation of design measures to: Make the plant less dependant to common cause failures; practically eliminate all high pressure core melt sequences which could lead to important radioactive releases to the environment; implement specific systems to face severe accident situation with low-pressure core melt. Use of the many years of experiences in two different nuclear designs to reach an overall availability figure over 91%, partly due to design improvements on the safety level. With such an objective, demonstrated by feedback of experience on already operating plants, the EPR project can be proposed as a competitive alternative to the most recent fossil plants.

1. HISTORICAL BACKGROUND UP TO 2000

Considering the development of an evolutionary reactor the European Pressurised Water Reactor (EPR) able to fulfil the requirements of the 21st century, Framatome and Siemens founded their joint subsidiary, Nuclear Power International (NPI) in April 1989.

In 1991, Electricité de France (EDF) and the major German Utilities decided to join the development program by merging their domestic development programs, N4 Plus and REP2000 on the French side and the further development of the KONVOI plants on the German side, with the NPI development program.

One of the major targets of the EPR development is to ensure its licensing in France and Germany. In this context the French and German Licensing Authorities were associated to the project from the very beginning of the design studies. In such a way on line corrections could be envisaged or discussed to consider some specific wishes of the safety authorities in order to avoid any further blockage in the licensing process of the future plant.

A second important development goal was to assure the competitiveness of nuclear power generation with any alternative energy sources. As a milestone for this target economic assessment criteria were defined in 1995 at the beginning of the EPR basic design phase. With the completion of basic design at the end of 1997, it could be proved that the economic target defined two and half years ago was met with a sufficient margin. However, the power generation cost of conventional power plants had decreased drastically during this period of time. This was mainly caused by reduced investment cost for the plants and itself due to international competition and by fuel prices, especially that of natural gas, that fell down to historic low levels, thus diminishing the competitiveness margin of EPR.

In order to fully optimize the generation cost of the future EPR to keep it competitive as far as possible, the suppliers and the utilities jointly decided to address several optimisation options in the EPR design. During this phase, the electrical output of the plant was increased by about 15% to an electrical output of 1750 MW with a NSSS thermal power level of 4900 MW in order to benefit from economy of scale effects. Furthermore, the building layout was optimised mainly by reducing the building volumes. In terms of system design the Residual Heat Removal system (RHR) was combined with the Low Head Safety Injection system (LHSI) and located outside Containment. These measures provided significantly reduced investment and generation costs.

Independently of this demonstration, some concerns were raised about the increased power level. The main argument was that the power increase could induce technical risks in particular with respect of public acceptance and also towards a proper feasibility of the turbo-generator set.

In this context, EDF, with the approval of the project partners, decided to come back to the initial operating power level which was fixed during the first phase, that means a core thermal power of 4250 MW keeping nevertheless the optimisation options on layout and systems design.

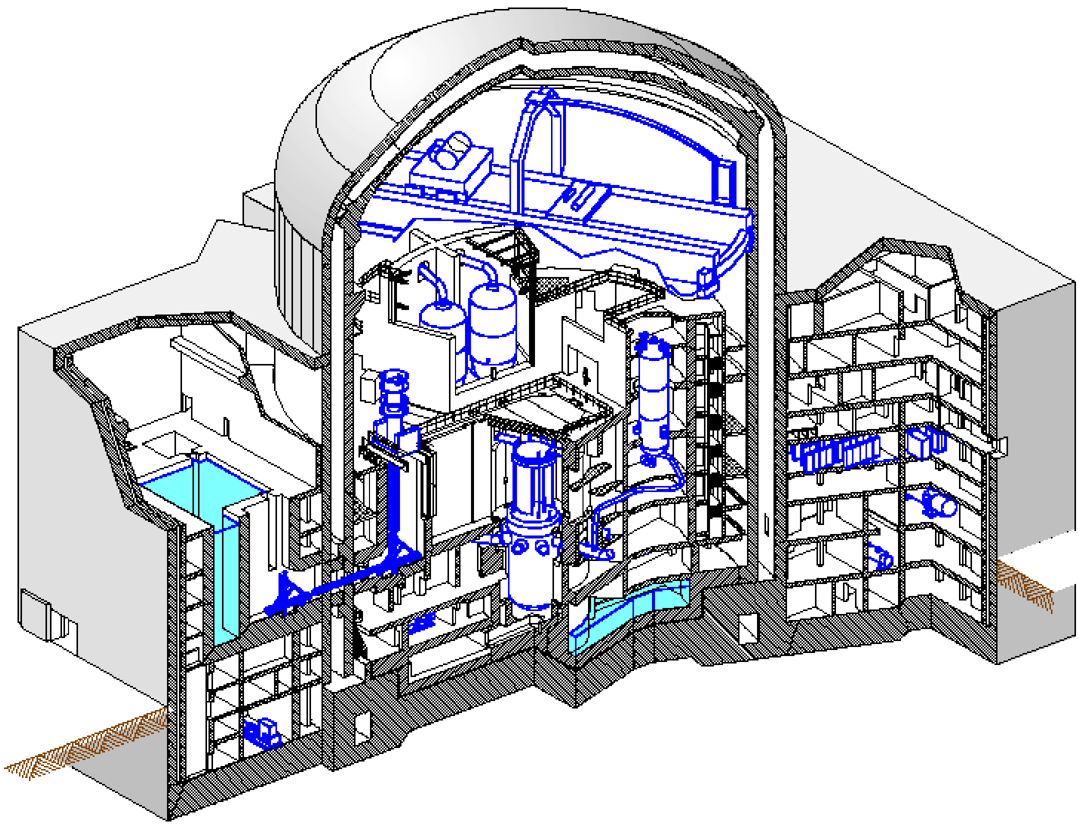

FIG. 1. Plant overview.

2. INTRODUCTION

To increase the overall safety level in a NPP evolutionary design means basically a rise in the investment costs compared to the last units put into operation.

In order to ensure the ambitious competitiveness target of the EPR it became therefore essential to demonstrate a very high availability factor over an extended lifetime. Necessary design improvements to reach this goal became consequently compulsory and all the benefits of safety improvements were to be considered if they could have a positive impact on operation or maintenance. Considering some restrictive national set of licensing rules, the context of an international co-operation involving the licensing authorities at the beginning of the project was favourable for a reasonable updating of these rules - most of them having been issued long before any feedback of experience on NPPs.

The following paper will be organised in presenting first the EPR safety improvements with no induced impact on cost savings and then those that have a consequential positive impact on availability and (or) maintenance. Finally some basic improvements on operation strategies coming the many years of experience in France and Germany will be considered.

3. SAFETY IMPROVEMENTS RESULTING IN ADDITIONAL DEVICES WITH NO COST SAVINGS IMPACT

These safety orientations are only listed here to give a general overview of the approach.

The basic idea leading to these improvements was to improve the result of PSA level 1 (before core melt) and level 2 studies (after core melt). Additional systems were introduced like diversified small diesel generator , a containment heat removal system for severe accident with its dedicated cooling chain , dedicated severe accidents depressurization valves …. These new devices were nevertheless designed with a less stringent functional and mechanical qualification level compared to the traditional safeguard systems to limit the overall cost impact.

– Mitigation of severe accidents up to core meltdown accidents in order to restrict offsite emergency response actions (evacuation or relocation of the population) to the nearby plant vicinity for a very limited duration — No food restriction at the site limits for the second harvest following a severe accident.
– This safety option led to the implementation of the core catcher, core catcher cooling and the containment heat removal system
– External hazards must not constitute a large part of the residual risk so specific design measures were taken to consider a more appropriate approach on earthquakes, explosion pressure waves and military aeroplane crashes. This requirement had a significant impact on layout requirements .
– Additional devices have been implemented to face extreme situations like station blackout or total loss of the heat sink. This requirement led to the implementation of diversified emergency diesel generators and a diversified service water-cooling.

4. SAFETY IMPROVEMENTS INVOLVING SYSTEM SIMPLIFICATIONS AND AVAILABILITY IMPROVEMENTS

4.1. Updating of licensing rules

4.1.1. Boiler rules

The former French rules dealing with overpressure protection were basically issued for fossil plants were the power generated in the boiler can only be reduced when all the coal already present will be fully burned out.

A specific updating was issued for the primary nuclear part of future NPPs in 1974 but no updating was accepted by the overpressure experts on the secondary side where the effect of the reactor scram was only to be considered with a very large delay (last reactor trip). The secondary side overpressure protection led therefore to an over-sizing of the safety valve capacity on the secondary side increasing the risk of overcooling accidents due to inadvertent safety valve opening (7 spring operated 20 % safety valves implemented per steam generator in the French units).

On the other side the German Boiler Rule imposed a global safety valve capacity of 200% per steam generator realize on the German units by two 100% pilot operated safety valves. (One 100% safety valve + one 100% relief valve)

The EPR harmonization process led to another concept considering the reactor scram as a pressure reducing measure, which allows to reduce the overall discharge capacity, provided the reliability and diversity of the provisions related to the reactor scram are similar to those of the protection of the core.

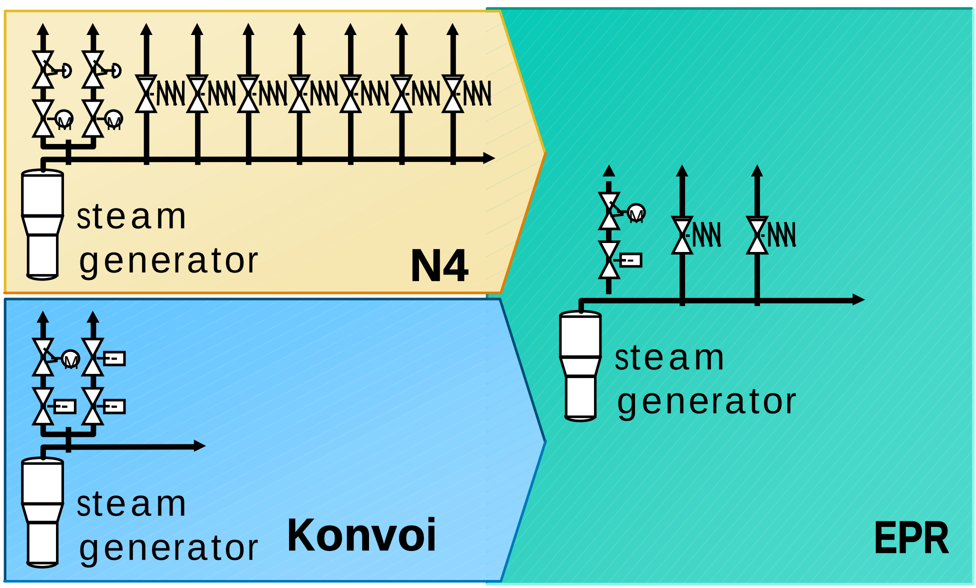

FIG. 2. Secondary side overpressure protection.

The final EPR solution proposes today two 25% safety valves and one 50% relief valve per steam generator .The induced savings are very significant on the equipment cost and on the piping design. The overall safety is also improved due to a reduction of the risk and consequences of overcooling accidents.

4.1.2. Mechanical classification related to the fluid activity: the barrier classification

Up to now a pipe is designed depending on it's operating temperature and pressure. Considering nuclear industry some additional rules were issued related to quality insurance in the design process. Some differences in these types of rules were identified between the French and the German practices.

The new idea issued from the EPR cooperation was to clearly identify a pipe or a system arrangement depending on its fluid contamination level. With the new barrier classification the design effort will be clearly focused where a need of high level quality has been deterministically identified. Three different barrier classes were defined depending on the contamination level of the fluid kept in the "barrier" and on the isolation devices implemented on the barrier . A direct link between the barrier class and the mechanical classification was made.

Cost savings are induced by the avoidance of arbitrary classifications. The gain in safety is to have defined a clear classification approach combining the functional needs (accident mitigation, prevention and management) and the mechanical ones depending on the environmental consequences.

4.1.3. Reconsideration of the requirements imposed to the equipement suppliers

Considering nuclear engineering it appears that important cost savings can be made on a lot of equipment if the design specifications imposed to the suppliers are periodically adapted to the recent technological evolutions.

The first results in that field show that a significant optimization of the specifications can be reached, leading for example to reduce the amount of the imposed nuclear documentation related to an equipment.

4.2. Contribution to defense in depth improvment and unplanned availability reduction

The « defense-in-depth » principle remains the fundamental principle of safety for the nuclear power plants of the next generation, with an implementation of several levels of protection. A good way to contribute to achieve a good level of defense in depth is to reduce the number of significant incidents. This involves to look for improvements of the equipment and systems used in normal operation, with a view to reduce the frequencies of transients and incidents and hence to limit the possibilities of accident situations developing from such events.

4.2.1. Implementation of a limitation system

The best way to reduce the frequencies of transients and incidents is to act and to correct the deviation leading to an incident or a transient by a specific I&C system to be located in-between the protection and the standard control functions of the plant. This system already implemented on the German NPPs since many years is acting with staggered measures in addition to the controls to avoid, when possible, the actuation of a reactor trip protection set points. A degree of independence between this limitation system and the protection system can be asked for to get a valuable PSA benefit and to fulfill the licensing authority requirement related to common cause failures.

The comparison of reactor trip occurrences and the induced plant unavailability in France and Germany showed a clear benefit related to the introduction of the German limitation concept in a new plant design. French and German engineers have analyzed the today existing German limitation system in detail and an optimized set of functions has been retained based mainly on the feedback of experience. A direct generation cost saving is therefore related to the implementation of the limitation system in the EPR.

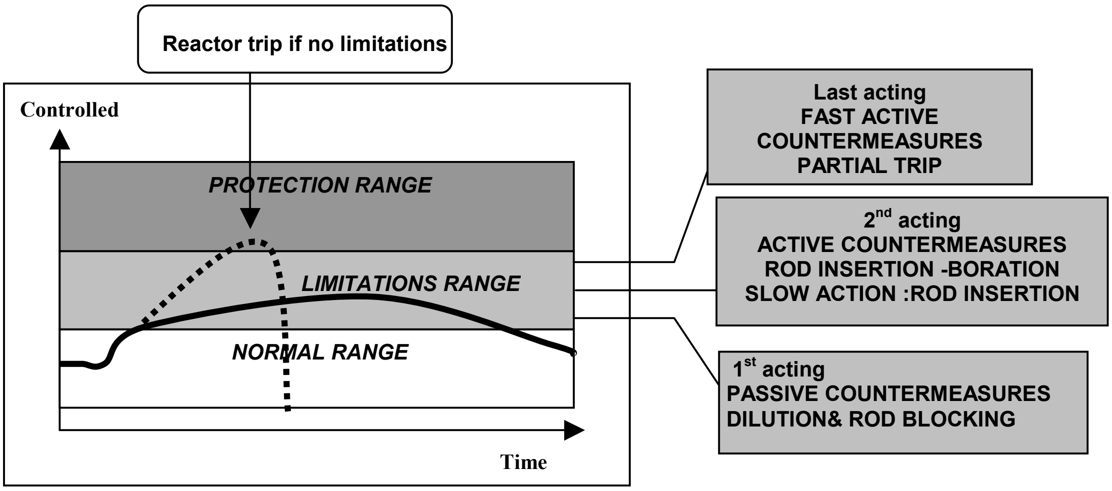

FIG. 3. Limitation staggered actions.

4.2.2. Degree of automation

The main idea for automation is to make the future plant less sensitive to human errors. In the EPR the required degree for automation has been fixed by comparison of the French and German existing solutions: all elements which have an influence on availability or the human failure risk considering the overall safety will be automated in the future EPR (e.g.: main coolant pump automatic start up sequence to avoid a potential human failure considering the complexity of the starting procedure and the consequences on safety and availability in case of a pump failure, automatic start of the third main feedwater pump when required)

The general rule to decide about the need for an automation of a function was:

- when the automation improves the overall safety or availability of the plant
- when the function is time critical during shutdown or start up
- when there is an investment risk
- when the function would disturb the operator in a busy period
- when the function is a boring and repetitive work which is a high potential for human failure.

The additional investment cost to reach this level of automation is minor in comparison to gains identified comparing the feedback of experience on German plant and French ones, the French ones being less automated than the German ones.

4.2.3. Digital I&C — computerized control room

The French N4 plants are equipped with digital I&C and a computerized control room. The latest German plants are still based on hardwired technologies. The new French concept gives advantages in terms of plant surveillance, better overview of the operators, faster capability to restart, more flexible use of computerized operating procedures, more freedom for automation and a much more flexible man machine interface. Sufficient and appropriate information will be made available to the operators for a clear understanding of the plant status, including severe accident conditions. Considering the use of computer techniques reliable diagnosis for operator support are now possible. These elements, for which the N4 feedback of experience is taken into account in an on-line process (the team working on the EPR is in close contact with the people in charge of the N4 man machine interface) contributes to defense in depth due to the improvement of the man machine interface.

A more reliable diagnosis will also help to reach the ambitious objectives for unplanned unavailability.

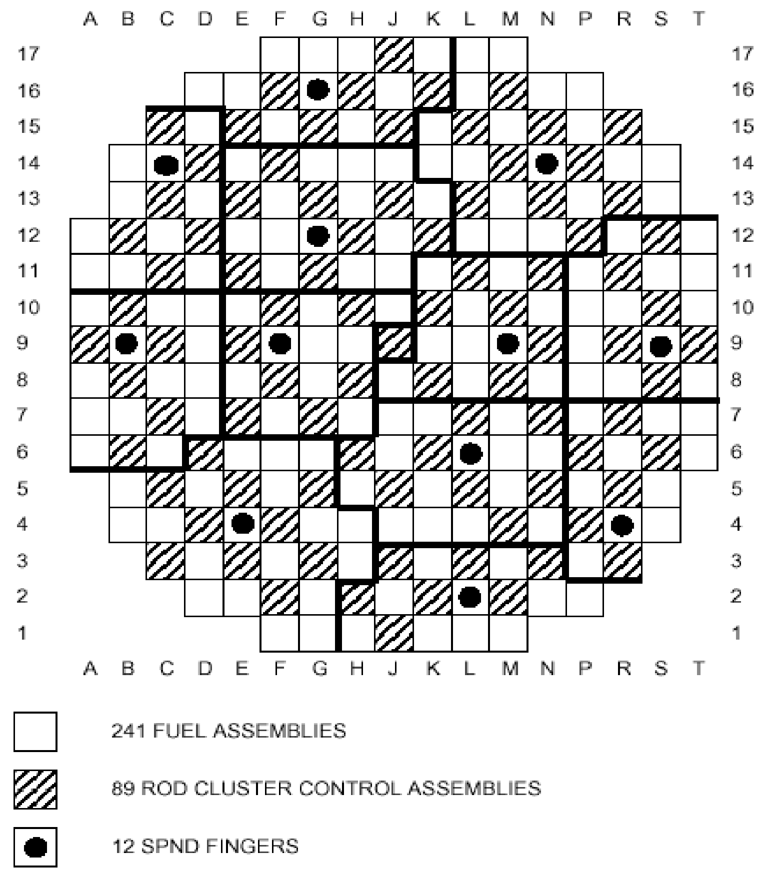

FIG. 4. Fixed in-core instrumentation.

4.3. Contribution to optimized safety margins inducing reduction of unplanned unavailability

4.3.1. Fixed incore instrumentation

German KONVOI plants are equipped with a permanent in-core instrumentation. N4 plants have very accurate movable in-core instrumentation with long time response used to calibrate the less accurate ex-core instrumentation used in the core protection system. With the fixed in-core measurements, periodically calibrated by the very quick movable aeroball system, the on-line core surveillance and protection is far more precise and contributes to a significant increase of the safety margins.

In comparison with the French N4 a long period (up to 3 days) for core stabilization (the plant being no more able to follow the grid requirement) is not required. The quality of the electricity produced becomes therefore much better and the kWh price for such a production will certainly be valuable in a near future. Considering also the safety impact it can be stated that the operations restrictions can be reduced (e.g. Pellet Clad Interaction constraints limiting today the N4 capacity) and the required operation margins to protection actions can be optimized. Operation gets easier, nominal power can be safely increased and the risk to reach the core related protection set-points is reduced assuming the optimized staggered limitation measures (see § 4.2.1).

4.4. Contribution to defense in depth improvment and reduction of planned availability (outage reduction)

4.4.1. General

The outage duration is the main contributor to the overall plant unavailability. Former general maintenance outages had a duration of about 30 days in Germany and somewhat more in France - this difference being mainly linked to specific licensing requirements due to core unloading with regard to the fuel pool cooling. In the last years, the KONVOI plants succeeded in outage reduction with maintenance down to 15 days. This reduction has been performed with modification of systems, tests and practices. The EPR designer considered this feedback of experience with major attention and detailed outage schedules could be presented:

- One with a total duration of 16 days from breaker to breaker considering the Normal Refueling Outage. Such a Normal Refueling Outage includes most of the preventive maintenance tasks except those concerning the safety trains, which are realized in power operation. (see Fig. 6)
- A second one with a total duration of 40 days from breaker to breaker considering the In-Service Inspection outage each 10 years.

4.4.2. N+2 Concept

N+2 Concept or the four train-redundant engineered-safety system structure

In current plants, the French safety systems are designed 2 × 100% (N+1), the German safety systems are 4 × 50% (N+2). From the first point of view, the 2 × 100% concept requires fewer components and looks therefore cheaper. For a 4 × 50% concept, even two additional

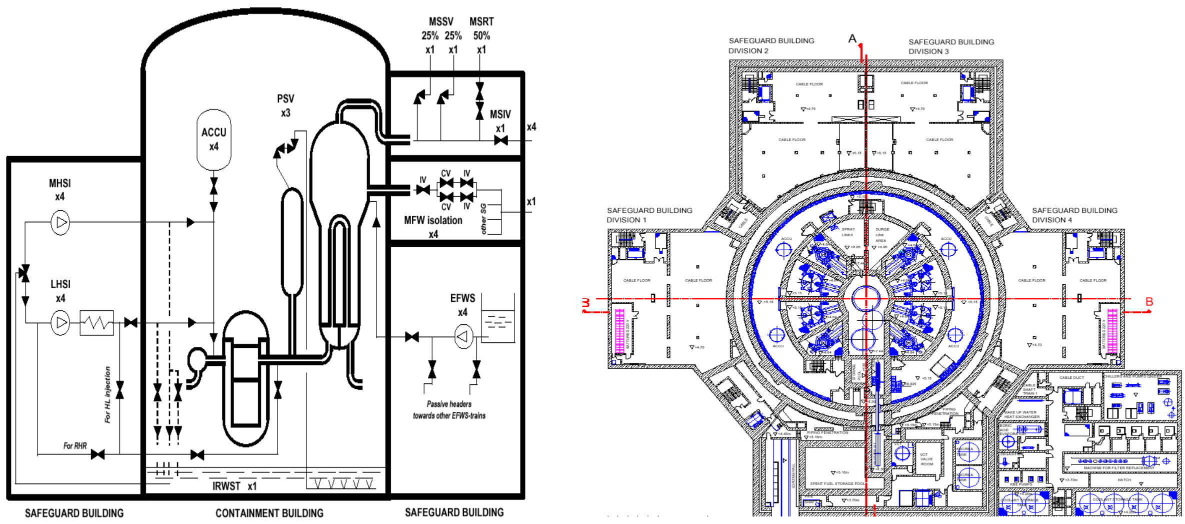

FIG. 5. The 4 train safeguard systems and divisions.

building compartments have to be implemented to separate the trains. After a more detailed analysis, it has been found out that the costs are not much higher with a 4-train concept. Compared to the 2 × 100% concept the components are smaller, more standardized equipment can be chosen and some big pipes like headers between the safety trains can be deleted. Beneath the associated savings this constitutes also a significant safety improvement considering the failure risk.

Considering the subject of this paper the 4 × 50% concept, commonly described as the N+2 concept provides in addition the following advantages directly linked to performance improvements and induced generation cost savings:

- **On power maintenance**

  The N+2 concept allows considering the maintenance of one safety train during power operation and to postulate in addition a single failure in case of an accident without impairing the safety criteria. On power maintenance can therefore be performed with the following advantages:

  1. On power maintenance can be carefully planned and carried out with the best staff in a free time window over the year
  2. On power maintenance reduces the workload during outage and makes the outage planning more safe.
  3. On power maintenance reduces the maintenance costs because the work is decompressed and more can be performed by own staff.
  4. On power maintenance concerning rotating equipment (including the diesel generator sets) is based on a systematic replacement of the equipment by the equipment kept in reserve, the replaced equipment being then maintained in the workshop. Once the maintenance is achieved the maintained equipment is considered as new reserve equipment. On power maintenance can then be planed without any potential failure risk and sufficient time is provided to maintain an equipment in the workshop even if a major problem could appear during the dismounting of the equipment.

- **Increase of admissible repair time allocation**

  The degree of redundancy becomes higher compared to a 2 × 100% or N+1 concept. Therefore the allocation of time to repair in case of unplanned unavailability of an equipment becomes higher too .The repair can therefore be performed with a better preparation and without excessive pressure put on the working staff. Considering human factor issues there is an evident gain in safety linked to maintenance costs reduction.

- **Works on safety trains during outage**

  Also during outage the N+2 concept allows a more flexible planing of control and maintenance work on a safety train. More trains are kept available .due to the time dependant decrease of residual decay heat the required capacities for heat removal are decreasing too. After a certain time the 4 × 50% configuration becomes a 4 × 100% configuration regarding the heat removal function. Maintenance is then possible in outage situations where it would have been impossible to envisage anything in a 2 × 100% configuration (even if the configuration would be equivalent to a 2 × 200% situation) . In certain plant conditions, maintenance can even be performed on two trains in parallel. The combination of these two trains is nearly free depending on the system configuration. A degree of redundancy of 2 × 100% can be kept all the time. So even with more trains the required time for safety train maintenance during outage can be reduced.

- **Cost savings on spare parts**

  With 4 × 50%, the single components are smaller and less expensive. Therefore the costs related to the spare parts management are also reduced.

EPR REFUELING OUTAGE DURATION

ACTIVITIES	Hours	DAY1	DAY2	DAY3	DAY4	DAY5	DAY6	DAY7	DAY8	DAY9	DAY10	DAY11	DAY12	DAY13	DAY14	DAY15	DAY16
TOTAL OUTAGE (BREAKER to BREAKER)	**378**																
Cooldown -RCS opening -Reactor cavity flooding	**58**																
Internal removal - Fuel unloading -Pool draining	**67**																
SG drying & controls	**118**																
Reactor pool flooding to refuelling and core mapping	**59**																
RCS closing and restart to synchronization	**88**																
RCCA reshuffling -Fuel elements contol	**80**																
Draining - Tag in - Works to nozzle low level	**60**																
Train 1-Mechanical & electrical contol	**60**																
Train 2-Mechanical & electrical contol	**60**																
Train 3-Mechanical & electrical contol	**60**																
Train 4-Mechanical & electrical contol	**60**																

Critical path : Non critical path:

FIG. 6. Outage planning taking into account on power preventive maintenance.

4.4.3. System simplification

The Safety Injection and the Residual Heat Removal Systems have been combined and located outside the containment. In order to lower the cost without impairing the safety and reliability, simplifications of the standard well known solutions were examined.

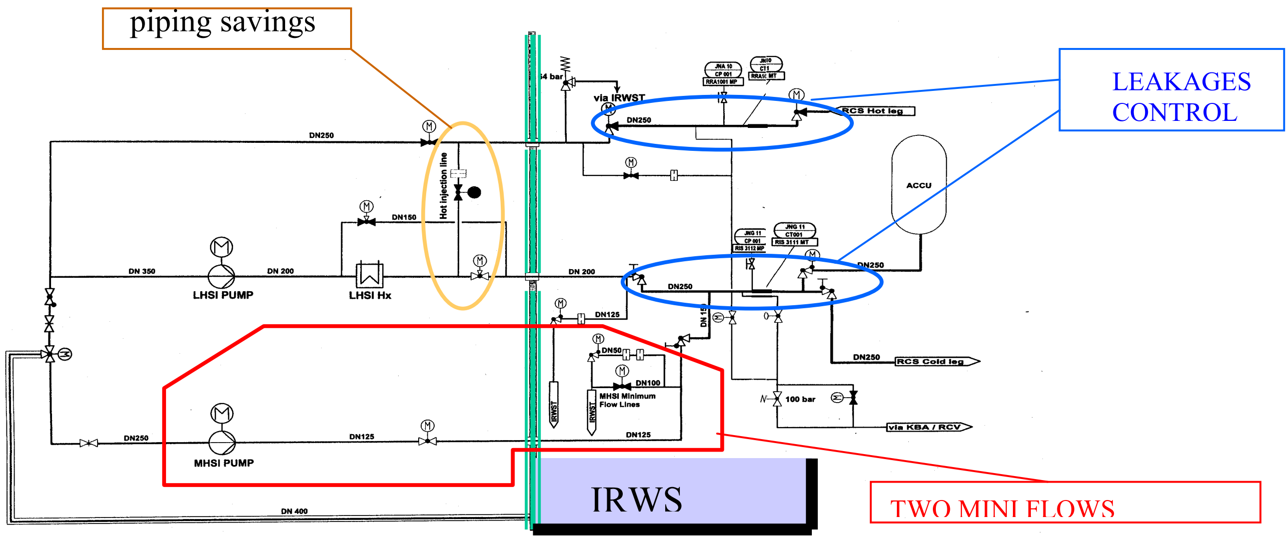

FIG. 7. Combined Safety Injection System IS/ Reactor Heat Removal System.

Three examples can illustrate this engineering process:

- RHR suction line is used as SIS hot leg injection. This approach is made possible because SIS hot injection is manually actuated after 90 minutes in case of a loss of coolant accident to avoid boron crystallization. With such a design, a big pipe and its associated valves together with the injection nozzle could be deleted. This contributes to a certain gain in safety (reduction of the number of nozzles on the primary coolant lines) and in investment costs.
- An overpressure protection in intermediate state had to be investigated concerning the pressurization risk linked with a cold shock on the vessel, in case of Medium Head Safety Injection start up. To solve this problem it has been decided to limit the MHSI pressure by adding a larger mini-flow line sized to limit the RCS pressure to 40 bars. This mini-flow line will only be in service when the reactor is under 120°C/30 bars. By such a design it has been possible to avoid the introduction of a specific cold overpressure protection in shutdown in addition to the gliding set point already decided on the pressurizer safety valves.

  This contributes to a gain in safety by elimination of a potential overpressure transient and in investment by replacing safety valve requirements by small piping.
- It has been required to take into account the feedback of experience related to the mechanical problem of isolated safety injection lines when small leaks through the isolation valves may occur (“Farley-Tihange” problem responsible for several small breaks on the safety injection piping). Implementation of on-line temperature and pressure measurements will allow to identify any small leak through the isolation valve to the primary circuit and the dedicated corrective actions can be taken to evacuate the leak to the Chemical and Volume Control System .

4.4.4. Mid loop operation and nitrogen flushing

Mid loop operation is a plant state with reduced water inventory and core loaded. Reduced inventory is necessary for the vessel head opening. Mid loop operation is advantageous for flushing the reactor coolant system gas phase with nitrogen before opening to reduce the radiological impact on the workers inside containment. Nevertheless considering the mid loop situation with reduced coolant inventory, the safety must be guaranteed.

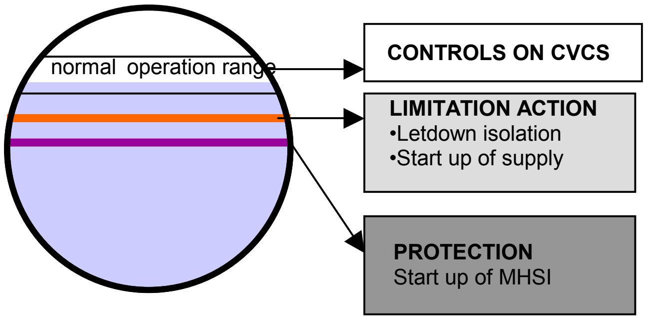

FIG. 8. Staggering of protective actions in mid-loop.

The retained solutions to reach this required safety level are accurate, reliable and redundant mid loop level measurements, automatic level control and automatic make up with the Medium Head Safety Injection in case of level drop. The investment effort is limited due to the already existing I&C and sensors, the safety is improved because radiological releases in normal operation are reduced and staggered countermeasures are implemented in case of problems in this critical situation.

As a consequence, the requirements related to the draining to mid loop can be less stringent.

In the present units this operation is felt as a very critical one and a lot of time is lost due to a permanent checking by the operators .

4.4.5. Maintenance provisions on electrical busbars

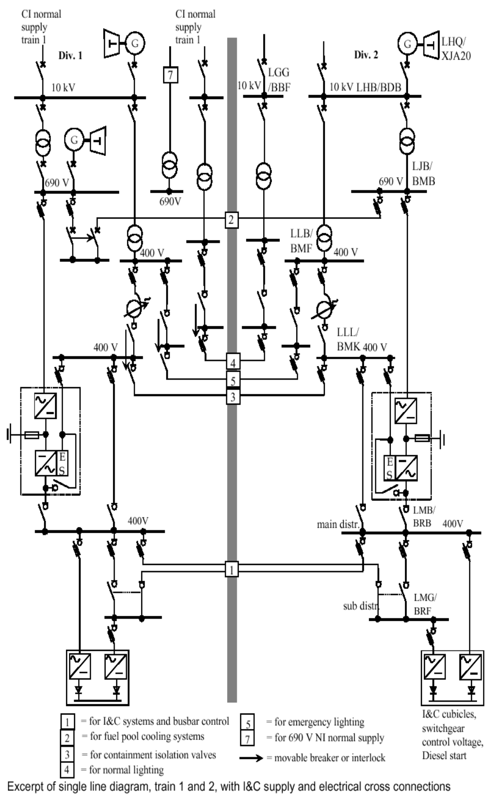

FIG 9 : Single line diagram (train 1 & 2).

The electrical house load grid is separated in 4 trains. Each train is composed of an operational and a safety part. The safety part supplies the safety trains and is fed by the operational part or by the diesel generators. Considering maintenance requirements both safety and operational parts have to be maintained together. During this maintenance phase the mechanical safety train is not available but there some specific consumers that have to be kept in operation like safety lightening, fire detection and fire fighting equipments, site security, communication equipments, lifts, coolant evaporators, etc. No provisions were provided in the former designs for these consumers, so mobile cables are implemented through the corridors during the duration of the maintenance. This practice has a negative impact on an overall safety of the plant due to the communication paths introduced between the divisions.

For EPR, the operators identified from the very beginning the sensible equipment. They fixed the allowed maintenance combinations of electrical busbars and put the consumers on the right busbar combination, to keep the function under voltage in case of maintenance on one or even two trains. For some equipment, double electrical supply has been foreseen from busbars, which physically cannot be isolated at the same time due to specific design choices.

Special busbars have been implemented, which are kept under voltage during the whole outage, their maintenance being performed during power operation

Which such measures the outage planning is more safe, potential human failure are seriously reduced, the overall safety is improved because there no risk of any endangering of the physical separation concept between the divisions and even the operating costs are limited because they is no need to implement expensive movable devices that requires a qualification status considering the material but also the operation procedure to follow (this is quite difficult to guarantee with movable cables in relation with potential human failure)

4.5. Reduction of investments by proper design and layout

A large field for cost savings is also concerned by the design of the nuclear auxiliary systems and not only the safety systems. The solutions used in France and Germany are quite different. The designers knew only their own design and were not able to harmonize themselves. This harmonization process was then made on the Utilities side mainly in comparing each feedback of experience.

Significant improvements were reached to adapt the design requirements to the real operation need on:

- Optimization of the coolant storage capacity;
- Gaseous waste processing system;
- Implementation of the fuel pool outside the reactor building;
- Separate independent waste building.

4.6. Dose reduction and ALARA principle

Dose reduction, which is also a safety concern, means low activity inventory in the systems, proper possibilities for system cleaning before maintenance, proper system arrangement in the layout, adequate shielding, and reduction of the amount of maintenance work. The ALARA principle "As Low As Reasonable Achievable" is a sound balance of all these aspects.

4.7. Dedicated severe accident solution involving cost reduction and safety improvement

In the first issue of the EPR basic design report in 1997, the spreading concept was only considered on a passive basis combined with an active depressurization of the containment atmosphere. Such a solution was quite expensive due to the need of a large width of zirconia layer to support a long term molten corium.

Considering this solution the French and German safety authorities were not so much satisfied with the proposed concept because of the long lasting molten core and because of a permanent steam production in the spreading area.

The project decided therefore to reconsider the design taking into account the safety authorities demand. A mixed solution was then proposed with a combined passive and active mitigation principle leading to a better cooling a the corium.

a) the passive flooding of the corium is kept;
b) the active depressurization is kept too (CHRS –Containment Heat Removal System);
c) to accelerate the corium cooling and to avoid long term steaming in the spreading area an active feed is added , introducing a connection between the CHRS pumping devices and the passive flooding and cooling pipe.

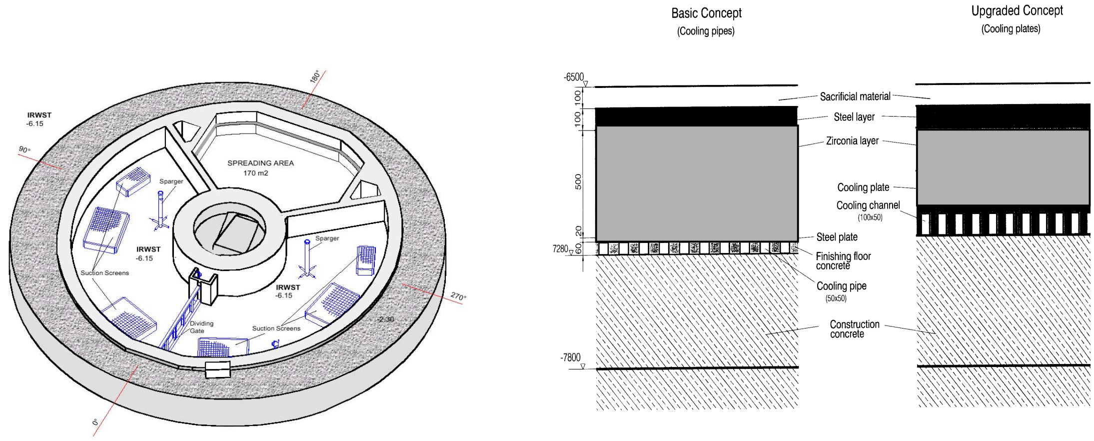

FIG. 10. Impact of improvement of severe accident concept on the Zirconia layer width.

With such a new solution the very expensive zirconia layer width of the spreading area can be seriously reduced, more diversity can be provided related to flooding and long term steaming is avoided. Safety is therefore improved together with cost savings

5. CONCLUSION

The biggest idea to reach a simultaneous enhancement of cost savings and safety can only be reached if the design choices are fully optimized. This basically means that the plant designer is completely aware of the feedback of experience concerning the former designs: such an objective can only be touched if the designers, the operators and their supporting teams on the plant are working close together in the conceptual phase of a project. This cooperation gets even more fruitfully and efficient if it is possible to combine two different designs like KONVOI and N4, which are honesty analyzed getting rid of any national consideration.

It appears also very fruitful to associate the licensing authorities very early in the design because the requests can be discussed on a positive technical basis without endangering the whole structure of the project. As a consequence a better focus on the most important points can reached and even costs savings can be made considering a former proposed design solution .

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