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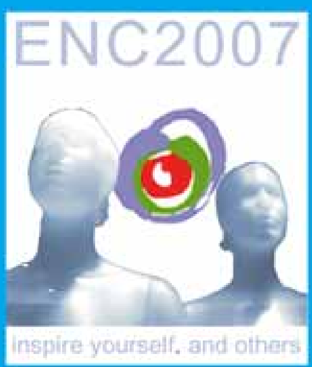

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ISBN 978-92-95064-14-0

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RESEARCH – A SINE QUA NON FOR THE NUCLEAR SECTOR

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ABSTRACT

Not only does 2007 mark the 50th anniversary of the Euratom Treaty, it is also a landmark year for nuclear research in Europe: the 7th Euratom Framework Programme has started and the first European Technology Platform in the nuclear field, a major initiative fostering enhanced cooperation between leading players in nuclear R&D, will be launched on 21st September. Furthermore, in January the Commission published its long-awaited "energy package" framing the coming debate over the "Energy Policy for Europe" and the measures needed to counter the increasingly urgent problems of security of supply, competitiveness and climate change. In particular, the "Strategic Energy Technology Plan" is an extremely important initiative, and as one of the technologies under investigation, the nuclear sector has a real opportunity to influence strategic thinking and decision making. One thing is clear – political and societal acceptance of any nuclear renaissance must go hand in hand with an integrated, effective, well-funded and long-term European research effort.

1. Introduction

As one of the original treaties of Rome, the Euratom Treaty celebrated its 50th birthday in March this year. The Treaty prioritised research, in particular promoting the establishing of a Community research programme in the area of nuclear science funded out of the EU budget. This led to the adoption of the first multi-annual Euratom Framework Programme in the 1980s, a model for research funding that was also borrowed by the more general EC Treaty. Indeed, 2007 also sees the launch of the EU's 7th research Framework Programme (FP-7), heralding a significant increase in the overall EU funding for R&D in general. This is recognition by the European Institutions of the fundamental role that research must play, as one of the three pillars of the Lisbon Agenda, in the EU's overall socio-economic and political strategy for growth, jobs, competitiveness and the development of the knowledge-based society. Within the research field, energy has been identified as a priority in both the EC (non-nuclear) and Euratom programmes, and initiatives are being launched that could herald major changes to our energy supply and usage in the future. In parallel, there are important developments in the area of energy policy and strategy at the European and global level that will also have profound implications. In section 2, these EU initiatives in the areas of energy policy will be outlined, followed in section 3 by a summary of the status of R&D in the nuclear field, including developments within the Euratom FP and the initiative to establish the "Sustainable Nuclear Energy Technology Platform".

2. Developments in EU energy policy

On 8/3/2006 the European Commission (EC) published a Green Paper entitled *"A European Strategy for Sustainable, Competitive and Secure Energy"* [1] that kicked off a major debate on energy supply and security as well as greenhouse gas (GHG) emissions and climate change. The Green Paper clearly stated that an EU energy policy should respond to three main objectives: sustainability, competitiveness and security of energy supply. In this context it considered a number of priority areas, including the diversification of the energy mix, an integrated approach to tackling climate change and the establishing of an EU energy technology plan. Though much of the document referred to energy in

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1 The views expressed in this paper are those of the author and do not necessarily reflect those of the EC

general, without distinction, and entire sections were devoted to energy efficiency and renewables, there were nonetheless important references to nuclear energy and innovative nuclear technology.

2.1 Energy package

Following the Green Paper, on 10/1/2007 the EC proposed an integrated energy and climate change package under the banner *"Energy for a Changing World"* to cut GHG emissions for the 21st Century, increase the EU's independence and security of supply and boost competitiveness. Nuclear energy features at several points and is clearly implicated in a number of the proposed measures. An overarching Communication entitled *"An Energy Policy for Europe" (EPE)* [2] addresses all the challenges and issues. On the subject of nuclear, it recognises the important contribution that nuclear power makes in limiting GHG emissions and in Europe's security and independence of supply. It reiterates that each EU Member State must decide for itself whether to resort to this form of energy, but nonetheless endorses further expansion of nuclear generation providing the highest standards of safety, security and non-proliferation are maintained, as required by the Euratom Treaty. Detailed information on the nuclear sector is presented in the *"PINC"* [3], or Illustrative Nuclear Programme, foreseen under Art. 40 of the Euratom Treaty and presented as part of the overall package. A further document – *"Towards a European Strategic Energy Technology (SET) Plan"* [4] – introduces another initiative of considerable relevance to the future development of nuclear energy in Europe (see 2.3).

2.2 Conclusions to the European Council summit

The initiatives put forward by the EC in the energy / climate-change package were a major topic of discussion by the Member States at the spring summit in Brussels on 18-19 March 2007. This led to the formal adoption of a number of key policies in the area of energy / climate change as well as commitments on use of renewables, biofuels and GHG reduction targets, which must now be developed further by the EU Institutions, leading to possible introduction of new EU legislation.

At the summit, the EU Member States endorsed a strategy to reduce the EU's GHG emissions by 20% relative to 1990 levels by 2020, and to increase the contribution of renewables to 20% of primary energy by the same date. However, the means to achieve these goals, and the respective contributions of individual Member State, have yet to be decided. The Council Presidency, in its conclusions to the summit [5], goes on to confirm that the Member States approve the development of an SET-Plan, and:

- *notes the Commission's assessment of the contribution of nuclear energy in meeting the growing concerns about safety of energy supply and CO2 emissions reductions while ensuring that nuclear safety and security are paramount in the decision-making process;*
- *confirms that it is for each and every Member State to decide whether or not to rely on nuclear energy and stresses that this has to be done while further improving nuclear safety and the management of radioactive waste, and to that effect it:*
  - *supports R & D on waste management, particularly under the 7th FP;*
  - *can envisage the creation of a high-level group on nuclear safety and waste management.*
- *suggests that broad discussion takes place among all relevant stakeholders on the opportunities and risks of nuclear energy."*

The high-level group referred to above will be coordinated by the EC's Directorate-General for Energy and Transport in close collaboration with the national nuclear regulatory authorities, and is a natural outcome from the two years of discussions in Council following attempts by the EC in 2003-04 to introduce binding legislation in the areas of radioactive waste management and nuclear safety. The stakeholder discussion group, or "nuclear forum", would be established in close consultation with the nuclear sector, in particular industry, and other interested groups. Following expressions of interest from both the Czech Republic and Slovakia, the forum will be hosted alternately in Bratislava and Prague. However, it is too soon to know the exact composition and mandates of these two groups.

2.3 The Strategic Energy Technology Plan

The SET-Plan will be the principal vehicle for identifying where action by the EU and Member States can accelerate the development and market deployment of key technologies capable of responding to the challenges of GHG emissions, sustainability, security and independence of supply. Crucially, both nuclear fission and fusion systems are amongst the technologies under consideration. Three time-lines are considered: up to 2020, 2030 and 2050. In [4] it refers to the excellent work carried out by the experts in the EC's Advisory Group on Energy (AGE) over the previous 2-3 years, and their reports [6] represent an objective appraisal of the pros and cons of the various energy technologies.

The SET-Plan is currently being prepared by the EC services for adoption in November and discussion by the Member States at the spring Council in 2008. In March-May 2007, the EC organised a series of hearings with key actors in the respective technology areas, which drew heavily on expertise in existing, and (in the case of nuclear fission – see 3.2) "embryonic", technology platforms. The experts were asked specifically what actions would be needed at EU and national levels to ensure that the full potential of the various energy technologies could be attained. These actions constitute the essence of the SET-Plan. In this regard, both "technology push" or "market pull" instruments, including possible legislation, can be used to help accelerate the various technologies to the market: a "business as usual" strategy is not an option! The challenge for the R&D community has been to provide clear, well argued and rational messages with the correct level of ambition. The European nuclear R&D sector has contributed to this process, their contribution being allied closely to the ambitious vision of the new technology platform (see 3.2) and the corresponding technology roadmaps being prepared for the platform's launch.

3. Developments in EU research

At the end of 2006, the EC launched the 7th Framework Programme (FP-7, 2007-2013) and the 7th Euratom Framework Programme (2007-2011). FP-7 will channel more than €2.3 billion to non-nuclear energy research and, through Euratom, another €1.95 billion will be spent on research into fusion energy (of which approximately half will be for ITER). Both these funding envelopes represent significant increases relative to FP-6 (2002-2006), though in view of the perceived R&D challenges in areas such as renewables, fuel cells, photovoltaics, etc., this level of funding is considered inadequate by many. Regarding research in nuclear fission and radiation protection, some €287 million will be available, though this represents no increase over inflation relative to FP-6. The reasons for this persistent low level of funding are essentially political, stemming from the opposition of some Member States to nuclear power. Nonetheless, FP-7 Euratom [7] will continue to support research on advanced nuclear systems for the benefit of the Community as a whole (see 3.1). Other priority areas include, as in FP-6, R&D on management of radioactive waste, nuclear installation safety and radiation protection, with support for infrastructures and human resources as key cross-cutting issues.

3.1 Euratom support to Generation-IV research

Gen-IV technology represents a revolutionary development relative to current designs of nuclear reactors. It promises vastly improved resource sustainability through the development of fast reactors and associated fuel cycles (enabling at least 50 times more energy to be extracted from the same quantity of uranium), even higher levels of safety than current designs (via increased dependence on passive and intrinsic safety attributes), co-generation of electricity and heat for use in a variety of chemical or industrial processes, and full actinide recycling thereby greatly reducing quantities of long-lived waste for disposal and minimising the risk of nuclear proliferation.

Pre-conceptual design research on the six most promising innovative nuclear concepts is being coordinated at the global level by the Generation-IV International Forum (GIF). Euratom became a member of the GIF in May 2006 following the approval granted by Member States in the Council

Decision of December 2005. Other members include USA, Japan, Korea, Canada, France, UK and Switzerland, with China, Russia and S. Africa all set to join during 2007. The Euratom FP-6 projects in Table 1 are a focal point for the Euratom contribution to the GIF research effort. Further projects in the field of Gen-IV technology are in preparation and will be supported under FP-7.

Project acronym and title	**Key areas of R&D**	**Coordinating organisation & total no. of partners***	**Start date & duration**	**Total budget / EU contribution**
RAPHAEL: Reactor for Process Heat, Hydrogen & Electricity Generation	Performance of fuel, materials and components of VHTR	AREVA (FR) 33 partners (10 countries)	15/4/05 48 months	€19.8M / €9.0M
GCFR: Gas-cooled Fast Reactor	Conceptual design, direct coolant cycles, transmutation, safety, etc.	NNC Ltd. (UK) 9 partners (7 countries)	1/3/05 48 months	€3.6M / €2.0M
HPLWR Phase 2: High Performance LWR – Phase 2	Critical issues and technical feasibility of SCWR	FZK (DE) 10 partners (8 countries)	1/9/06 42 months	€4.65M / €2.5M
ELSY: European Lead-cooled System	Core design, PA, main components & systems, system integration, safety, etc.	ANSALDO ENERGIA S.p.A. Nuclear (IT) 20 partners (12 countries)	1/9/06 36 months	€6.5M / €2.95M
ALISIA: Assessment of Liquid Salts for Innovative Applications	Support action – preparation of future activities/proposals	CEA (FR) 15 partners (9 countries)	Jan. 07 1 year	€250k / €500k
EISOFAR: Roadmap for a European Innovative Sodium-cooled Fast Reactor	Support action – preparation of future activities/proposals	CEA (FR) 14 partners (9 countries)	Jan. 07 1 year	€250k / €500k

*only partners from EU Member States and Euratom Associate Countries can receive EU funding
For a presentation of all FP6 projects refer to http://cordis.europa.eu/fp6-euratom/projects.htm

Tab 1: Overview of FP6 support to Gen-IV systems

The Gen-IV objectives are ambitious and will require an extensive and concerted European programme of research, coordinated at the global level through the GIF. The EU strategy for R&D and eventual deployment of advanced reactor and fuel cycle technology is being developed in the context of the new technology platform (3.2) and will be clearly reflected in the forthcoming SET-Plan.

3.2 The Sustainable Nuclear Energy Technology Platform

A technology platform brings together all key research stakeholders – industry, research institutes, academia, even regulatory authorities – around a common "vision" for research, development and deployment in a particular sector. The stakeholders agree collectively on a "strategic research agenda" and then cooperate using their own financial and human resources to implement this agenda. So far, some 30 technology platforms have been launched in Europe, including several in the area of non-nuclear energy technology.

The first technology platform in the nuclear field – the *Sustainable Nuclear Energy Technology Platform (SNE-TP)* – will be formally launched on 21/9/2007 at a major event to take place in Brussels [8] under the auspices of the EC and in the presence of the EC Commissioner for Research, Janez Potočnik. Members of the European Parliament will also play an active part, as will many high ranking officials from the nuclear sector. The scope of SNE-TP includes nuclear installation safety and nuclear systems (including P&T and the fuel cycle), related research infrastructures and human resources. It is built around three "pillars": the safety of current generations of light-water reactors; the development of next generation fast reactors with closed fuel cycles and full actinide recycling; very/high temperature reactors (V/HTR) for the cogeneration of both electricity and process heat for industrial applications. The platform will be the key technical nuclear forum in Europe, and will ensure that Europe's world-leader status in nuclear technology can be further consolidated and extended to include advanced nuclear technology. It is an essential complement to the SET-Plan initiative.

4. Conclusions

The EC acknowledges the role played by nuclear power and the potential it has to respond to the energy challenges faced by Europe. The Council endorses this view, at the same time recognising that the choice whether or not to resort to nuclear energy must be taken at national level. It also reiterates the importance of safety and waste management, proposing that a high-level group of regulators be set up, and believes a forum to discuss the pros and cons of nuclear in general should also be established.

The Euratom fission programme in FP-7 has not benefited from the same increase in funding witnessed in the fusion and non-nuclear energy sectors. However, it continues to support important Community research in the area of nuclear safety and systems, including advanced nuclear technology, waste management and radiation protection, thereby stimulating and further structuring the research efforts across Europe. This process has already started during FP-6, thanks mainly to the new funding instruments such as Integrated Projects and Networks of Excellence, but must now be consolidated by the establishing of a technology platform thereby better integrating contributions from national – and industrial – programmes. The Sustainable Nuclear Energy Technology Platform is a major initiative that is attracting widespread support, and should enable the available resources in this field to be better utilised. The next five years will mark a crucial period in nuclear research. In particular, the viability of the various Gen-IV systems will continue to be investigated, culminating in decisions on pilot/demonstration facilities to take this technology through to industrial deployment.

The SET-Plan is a bold and challenging initiative. It will cover the full range of low carbon energy technologies, and is being prepared by the EC, with input from a range of experts, for discussion by the Member States at the spring 2008 Council. The Plan clearly demonstrates the broad portfolio approach of the EC, eloquently summarised in a speech by the Commissioner for Research, Janez Potočnik:

"The EC believes that the answers to the EU's energy problems lie in developing a diverse mix of options supported by appropriate strategies and policies. That is why we are funding, through the FPs, a comprehensive research effort looking at a broad range of energy technologies; from renewables, through clean coal, to nuclear fusion and fission. Many questions are currently being asked in all these areas and society as a whole is not yet in a position to provide adequate responses. A well-focussed and effective Community research programme is helping to deliver these urgently needed answers … Ultimately, the decision whether or not to use nuclear power – just like any other energy source – is a political and societal one taken at the national level. However, this should be a decision based on knowledge, not one taken in ignorance. Research can and must supply this knowledge"

5. References

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Track 1

New Reactor and energy technologies

Session 17.1.1:

Advanced reactors

IRIS – AN ADVANCED, GRID-APPROPRIATE PWR FOR NEAR-TERM DEPLOYMENT

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ABSTRACT

With the resurgence of nuclear power there is an increasing need for a range of new reactor designs, including smaller units of several hundred MWe. Such reactors fit not only the developing and smaller countries or electric grids, but also provide commercial flexibility to mature markets with large grids by matching the growth, reducing risk, and minimizing financing resources. The International Reactor Innovative and Secure (IRIS) offers an advanced, modular 335 MWe design. IRIS features an integral primary system configuration with all main components located within the reactor vessel. This configuration enables a simplified design with enhanced reliability and economics and supports its safety-by-design™ approach, which results in exceptional safety characteristics. In addition to electricity-only production, IRIS is well suited for cogeneration, including water desalination, district heating, and process steam generation. IRIS is being developed by an international team, led by Westinghouse, incorporating 19 organizations from 10 countries, about half of them European. IRIS development started in 1999 and has reached the level of maturity indicating potential for being commercially offered by the mid of next decade. The preliminary design has been completed and the testing needed for design certification has started last year. The centrepiece of the experimental program is the integral system performance testing to be performed at the SIET facility in Italy. The pre-application review process with the US NRC was initiated in 2002 to address long-lead items, and enable obtaining the Final Design Approval (FDA) by 2013. Economic analyses indicate that IRIS will be competitive with other nuclear and non-nuclear energy sources, whether deployed gradually in single units in smaller grids, or in multiple twin units for larger grids. Additionally, IRIS fits well the recently announced US DOE initiative, GNEP (Global Nuclear Energy Partnership) aiming to support worldwide expansion of the use of nuclear energy in a responsible and proliferation resistant manner. Within the GNEP framework, IRIS can in the near term offer an advanced reactor design to satisfy needs for smaller, grid-appropriate reactors.

1. Introduction

With leading indicators predicting the renaissance of nuclear power, there is an ever increasing need for a range of advanced reactor designs to satisfy diverse needs of worldwide markets. While some requirements, such as safety, security and economics, are common to all applications, others such as reactor size (power level) are market or application dependent. While for developed, fast growing

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markets large reactor units may be preferred, smaller units are also needed, both for smaller/emerging markets (due to financial and electric grid limitations), as well as for larger stable markets with limited growth rate, providing better match to needs and improved cash flow. The Westinghouse portfolio of advanced power plants offers designs, from the larger AP1000, to the medium and small size IRIS and PBMR, to satisfy the needs of all customers.
Many emerging nations and energy markets with small grids will start introducing nuclear power plants in the next decade. Due to their grid size, units larger than a few hundred MWe are not optimal or in many cases even not technically feasible. With its smaller size (335 MWe), simple design and operation, exceptional safety, moderate cost, limited financing burden, and possibility to gradually add capacity by adding more modules, IRIS offers an optimum solution that is technically and economically viable and technologically immediately acceptable. This same size fits well developed markets having large grids with a limited growth and frequently small margin in transmission lines, that need to improve energy security through redundancy, and optimize investment by "just-in-time" build.

2. The IRIS Project

IRIS[1-5] represents the latest evolution of the LWR technology which has been the overwhelming mainstay of nuclear power development and deployment. While the integral configuration in general, and the IRIS design in particular, embodies advanced engineering solutions, no new technology development is necessary and therefore a demonstration prototype is not required to attain design certification from the regulatory body. A first of a kind (FOAK) commercial plant is thus envisaged past the mid of the next decade, as shown in the project schedule in Table 1.

Program started	1999
Assessed key technical and economic feasibility	2000
Performed conceptual design, preliminary cost estimate	2001
Initiated NRC pre-application licensing for Design Certification	2002
Completed NSSS preliminary design	2005
Initiated testing necessary for NRC Design Certification	2006
Complete testing	2010
Start formal Design Approval with NRC	2010
Obtain Final Design Approval from NRC	2013
Ready for deployment	2015-2017

Table 1: IRIS project schedule

The US Department of Energy (DOE) unveiled a major new initiative in February 2006 [6], the Global Nuclear Energy Partnership (GNEP). Its ultimate objective is to safely expand nuclear energy without increasing proliferation concerns. One of its key elements is the development of smaller-scale grid-appropriate reactors: "These reactors will be safe, simple to operate, more proliferation-resistant and highly secure. … The GNEP seeks to form international partnerships to accelerate certification of marketable designs, and deploy operational demonstration plants…"[6] IRIS satisfies very well the GNEP requirements and has been selected by DOE to exemplify such smaller reactors.

From its very beginning, IRIS has been developed by a strong international team comprised of world renown organizations and led by Westinghouse.[7] The team currently includes 19 organizations from 10 countries, over 4 continents, with a strong European component (about half the organizations and half the countries are European). These organizations represent leading nuclear manufacturers, academic institutions, national laboratories and power producers. Universities are vibrant team members with more than one hundred students involved to-date, a majority of them having conducted graduate theses at the master or doctoral level.[8]

3. Innovative Approach and Solutions in the IRIS Design

While firmly based on the proven LWR technology, the IRIS project has introduced many engineering and project innovations which define its unique characteristics, such as:

- Design: based on simplicity to simultaneously improve safety, reliability, and economics
- Primary system: integrated primary system design
- Safety: a safety-by-design™ approach
- Security: enhanced and easier to implement security, based on its design characteristics
- Proliferation resistance: enhanced through extended refuelling cycle, while retaining use of current demonstrated fuel, facilitating international safeguards
- Economics: simplicity, modularity, and economy of serialization in lieu of economy of scale
- Operation: Simple operation, minimizing need for operators action in incident situations
- Construction: Less than 3 years construction period, reduced nuclear in-country infrastructure required
- Workers safety: significantly reduced dose to personnel in operation, maintenance, and ultimately in decommissioning activities
- Project management: development by an integrated international team (led by Westinghouse) of 19 organizations from 10 countries, with all team members contributing resources to IRIS development
- Research: effectively incorporating national laboratories and academia in the development efforts
- Market segment: targeting markets and utilities that require a smaller-scale reactor design, due to grid size or financial limitations
- Market penetration approach: reaching more markets through the international team and wide partnership in team member countries
- Licensing: based on its outstanding safety, aiming at achieving licensing with lessened and, if possible, eliminated off-site emergency planning requirements. Licensing supported through a multinational design evaluation program (MDEP) will be pursued.

IRIS is innovative in design – employing an integrated primary system that incorporates all the main primary circuit components within a single vessel, i.e., the core with control rods and their drive mechanisms, eight helical coil steam generators with eight associated fully-immersed axial flow pumps, and a pressurizer (Fig. 1).
The integral configuration offers intrinsic design improvements:

- Pressurizer: A dedicated pressurizer is eliminated, as the vessel head will fulfil the function. Much larger volume/power ratio gives much better control of pressure transients. Additionally, no sprays are required.
- Primary coolant pumps: The axial fully immersed pumps result in no seal leak concerns, no possibility for shaft breaks, and no required maintenance.
- Internal CRDMs: This solution eliminates head penetrations and possibility of seal failures, as well as any future head replacements.
- Steam generators: With the primary coolant outside, tubes are in compression, thus eliminating tensile stress corrosion cracking.
- Thick downcomer: The 1.7m thick downcomer reduces the fast neutron flux on the reactor vessel by 5 orders of magnitude. This leads to "cold" (i.e., not activated) vessel, almost no outside dose, no vessel embrittlement, and no need for surveillance. The vessel is essentially "eternal", and decommissioning is simplified.

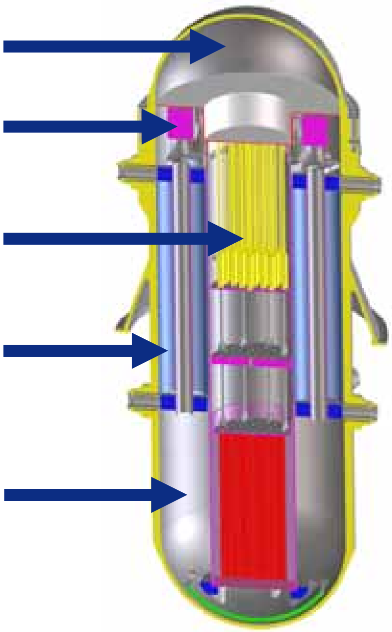

Fig. 1: Integral configuration and components

- Compact layout: While leading to a larger reactor vessel, the integral layout results in a smaller containment and overall a more compact site, with positive impact on safety, security, and economics (Fig. 2).
- Maintenance: Intervals between maintenance outage can be extended to 48 months. A core design has been developed enabling uninterrupted operation for up to 4 years if so desired.

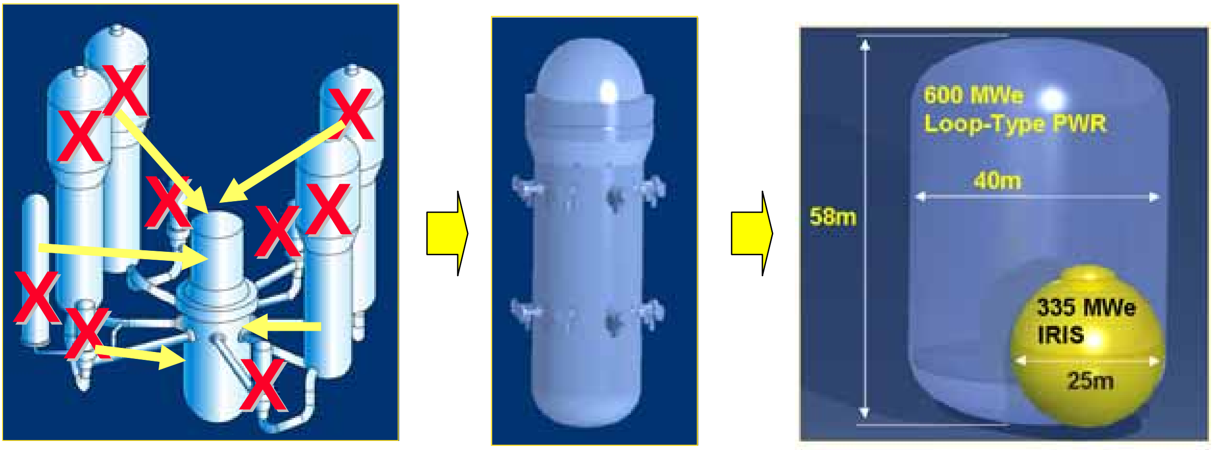

Fig. 2: Integral configuration and compact containment layout

In addition to the design improvements, the integral configuration offers very significant intrinsic safety advantages, which have led to the unique IRIS safety approach articulated over three tiers.
The first tier is safety-by-design™ which aims at eliminating by design the possibility for an accident to occur rather than dealing with its consequences. By eliminating some accidents, the corresponding safety systems (passive or active) become unnecessary as well.
The second tier is provided by simplified passive safety systems, which protect against the still remaining potential accidents and mitigate their consequences.
The third tier is provided by active systems, which are not required to perform safety functions (i.e., are not safety grade) and are not considered in deterministic safety analyses, but may contribute to reducing the core damage frequency (CDF).
Table 2 summarizes the IRIS design characteristics and their safety implications, together with their impact on accidents, with particular emphasis on condition IV events. Systematic implementation of the IRIS safety-by-design™ approach has enabled outright elimination of 3 out of 8 Design Basis Events (DBEs) typically considered for LWRs. Severity has been reduced for another 4, while only one DBE (fuel handling accident) remains the same.
Furthermore, by consistently applying the safety-by-design™ approach (guided by use of Probabilistic Risk Assessment from the very beginning of the design process), IRIS has lowered the predicted Core Damage Frequency (CDF) to below 10^{-7}/yr and Large Early Release Frequency (LERF) to below 10^{-9}/yr. While the present nuclear power plants already demonstrate remarkable safety, further safety advances achieved in IRIS may enable plant licensing with a reduced or even eliminated off-site emergency planning zone[9]. This feature not only should increase public acceptance but will produce a positive financial impact by reducing infrastructure cost, as well as enabling efficient co-generation for desalination, district heating and process heat.

To enable deployment in the next decade, consistent with the projected worldwide energy needs growth, the reference IRIS core design is based on the current, available and demonstrated LWR fuel technology. However, the design includes features to enable future improvements in fuel management and further enhance most of its proliferation resistance. This will be achieved by gradually increasing fuel discharge burnup and cycle length, requiring that in parallel improved fuel performance is demonstrated.

To further simplify safeguards and make them more effective (as well as to improve economy) IRIS extends the fuel reloading interval. It is anchored to the IRIS optimized maintenance with outage required only every 48 months, therefore directly enabling refuelling interval of up to four years. The reference IRIS design with 4.95% UO_2 fuel presently enables a 3 to 4 years cycle.[10] In the future, employing UO_2 or MOX fuel with ~10% fissile content (still comfortably below the HEU limit), an eight-year refuelling cycle with a short maintenance outage halfway through will be feasible.[11] With a four- or eight-year refuelling cycle, and the possibility to limit spent fuel kept at site to one core-load, safeguards will be even more simple and effective, and any diversion timely identified.

IRIS Design Characteristic	Safety Implication	Accidents Affected	Condition IV Design Basis Events	Effect on Condition IV Event by IRIS Safety-by-Design™
Integral layout	• No large primary piping	• Large break LOCAs	Large break LOCA	Eliminated
Large, tall vessel	• Increased water inventory • Increased natural circulation	• Other LOCAs • Decrease in heat removal various events		
	• Accommodates internal Control Rod Drive Mechanisms	• Control Rod ejection • Head penetrations failure	Spectrum of Control Rod ejection accidents	Eliminated
Heat removal from inside the vessel	• Depressurizes primary system by condensation and not by loss of mass	• Other LOCAs		
	• Effective heat removal by Steam Generator and Emergency Heat Removal system	• Other LOCAs • All events requiring effective cooldown • Anticipated Transient Without Scram (ATWS)		
Reduced size, higher design pressure containment	• Reduced driving force through primary opening	• Other LOCAs		
Multiple, integral, shaftless coolant pumps	• No shaft	• Shaft seizure/break	Reactor coolant pump shaft break	Eliminated
	• Decreased importance of single pump failure	• Locked rotor	Reactor coolant pump seizure	Downgraded
High design-pressure steam generator system	• No Steam Generator safety valves • Primary system cannot over-pressure secondary system • Feed/Steam System Piping designed for full Reactor Coolant System pressure reduces piping failure probability	• Steam generator tube rupture	Steam generator tube rupture	Downgraded
		• Steam line break • Feed line break	Steam system piping failure	Downgraded
Once through steam generators	• Limited water inventory	• Feed line break • Steam line break	Feedwater system pipe break	Downgraded
Integral pressurizer	• Large pressurizer volume/reactor power	• Overheating events, including feed line break • ATWS		
			Fuel handling accidents	Unaffected

Table 2: Implementation of safety-by-design™ in IRIS

IRIS has been designed to satisfy all the current licensing requirements with the U.S. NRC. However, an additional option for IRIS licensing is being pursued through the NRC's recent multinational design evaluation program (MDEP), which would facilitate its worldwide deployment. According to Ref. [12], "…NRC has formally approved moving forward with implementation of [MDEP] aimed at improving the effectiveness and efficiency of regulatory design reviews for new reactors". MDEP is envisioned in three stages, with increased level and formalization of international cooperation in licensing. One of the objectives of Stage 1 is to identify areas where national standards overlap with the U.S. regulations and where foreign regulatory expertise could complement the expertise of the NRC's staff. This provides an opportunity to regulatory bodies of countries potentially interested in IRIS to join the IRIS multinational licensing efforts, become familiar with relevant characteristics of the IRIS design while strengthening their expertise in licensing. This route has been already taken by the Croatian regulatory agency, which in December 2006 has requested MDEP participation in the IRIS review, a request accepted by NRC.

Need for potable water is even more critical in many developing nations than the need for energy/electricity. Moreover, the co-generation market segment has specific needs as compared to electricity-only generation. Transportation of co-generation produced water, process heat/steam,

district heating over long distances is not practical, and in many case several smaller, geographically distributed plants are preferable to a large single plant.
With its moderate power level, simple design, and a possibility to attain licensing with a reduced Emergency Planning Zone (EPZ), and thus locating plants closer to end users, IRIS is well suited for co-generation. A preliminary design of an IRIS desalination co-generation plant[13] has been performed by the IRIS team member, OKBM, which has a vast experience with desalination units. Several IRIS team members (including from Brazil and Mexico) have performed economic studies demonstrating attractiveness of IRIS to fulfil the combined electricity and potable water needs in arid regions of their countries.[14-16] Lithuania has examined use of IRIS in district heating. Application of IRIS to ethanol production in the U.S. is also being considered.

4. Some Current Efforts

IRIS economic competitiveness in all markets is achieved through a synergistic positive effect of several technical and economics factors that add up to counterbalance the negative impact of the economy of scale. To mention just one of the positive factors, IRIS enables a gradual increase in generating capacity to match growth needs. Financial risk and needed investment capital are thus largely reduced since the staggered construction of modules deployed several years apart enables income to be generated from previous unit(s) while the next unit is being built. Details of the economics analysis are presented in a companion paper.[17]
Preliminary economic analyses presented previously [18] have recently been extended to include a relatively large contingency, added up front to the estimated cost to address uncertainties and any unforeseen factors. Additionally, the whole first core cost has been accounted as capital cost. While this approach may be considered overly conservative, it does provide a very robust economic case. In spite of this conservativism, the estimated total cost of electricity is about 4-5 ¢/kWh, competitive with other nuclear and non-nuclear sources.

IRIS is currently in the pre-application review process with the U.S. Nuclear Regulatory Commission. This pre-application phase is intended to address long-lead items (such as testing) before the full-scale formal design certification process is started, thus allowing the latter to be completed expeditiously. Additionally, in its licensing IRIS will take maximum advantage of the successfully completed Design Certification of Westinghouse's AP600 and AP1000 for all those design features, analyses, and supporting tests which are similar in the three designs. However, further testing is necessary to address the new IRIS design features and components, including:

- Integral Reactor Coolant System
- Passive Safety Features specific to IRIS
- Reactor Vessel and Containment Interaction

The tests have been divided into three types according to their scope and primary purpose:

- Basic Engineering Development Tests.
- Component Separate Effects Tests.
- Integral Effects Tests.

The testing program started in 2006. A detailed design of testing facilities is underway. A large part of the safety related tests, and in particular the integral system test will be conducted in Italy, at SIET, at the same facilities where testing of the passive systems for AP600 was conducted in the 1990's. Further details are provided in the companion paper [19].

The safety-by-design™ philosophy in IRIS has lead to significant enhancement of its safety performance, as demonstrated in the reduction of the estimated core damage frequency (CDF) due to internal events to below 10^{-7} events per reactor-year. Currently, efforts are under way to implement the same approach to external events, including seismic. The compact integral design facilitates this task. The reactor building is cylindrical in shape, of moderate diameter, only about 30 meters above the ground, with the spherical containment fully contained within. This greatly increases security, providing robustness and resilience to external malevolent acts. Additionally, it improves the seismic response of the building, and if necessary it makes feasible the use of seismic isolators in regions with strong seismic activity. More details are provided in a companion paper [20].

5. Conclusions

IRIS is an advanced integral PWR of medium power (335 MWe), developed by an international team, led by Westinghouse, and with a strong participation of European organizations. It is well suited to satisfy the needs of both the smaller/emerging markets, as well as larger developed markets with limited growth rate or desire to optimize cash flow. Combined with its foreseen role within the U.S. DOE GNEP program, IRIS offers potential for a worldwide deployment. A major testing program was initiated at SIET, Italy, in 2006, to support submittal of the FDA/DC application in 2010, thus enabling deployment mid next decade.

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ACR-1000: Product Update

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1. Introduction

Atomic Energy of Canada Limited (AECL) has adapted the successful features of CANDU®* reactors to design Generation III+ Advanced CANDU Reactor®** (ACR®**) technology [1-3]. The ACR-1000®** nuclear power plant is an evolutionary product, based on proven, traditional CANDU reactor technology, coupled with thoroughly demonstrated innovative features to enhance economics, safety, operability and maintainability. This evolutionary strategy ensures that AECL's innovations are based on proven experience, and focuses the development programs on a select number of innovative features. The ACR-1000 basic design is complete, ready and regulatory review of formal licensing document is underway. Detailed pre-project design has started. The ACR program covers all activities required to achieve a first unit in-service date, and the program is being executed using full-scale project management principles.

The ACR-1000 has been chosen for generic design assessment in the UK. Additionally, there are active ACR-1000 new build initiatives in Canada: Ontario, Alberta and New Brunswick.

2. ACR-1000 Product Description

The standard ACR-1000 design is a 1200 MWe class nuclear power plant, which has evolved from AECL's existing successful product lines. The ACR-1000 applies the advanced CANDU technology developed in the ACR program. All innovative features of the ACR-1000 will be fully tested and proven before the first project. The design also makes extensive use of successful features of existing CANDU technology. By doing this, the ACR-1000 can be developed and applied in initial projects with a high degree of confidence. Additionally, it fully exploits the construction techniques that contributed to the impressive schedule accomplishments at Qinshan Phase III.

The ACR-1000 has the following major features:

- Twin-unit configuration with common control room building (can be delivered in single unit configuration)
- Compact, horizontal pressure tube core design following traditional CANDU overall configuration
- Enhanced inherent and passive safety with Moderator and Shield Tank heat sinks supplied by passive water makeup from Reserve Water Tank.
- Core consists of low-pressure, low-temperature calandria tank containing heavy-water moderator, within which fuel channels are located, each containing 12 standard-length, enriched, CANFLEX-ACR fuel bundles.
- Coolant is light water.
- Fuel channel consists of a Zirconium alloy pressure tube, surrounded by a Zircaloy calandria tube, and attached to coolant system feeder piping by individual end fittings.
- On-line core refueling is carried out via two computer-controlled fuelling machines
- Reactivity control and shutdown mechanisms are located in the low-pressure calandria tank with no possibility of accidental high-pressure ejection.
- Four-quadrant layout with four-way redundancy of safety support systems,
- Indirect thermal cycle (similar to PWR reactors), with the reactor coolant system transferring the heat from nuclear fission, through vertical shell-and-tube steam generators, to a conventional secondary turbine cycle.

** Advanced CANDU Reactor®, ACR® and ACR-1000® are trademarks of AECL.

SMART CANDU®*** is a registered trademark of AECL.

- Customer-driven improved features for operability and ease of maintenance.

While retaining proven CANDU features, innovations in the ACR-1000 design include:

- Use of light-water coolant in the CANDU coolant system, in conjunction with the continuing use of heavy water moderator in the calandria
- Design of a more compact core configuration to enable optimized reactor physics characteristics
- Use of low enriched fuel with higher burnup than the Natural Uranium (NU) fuel used in traditional CANDU reactors.
- Increased coolant system and turbine pressure to increase the overall thermal efficiency of the power plant.

The plant layout is designed to achieve the shortest practical construction schedule. This is achieved by simplifying the design, minimizing and localizing interfaces, parallel fabrication of module assemblies and civil construction, reducing construction congestion, improving access, providing flexible equipment installation sequences, and reducing material handling requirements.

Security and physical protection have also been taken into consideration in the development of the plant layout. Physical protection is provided through ample separation. The reactor building consists of a steel-lined, prestressed concrete containment structure and a reinforced concrete internal structure supported on a reinforced concrete base slab. The containment structure provides an environmental boundary, biological shielding, and a pressure boundary in the event of an accident. The building layout is arranged to provide separation by distance, elevation or barrier for safety related structures, systems and components. These features reduce the likelihood of common mode failure of safety systems due to malevolent acts such as an aircraft crash.

The ACR-1000 reactor core has the following characteristics:

- Compact size combined with on-power refuelling.
- Reduced heavy water requirements due to compact core size (lattice pitch of 240 mm versus 286 mm in current CANDU units) and the use of light water as the coolant.
- Moderate negative coolant-void reactivity.
- Simplified reactor control through negative feedback in reactor power.

A high form factor of 0.94 is achieved, along with increased core stability.

3. ACR-1000 Safety Features

The ACR-1000 design takes advantage of inherent and engineered safety characteristics, including distinctive features that arise from CANDU design principles. The core is designed for small-magnitude negative reactivity coefficients, which provide inherent protection against transients with inadvertent increase of reactor power. Additionally, two diverse and fully capable, fast-acting, independent shutdown systems are provided. Each system can shut down the reactor for the entire spectrum of design basis and anticipated events. Also, the separate control system shuts down the reactor for Anticipated Operational Occurrences.

Further defences - in-depth is derived from the inherent passive-safety design features of the CANDU fuel channel core [4]. The moderator heavy water surrounding the fuel channels water in the calandria is itself an additional, diverse active/ passive heat sink. The calandria is filled with heavy water to a level well above the top of the calandria shell. This heavy water acts as both a moderator and reflector for the reactor, as well as an assured heat removal option. The moderator system is a low-pressure and low-temperature system that is fully independent of the heat transport system. Moderator heat exchangers remove the heat generated in the moderator during reactor operation and shutdown. Passive make up to the moderator is provided, and long term cooling assured by a Reserve Water Tank

Core retention within the vessel includes both retention within fuel channels, and retention within the calandria vessel. The moderator heavy water in the ACR-1000 calandria vessel, as in any other CANDU-type reactor, provides ample heat removal capacity in severe accidents. The ACR-1000 calandria vessel design permits for passive rejection of decay heat from the moderator to the shield water. Also, the calandria vessel will be designed for debris retention. Core damage termination is achieved by flooding of the core components with water and keeping them flooded thereafter. Successful termination can be achieved in the fuel channels, calandria vessel or calandria vault by water supply by the Long Term Cooling pumps and by gravity feed from the Reserve Water System.

The ACR-1000 containment is required to withstand external events such as earthquakes, tornados, floods and aircraft crashes. Containment integrity maintenance is achieved through control of containment pressure, flammable gas control, and control/prevention of the core-concrete interaction. The containment system includes the steel-lined, pre-stressed concrete reactor building containment structure, access airlocks, building air coolers for pressure reduction, and a containment isolation system, consisting of valves in certain process lines and ventilation ducts that penetrate the containment structure.

4. ACR-1000 Operability and Maintainability

The design basis lifetime capacity factor for ACR-1000 is 90% over the operating life of 60 years. The design basis year-to-year design capacity factor is 93%. The engineering components of individual systems and components uses plant-wide models such as PSA models, and operations expert feedback to exceed these targets. Customer feedback has resulted in many detailed operational and maintenance improvements being incorporated into the design to meet the performance targets. Additionally, use of CANDU operating experience facilitated by the information network provided by the CANDU Owners Group (COG) will further improve the performance of the plant.

On-power maintenance and testing are optimized, reducing the frequency of outages to once every three years, with planned maintenance outages not exceeding 25 days. The unplanned outage rate target, is less than seven days/year through design optimization, and through the use of System-Based Maintenance strategy. The plant layout is optimized to facilitate online maintenance and inspection, to provide access for equipment exchange, and to provide effective common services for the two-unit plant design resulting in reduced maintenance costs. For Plant Life Management purposes, there is provision of space and services to support a rapid, mid-life full-scale fuel channel and steam generators replacement program. Maintenance activities are enhanced to maximize component life and minimize component replacement time, thereby minimizing radiation exposure, replacement costs, and the number of operating and maintenance personnel required.

Application of existing CANDU computer control knowledge and experience, enhanced by state-of-the-art information system technology, has produced advanced plant control and monitoring systems that enable the plant to operate at higher capacity factors with a reduced operations staff. SMART CANDU®[***] modules provide on-line health monitoring and diagnostics for plant chemistry, predict future performance of components, determine maintenance requirements and optimal operating conditions and ensure optimal margins and maximum power output.

5. ACR-1000 Design Status

5.1 General

The CANDU 6, CANDU 9 and ACR-700 programs produced the foundation for the ACR-1000. AECL selected the ACR-1000 as its new reference design to meet market requirements. The ACR-1000 program focus is to plan and execute work based on risk analysis, assessment and mitigation, to ensure licensability and address customer input, and to achieve an in-service date of 2016. The program plan is project based, using a comprehensive 10,000-activity schedule and is intended to ensure that all required documentation is available to support the Environmental Assessment and

Site Preparation and Construction License applications. The program is designed to have all design documentation completed prior to start of construction.

The ACR conceptual design has been completed and the ACR-1000 is now a project under the management of AECL's Commercial Operations group. A revised quality assurance manual covering the ACR-1000 and Enhanced CANDU 6 products has been issued. The framework for the overall project execution plan has been developed and identifies the key project execution elements.

The technology issues have been successfully resolved and the licensing basis has been established and all elements of the basic engineering program are in progress. Project risk management processes and procedures have been put in place. The 478 work packages required for the Preliminary Safety Case Package (PSCP) submission (the reference submission for use in generic pre-project reviews by regulators) and for input to the generic Preliminary Safety Analysis Report (PSAR) have been developed to support scope, cost and schedule management requirements. The Level-3 production schedule—covering the basic engineering program together with the remaining R&D work and licensing activities—has been issued. Approximately 400 full-time-equivalent staff work on the project on a day-to-day basis.

5.2 ACR-1000 Design Evolution

The design of the ACR-1000's systems, structures and components is based on the successful CANDU 6 and Darlington nuclear steam plants (NSPs). Minimal manufacturing and supply changes are anticipated due to the similarities of major NSP equipment and components for the ACR-1000 and CANDU 6. Major equipment and components have been proven through many years of continuous operation of 10 CANDU 6 plants. A proven licensing and safety basis builds on 40 years of CANDU licensing experience in Canada and around the world. The Balance of Plant (BOP), comprising 40% of total plant equipment, is a scale-up of the proven CANDU 6 BOP.

A number of innovations were accepted by the Canadian Nuclear Safety Commission (CNSC) in the pre-project licensability review of the CANDU 9 in 1997-98, and have been adopted in ACR-1000:

- large, steel-lined containment
- improved circulation in the moderator
- reserve water tank for accident coolant make-up
- high-pressure safety feedwater system
- distributed control system/plant display system and modern control centre incorporating human factors considerations

The ACR design includes the technologies enabling compact reactor core with light water coolant and low-enriched uranium fuel developed and reviewed by regulators at earlier stages in the ACR program. Other ACR innovations include:

- thicker pressure tubes and thicker and larger calandria tubes
- mechanical zone control replacing liquid zone controllers and adjuster rods
- stainless steel feeders and headers
- long-term cooling system to perform long-term emergency core cooling (ECC) and maintenance cooling

Additional design enhancements were incorporated into the ACR-1000 design to mitigate technology issues that represent perceived risks based on project risk management evaluations. Other changes were made to meet new Canadian regulatory guidelines and regulations including the Design Requirements Documents (DRD) for new plants. Other design changes were made to improve operational performance based on customer feedback:

- simplified CANFLEX-ACR fuel bundle design:
    - 42 similarly-sized elements with 2.4% enrichment
    - larger centre element with burnable neutron absorber but no uranium
    - uniform enrichment

- simplified and more reliable emergency coolant injection
- new system classification based on safety function categories
- four-train emergency feedwater system as emergency heat removal system
- additional reactor trip to meet no-dryout requirement for end-of-life conditions

5.3 Design Readiness

ACR-1000 design and analysis work is well underway and will meet the owner's needs for EA, site and construction licenses, towards a first in-service date of 2016. The next milestone—completion of the Preliminary Safety Case Package by 2008 May—**will represent a standard licensing package appropriate for stand-alone regulatory review during any individual pre-project phase and analysis completion for ACR-1000.** The "design freeze" in 2007 March was a key step in integrating the design and getting ready for formal safety evaluation of the ACR-1000 product. The reference plant design documents from CANDU 6, CANDU 9 and ACR-700 documents will be updated for use in ACR-1000, as part of the project-engineering phase—with the objective of completing all design documentation prior to the start of construction.

6. SUMMARY

The ACR-1000 uses well-established, fundamental, CANDU design elements: core design with horizontal pressure tubes; simple efficient fuel bundle design; on-power refuelling and a separate low-pressure, low-temperature heavy-water moderator providing an inherent emergency heat sink. It includes adaptations for light-water coolant and low-enriched uranium fuel, and offers a compact core configuration and higher steam pressure for greater thermodynamic efficiency. The ACR-1000 links design with licensing, emphasizing operability and maintainability from the viewpoint of the customer—the utility operator.

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WHY THE WESTINGHOUSE ADVANCED, PASSIVE PRESSURIZED WATER REACTOR, AP1000?

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ABSTRACT

What does the AP1000 do that is an improvement over the earlier models? For one thing, it responds to the desire expressed by nuclear utilities for a simpler plant. It also responds to utility needs for a nuclear plant that can compete more favorably on capital investment with fossil power plants, and is even safer than current models. AP1000 meets these goals by preserving the essentials of the proven, robust, and reliable virtues of the power generating features of earlier Westinghouse plants while incorporating simpler but highly reliable passive cooling safety systems for the core and the containment. Combined with the use of PRA to guide the design, the AP1000 has a Core Damage Frequency (CDF) of 5.1x10-7, as certified by the US NRC. Compare this to currently operating plants with their active (pump-driven) safety systems that have a CDF typically at 5x10-5.

To address capital cost competition, AP1000 has a highly developed construction plan to minimize the time and cost of construction. It is designed from the outset for modular and “open top” construction techniques. The whole process of construction and construction planning is further abetted by the lower appetite of AP1000 for construction commodities afforded by the passive design’s more compact dimensions and more concentrated areas requiring less Seismic Category 1 construction. With less equipment required by the design, AP1000 represents a focused effort towards minimizing the traditionally high cost of nuclear plant construction.

Where do things stand today? Under its new licensing approach, the US NRC has reviewed and certified the AP1000 design. That makes it a licensed plant design that can be referenced in combined Construction Operating License (COL) applications. The AP1000’s debut has been received favorably. AP1000 has so far been identified by five US utilities for ten units as the plant design in such applications. It has also been selected for four units to be constructed in China along with technology transfer to support additional AP1000s to be built there under license. And on May 15 the European Utility Requirements (EUR) organization certified that the AP1000 pressurized water reactor has successfully passed all the steps of analysis for compliance with European Utility Requirements, confirming that the AP1000 can be successfully deployed in Europe.

1. Introduction

It is worth looking back for a moment to examine the circumstances extant at the time that development of AP1000 got underway, and how a product with such a long development time emerges seemingly at the moment that interest in the nuclear option has taken hold world-wide. What we can say is that design work began at Westinghouse on the AP600, predecessor to the AP1000, in 1989. This was hardly an auspicious time to begin working on the next generation of nuclear plants. The market for nuclear power plants had for the most part retreated to Asia, where the need for new nuclear plants was still strong and supported by the electricity demands of vigorously growing economies. Indeed, I spent many years of my career during the 1990s with Westinghouse in the Republic of Korea. Despite the inauspicious circumstances outside of Asia at that time, there were, nevertheless, other significant developments occurring that would give us greater confidence in proceeding with AP1000.

First, the US NRC developed, and in 1989 enacted, a new plant licensing process, Title 10 CFR (Code of Federal Regulations) Part 52. The new process sprang from the long history of licensing all 104 plants now operating in the US under 10 CFR Part 50. Part 52, would essentially re-shuffle the licensing sequence by front-loading the approval of the site and the plant design prior to issuing a license to build and operate. This has the effect of settling design and site issues prior to making major investments. It is quite unlikely that any US utility would be considering a nuclear plant today if the Part 50 regulations still prevailed.

In this same period, U.S. utilities, showing foresight, joined together to develop -- in cooperation with the U.S. Department of Energy, the U.S. NRC, and nuclear power plant designers -- a set of design requirements for the next generation of nuclear power plants. The resulting multi-volume document, the "EPRI Utility Requirements Document (URD)", is a design specification for new nuclear power plants, incorporating the lessons learned in construction, licensing, operation, and maintenance of the existing fleet of operating nuclear power plants. A similar activity commenced in Europe producing the European Utility Requirements (EUR). AP1000, and its predecessor, AP600, were developed virtually in parallel with the URD. It was a valuable road map. Consequently, AP1000 embodies the URD design specifications for an advanced, passive plant design, and it has been certified by the EUR group confirming its suitability to being deployed in Europe.

And so we find ourselves today with a very different set of boundary conditions from 1989.

2. What about the design?

Over the past decades, what we have seen is steady improvement in nuclear plant performance. Capacity factors are now commonly greater than 90%. Electricity production costs of nuclear plants are typically comparable to coal-fired plants and often are the cheapest power on the grid next to hydroelectric power. All of this has been accomplished with exemplary levels of safety.

In essence, utilities have been perfecting their mastery of plant operation. Now they have become *virtuosi.* They also know what to look for in a new instrument. So it falls to plant designers to provide what utilities need for the next generation of plants.

Both the URD and the EUR have the general approach of preserving the virtues of the operating plants when it comes to the power producing system - the primary systems in particular - which have proved themselves so well. But there is also a requirement for a simpler plant this time, and one that is safer and costs less to construct. Both EUR and URD anticipate and address specifically the advantages of a passive plant for both cheaper construction and to reduce the reliance on operator action in case of an accident. In fact, the expectation for a passive plant is to achieve and maintain safe shutdown in case of an accident for 72 hours without operator action. This is substantially different than the 30 minute period for operator action specified for an "evolutionary," active system plant.

The URD also expects that for the new generation a complete plant design will be offered to utilities, encompassing the entire plant up to its connection to the grid.

3. How does AP1000 go about meeting these requirements?

Retain the virtues

AP1000 is an 1117 MWe plant. The power producing primary system is a familiar one based on proven and reliable Westinghouse PWR features, but with evolutionary improvements to be expected with the benefit of decades of operating experience, development of improved materials and better manufacturing techniques. Replacing Alloy 600 steam generator tubing with Alloy 690 tubing and the use of low cobalt-content alloys to reduce activation are some examples. This, of course, is a direct outgrowth of the steam generator replacements on the operating plants. The AP1000 reactor vessel is ring-forged, eliminating longitudinal welds. And there are no circumferential welds in the high flux core region. These features combined with improved materials allow for a *60 year vessel life.* The fuel design is closely based on the XL Robust Fuel Assembly design that has been operating in Doel, Tihange, and South Texas.

One of the improvements found in the AP1000 primary system design is the use of sealless, reactor coolant pumps. By eliminating the need for the complex shaft seal, a source of potential primary system leakage is eliminated. The sealless RCP requires no oil lubrication system, and is designed to be maintenance-free. In fact, sealless motor pumps were used in the first generation of Westinghouse PWRs but, as the plants became larger with the second generation designs, they out-grew the capacity of that type of pump available at the time. Since then sealless pump sizes have increased, enabling their application to power reactors once again.

Gain the Advantages of Passive Safety Systems

Here is where we take a different path with AP1000, one that re-casts the safety-related cooling systems in light of many decades of experience with the old systems. AP1000 features passive safety systems for emergency core and containment cooling. It is the essential means of simplifying the PWR while at the same time increasing the level of safety. In designing the systems, we had the benefit of using highly developed Probabilistic Risk Assessment methods. This new approach of using PRA as an integral element of the design had the effect of driving the Core Damage Frequency (CDF) to unprecedented low probability levels: $5.1x10^{-7}$ for power and shutdown conditions combined. A CDF for the operating plants is typically $5x10^{-5}$. The URD and EUR goal for new plants is that they be less than a CDF of $1x10^{-5}$. The exceptionally low AP1000 CDF results from, among other things:

1) Use of non-safety related active systems, such as startup feedwater, for first response to transients, backed up by the passive safety systems
2) An effective design deploying not only redundant systems but systems incorporating diverse equipment, such as valve types, where it has the most beneficial effect
3) Eliminating the extra safety related equipment needed to generate emergency ac power.

The AP1000 takes a direct and simple approach for the severe accident scenario. Our design avoids ex-vessel molten core interactions altogether. The AP1000 design allows for cooling of the vessel exterior with water from the large In Containment Refueling Water Storage Tank fed by gravity to the reactor cavity. It floods the cavity and flows up along the exterior of the reactor vessel, removing heat, and then is ultimately re-circulated through a steam condensation cycle within containment. Pressure build up within the vessel is relieved by the automatic de-pressurization system. The cooling is sufficient to maintain the vessel integrity, thereby retaining the molten core inside. The steel containment, meanwhile, is cooled by natural air convection channeled by the reinforced concrete shield building featuring a system of air intake and exhaust vents. The ambient air is the AP1000's ultimate heat sink in case of accidents. The air cooling can be augmented by evaporation cooling from a water storage reservoir poised on the top of the shield building.

The simplicity in AP1000 derives from passive systems not needing as much equipment to carry out their mission. In the AP1000 the HVAC, service water and circulating water systems, among others, are re-classified as non-safety related and serve only non-safety related equipment. The elimination of the usual network of safety-related pumps and supporting systems results in not needing safety-related emergency ac power. All of this greatly reduces the volume of Category 1 seismic buildings and allow most safety equipment to be concentrated within the containment. It also results in 40 to 50% fewer containment penetrations for AP1000 compared to a conventional plant.

All of the forgoing achieve new levels of safety *and* drive down plant cost. We estimate AP1000 compared to a conventional plant to have:

- 50% fewer safety class valves
- 80% less safety class piping length
- 35% fewer pumps of all types
- 70% less cable.

The net effect of AP1000's reduced requirements for equipment and the building space needed to house the equipment is a very compact footprint. To gauge the effect on construction and size of buildings, following here is a comparison of some construction quantities for AP1000 compared to Sizewell B, a Westinghouse PWR commissioned in the UK in 1995:

	Concrete, m3	Rebar, metric tons	Power, MWe
Sizewell B:	520,000	65,000	1188
AP1000:	<100,000	<12,000	1117

Such reduced quantities of building materials will also translate proportionally into less costly decommissioning as well.

The URD defines a passive Advanced Light Water Reactor as: "Simpler, smaller and much improved LWRs employing primarily passive systems for essential safety functions." That definition accurately describes the AP1000.

4. Get it licensed

Our application for design certification of AP600, which first embodied these passive design features, was approved and the design certified by the US NRC in 1999 under 10 CFR Part 52. AP1000 was certified by the NRC at the end of 2005. That means that the AP1000 design has accomplished what NRC considers its most resource-consuming licensing interaction under 10 CFR 52. AP1000 can be referenced as a pre-approved plant in any US utility's application for a combined, construction and operating license (COL) at a specific site.

5. Reduce the Cost

AP1000 is designed for modular construction and is a standardized plant design. Only necessary variations for site-specific interfaces are contemplated and even for that the basic design already accommodates a wide array of site attributes such as 0.3g seismic acceleration, for example, and 50 or 60Hz electrical systems. Both standardization and modular construction are cost effective improvements over traditional methods. But the promise of simplification and more cost effective means of construction nevertheless have to result in competitive capital cost in relation to other types of power plants.

In the AP1000 we have a most cost-efficient design to propose to utilities. Nevertheless, costing remains a complicated matter. Costs for commodities are not standing still for any type of power plant project. The resurgence of demand for nuclear plants will also cause specialty suppliers to adjust their capacities upwards from the levels they have gotten used to over the past decades of a retrenched market. That adjustment is not instantaneous. For the earliest plants, we will need to be vigilant to avoid production bottle-necks. Another characteristic of large power plant costs is that there will still be a significant contribution to cost that is dependent on local labor costs, at least for construction craft labor. Since this varies widely, even within the US, there is never a single cost for a plant – even a standardized plant.

6. Conclusion

We are witnessing a world-wide need to re-orient our energy generating base for more secure energy supplies and for reduced greenhouse gas emissions. The resolution of these issues will ultimately rely on technology, whether for new, clean coal designs, renewable energy, or nuclear power plants. New nuclear power plants can provide a proven, cost-competitive, non-greenhouse gas emitting option that, as indicated by the World Energy Council's report of February 1, 2007 and the UN's Intergovernmental Panel on Climate Change report of May 4, needs to play an important role. To use the terminology often used these days, nuclear power is one of the "wedges" needed to make the change. The latest Westinghouse PWR, the AP1000, is a significantly improved version of the reliable PWR, and now available to perform that role.

Session 17.1.2:

Safety of advanced reactors

AP1000 PASSIVE DESIGN FOR IN-VESSEL RETENTION

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ABSTRACT

AP1000 is a Westinghouse two-loop 1100 MWe advanced pressurized water reactor that uses passive safety features to enhance plant safety and provide improvements in plant simplification, reliability, investment protection and cost. One of the passive safety features is In-Vessel Retention which passively provides sufficient external cooling of the reactor vessel to retain a molten core inside the vessel in the unlikely event of a severe accident. This concept was proven by a series of tests reviewed and accepted by the United States Department of Energy and Nuclear Regulatory Commission and offers numerous advantages over other severe accident core management designs.

The testing that proved the In-Vessel Retention Concept also identified a series of key features and functions for the Reactor Vessel Insulation System (RVIS), making it different from any other reactor vessel insulation. This paper presents the key functional and design requirements for the RVIS and the RVIS design configuration.

1. Introduction

Passive features are defined as those that do not rely on human intervention or devices such as motors or pumps to perform their function but instead use natural phenomena like gravity. One of the passive safety features of the AP1000 is named "In-Vessel Retention". In-Vessel Retention passively provides sufficient external cooling of the reactor vessel to retain a core that has been relocated to the bottom head of the reactor vessel in the unlikely event of this severe accident. The In-Vessel Retention concept was proven by a series of tests and analyses and offers numerous and obvious advantages over other severe accident core management designs. The testing and analyses for AP1000 built on testing and analyses that previously demonstrated the In-Vessel Retention concept for the Westinghouse AP600 reactor vessel.

The AP1000 testing and analyses were previously presented in a number of technical papers including:

- "In Vessel Retention of Molten Core Debris in the Westinghouse AP1000 Advanced Passive PWR", James H. Scobel, et al, ICAPP 2003
- "Westinghouse AP1000 PRA Maturity, D. McLaughlin, et al, ICAPP 2005.

These papers may be reviewed for more information on the testing and analyses. This paper focuses on the mechanical design and configuration that implements the testing and analyses results.

Like the reactor vessel insulation in other nuclear power plants, the AP1000 reactor vessel insulation insulates the reactor vessel to minimize heat loss to the cooling air in the reactor cavity. As in other nuclear power plants, the AP1000 reactor vessel insulation protects temperature sensitive structures and components from exceeding their maximum allowable temperatures.

However, to provide for In-Vessel Retention the AP1000 reactor vessel insulation must be different. The AP1000 In-Vessel Retention concept uses water that has flooded the reactor cavity as the cooling medium for the reactor vessel. The flood water must freely contact the external surface of the reactor vessel for this cooling to occur. Conventional reactor vessel insulation forms a barrier between the flood water and the reactor vessel.

Not only must the Reactor Vessel Insulation System (RVIS) allow flood water to reach the reactor vessel, the reactor vessel insulation must provide certain additional features and functions which were identified from the testing and analyses. These features and functions make the AP1000 RVIS different from the reactor vessel insulation in any other nuclear power plant, including AP600.

The following sections summarize the key functional and interface requirements for the RVIS for both normal and severe accident conditions and show the design configuration that meets those requirements.

2. Key Functional and Design Requirements of the RVIS

2.1. Normal Conditions

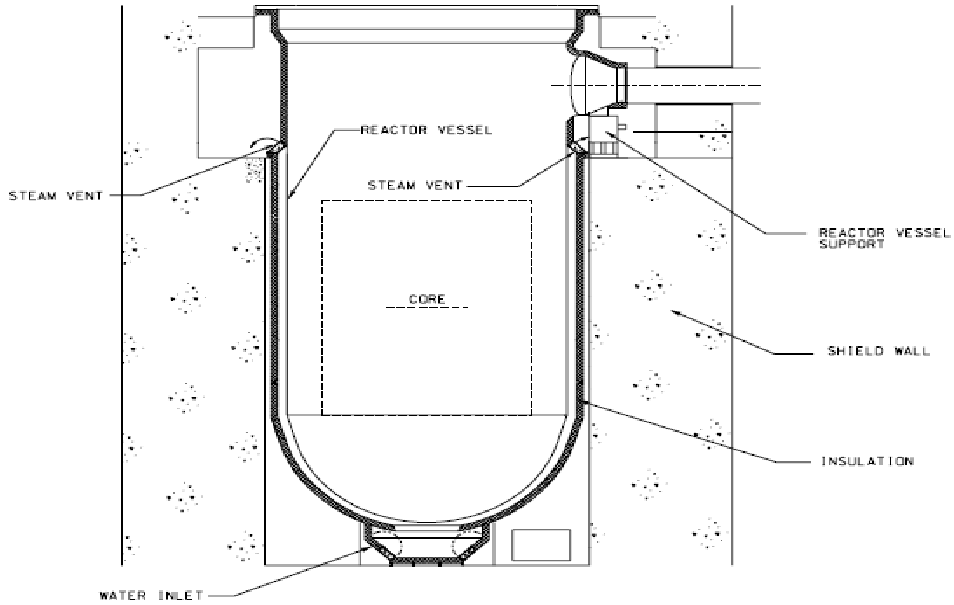

Fig 1. Reactor Vessel Insulation System General Arrangement. All details are not shown.

Figure 1 provides a representative pictorial of the RVIS. The RVIS includes the insulation below the top flange on the reactor vessel. As in other nuclear power plants, the RVIS is located between the reactor vessel and the reactor cavity walls. Space is maintained between the RVIS and the reactor cavity walls and floor for cooling air flow. During plant design basis conditions, the RVIS limits heat loss from the reactor vessel.   The RVIS and reactor cavity cooling air limit the temperatures of structures and components in the reactor cavity to within allowable limits during plant design conditions. The structures and components of concern are the concrete, the neutron shielding, and the ex-vessel neutron flux monitors, also called “ex-core detectors”.

Reactor cavity cooling air enters the bottom of the reactor cavity (floor elevation 71’-6”), flows under and around the insulation on the bottom head of the reactor vessel, up the outside of the reactor vessel sidewall insulation, through the reactor vessel supports into the nozzle gallery (floor elevation 98’-0”).

2.2 Severe Accident Conditions

Testing and analyses have shown that the reactor vessel can retain a molten core in the bottom head of the reactor vessel if the external surface of the vessel is sufficiently cooled. Testing and analyses have also shown that cooling will be sufficient if:

- An annulus with certain dimensions and characteristics is maintained between the reactor vessel and the reactor vessel insulation along the bottom head and up the sidewall of the reactor vessel

- Water can freely and continuously flow into the annulus at the center of the reactor vessel bottom head
- Steam and water can freely vent at the top of the annulus.

The scenario during a severe accident is that containment flood water flows into the reactor cavity. While the flood water rises to a level above the reactor vessel, it freely flows into the annulus between the reactor vessel and the reactor vessel insulation. The water in the annulus takes heat away from the reactor vessel and forms steam bubbles. The steam bubbles rise in the annulus pushing water ahead of them. The rising steam-water mixture flows out of the top of the annulus, returning to the containment flood water.

The RVIS provides the outside wall of the annulus required by the first bullet above, and must therefore remain intact under the loadings that occur during the accident. Because of the water outside the insulation and the steam void fraction inside the annulus, there is a differential pressure across the insulation. Additionally the formation and collapse of bubbles in the annulus creates a pressure oscillation. Testing and analyses have quantified these numbers to be 12.95 feet (3.95 meters) of differential water pressure on the outside of the RVIS with a pressure oscillation of +/-1.64 feet (0.5 meters) of water. These are loads that the RVIS has been designed to withstand.

The second and third bullets above are counter to the requirements during normal conditions when it is important to inhibit air flow into and out of the annulus in order to minimize heat loss from the reactor vessel, minimize the heat load on the containment cooling system, and maintain components and structures within their allowable temperatures. Unique RVIS features are required that inhibit air exchange during normal operations but permit the bottom and top of the annulus to passively open during a severe accident. A water inlet assembly and steam vents were designed to provide these diverse features. These are described in Section 3.

3. RVIS Design Configuration

The RVIS is primarily constructed of ASTM Type 304 stainless steel metal reflective insulation (MRI) 4.5 inches (11.43 cm) thick. MRI generally consists of inside and outside sheet metal enclosures and multiple layers of metal foils inside. MRI insulates by minimizing internal conduction paths between the inside and outside enclosures, minimizing internal convection currents, and minimizing internal heat transfer due to radiation. MRI is used extensively in nuclear applications.

The RVIS above the nozzle gallery floor is similar in design to MRI systems used on other pressurized water reactor (PWR) vessels. The MRI is manufactured in panels which can be handled by one or two workers. The MRI panels are attached together and allow a slight stand-off from the reactor vessel to allow for manufacturing tolerances and reactor vessel expansion due to the temperature and pressure increases. The MRI on each nozzle is a clam-shell design. The clam-shell design allows the MRI to be removed, for example for in-service inspection. The clam-shell sections fasten together along longitudinal seams and to the MRI on the sidewall of the reactor vessel. The MRI clam-shells are supported by the MRI panels on the reactor vessel sidewall and by the nozzle.

The RVIS below the nozzle gallery floor provides the annulus for In-Vessel Retention and must therefore be designed to withstand higher loads. Examples of these higher loads include the large and varying differential pressure discussed above during a severe accident and differential pressure due to containment pressurization, such as from a hypothetical pipe break in containment. To achieve the needed strength, the MRI panels are attached to a supporting structure which is fastened to the reactor cavity walls. In addition the MRI panels in this region have thicker inside and outside enclosures as well as additional stiffeners inside.

The structure that supports the bottom head attaches to the reactor cavity sidewall and to legs that extend downward and attach to the reactor cavity floor. The MRI panels provide a hemispherical shape generally conforming to the shape of the bottom head of the reactor vessel and providing the annulus prescribed by the testing.

As shown in Figure 2, there is an opening in the bottom head insulation at the axial centerline of the reactor vessel. The inlet assembly is centered on and fits around this opening. The inlet assembly extends downward from the MRI on the bottom head of the reactor vessel to the reactor cavity floor, attaching to each. The inlet assembly is constructed of MRI and includes four sides and a bottom. Each side has a vertical section at the top and an inwardly sloping side below it; similar to a hopper. A space is maintained under the bottom to assure acceptable concrete temperatures are maintained in this area.

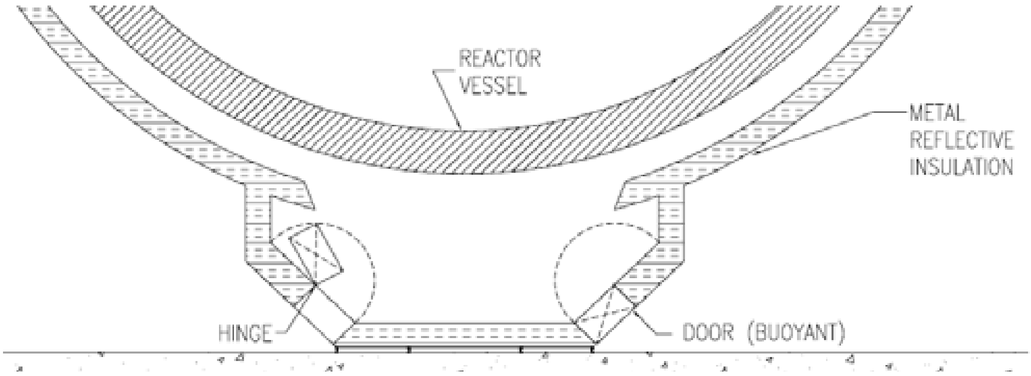

Fig 2. Water Inlet Assembly. Far door and other details are not shown.

Each of the four sloping sides of the inlet assembly contains a door. The door is hinged on the top inside edge and four open doors provide more than 6 ft^2 (0.56 m^2) of flow area for water to enter the inlet assembly. This is the minimum flow area established by testing and analysis. The weight of the door keeps it in the closed position, pressed against a continuous circumferential stop which inhibits the free exchange of air between the inside and outside of the inlet assembly. Each door is a hollow stainless steel enclosure filled with a buoyant hydrophobic material. The buoyant material causes the doors to rise as the reactor cavity floods with water. There is sufficient room inside the inlet assembly for the doors not to contact each other during opening. Fully open, the doors are in a vertical position with their center of gravity over the hinge. In this position the water flow past the doors will tend to push the doors further open. If the doors ever lose their buoyancy during the severe accident, gravity and the water flow work together to cause the doors fall outward instead of inward. The inlet assembly has sufficient space for the doors to fall outward beyond the vertical position. Small vent holes prevent pressure differentials inside the doors during reactor heat-up and cool-down.

Door freedom can be periodically checked by pushing them inward and checking their swing action. Each door is mounted in a frame and the door-and-frame assembly is removable as a unit. Removing and replacing the door-and-frame assembly as a unit eliminates fit-up concerns if the door only were to be replaced. Replacement is not anticipated over the design life of the plant.

As shown in Figures 3 and 4, at the top of the annulus at the floor of the nozzle gallery are steam vents. The steam vents are comprised of a series of narrow straight doors in the nozzle gallery that extend continuously around the reactor vessel from the neutron shield to the RVIS panels. The steam vents fit under each of the reactor vessel nozzles and their open doors provide a total flow area of at least 12 ft^2 (1.11 m^2). This flow area is a design requirement from the testing and analyses.

The steam vent doors are constructed of MRI panels. The doors are hinged on their outside bottom edge and slope inward toward the reactor vessel. Their top edge continuously contacts the RVIS panels in the nozzle gallery and their sides have flow barriers to inhibit air flow between the annulus and the nozzle gallery. The momentum of the steam-water mixture rising in the annulus opens the doors during a severe accident. Once open, the doors stay open due to gravity. If necessary, these doors could be filled with a buoyant hydrophobic material like the inlet assembly doors to provide an increased force for opening once the flood water level reaches this point, but this is not required. Like other MRI panels, the doors themselves are vented to prevent internal/external pressure differentials during heat up and cool down.

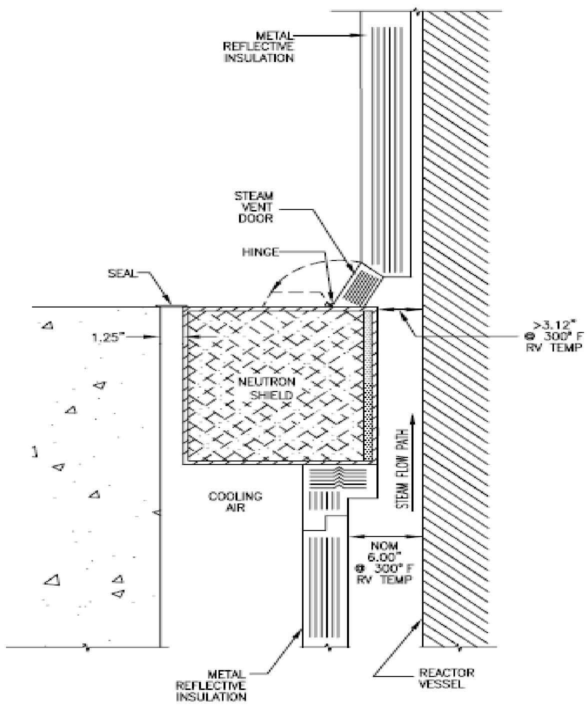

Fig 3. Steam Vent and Neutron Shield

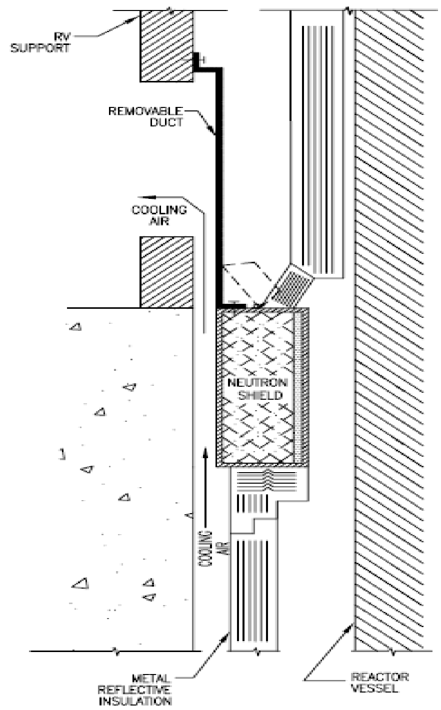

Fig 4. Reactor Vessel Support Cooling Ducts

The steam vent doors are accessible during routine periodic plant shut-downs. Door freedom can be checked by swinging them outward to check their swing action. In addition, the doors and steam vents are removable. Removal of the steam vents or doors is not anticipated to be required over the design life of the plant.

Also as shown in Figures 3 and 4, between the steam vents and the insulation below them is the neutron shield. The neutron shield forms part of the flow path for both the cooling air flowing upward from the reactor cavity during normal conditions and the steam-water mixture flowing upward to the steam vents during a severe accident. The neutron shielding material is enclosed in an insulated stainless steel enclosure to maintain the shielding material to less than 400^0 F (204.44^0) C. Cooling air flows upward over the lower RVIS and then flows over the underside and outboard surface of the neutron shield which also protects the concrete. The neutron shield has an octagonal outside shape and a cylindrical inside shape to interface properly with the octagonal reactor cavity and the cylindrical reactor vessel. This is the reason for the different neutron shield widths in Figures 3 and 4.

As shown in Figure 4, removable ducts mounted between the neutron shield and the reactor vessel support channel the reactor cavity cooling air to each reactor vessel support. Each reactor vessel support has an opening that allows the cooling air to pass through and baffles to make cooling more efficient. These features maintain the concrete temperature under each support below the limit.

Thermal and structural analyses have confirmed that the design meets its design requirements and interfaces for normal and severe accident conditions.

OPTIMIZING THE ACR-1000 CORE FOR SAFETY, ECONOMICS AND RELIABILITY

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ABSTRACT

AECL has adapted the successful features of CANDU®* reactors to establish the Generation III+ Advanced CANDU Reactor**™ (ACR™) technology. The ACR-1000™ nuclear power plant is an evolutionary product, solidly based on CANDU reactor technology, incorporating thoroughly demonstrated innovative features to enhance safety, operability, economics and maintainability.

The ACR-1000 core design is based on well-established, fundamental, CANDU design elements: fuel enclosed in horizontal pressure tubes; a simple, efficient fuel bundle design; on-power fuelling; a separate, low-pressure, low-temperature heavy-water moderator providing an inherent emergency heat sink.

This paper summarizes design optimisation features of the ACR-1000 Reactor Core design. The ACR-1000 reference core is described, as well as the main core features that distinguish the ACR-1000 from its predecessor CANDU reactors: the reactivity control devices and shutdown systems, including rods for maintaining the reactor in a guaranteed shutdown state; adaptations for a light-water cooled, low-enriched uranium fuel, leading to more benign neutronic characteristics; a more compact core configuration and higher steam pressure for greater thermodynamic efficiency.

Introduction

This paper focuses on enhancements in the ACR-1000 design with respect to existing CANDU reactors[1]. Table 1 shows a comparison of relevant core parameters between a generic CANDU 6 reactor as built in Qinshan, the larger CANDU reactors operating at Darlington and Bruce, and the ACR-1000.

The table shows that the ACR-1000 delivers a higher power in a core that is smaller than the CANDU-6 core. This is achieved by the combined effects of reduced lattice pitch, light water coolant and enriched uranium.

Parameter	**CANDU-6**	**Darlington**	**ACR-1000**
Heat to steam generators (MW)	2064	2657	3208
Gross/net electric output (MWe)	728/666	935/881	1165/1085
Number of channels	380	480	520
Core diameter (m)	7.6	8.5	7.44
Lattice pitch (cm)	28.6	28.6	24.0
Moderator D_2O volume (m^3)	265	312	235
Heat Transport System D_2O volume (m^3)	192	280	0
Total D_2O volume (m^3)	466	602	240
Fuel (wt% U^{235}/U)	0.71	0.71	up to 2.5
Number of elements per bundle	37	37	43
Total bundle weight (kg)	24.1	24.1	21.5
Reference discharge burnup (MWd/Mg(U))	7500	7800	20000
Outlet header operating pressure (MPa)	9.9	9.9	11.1
Outlet header operating temperature (°C)	310	310	319

* CANDU® is a registered trademark of Atomic Energy of Canada Limited (AECL).
** Advanced CANDU Reactor™, ACR™ and ACR-1000™ are trademarks of AECL.

Parameter	CANDU-6	Darlington	ACR-1000
Steam temperature (°C)	260	265	276

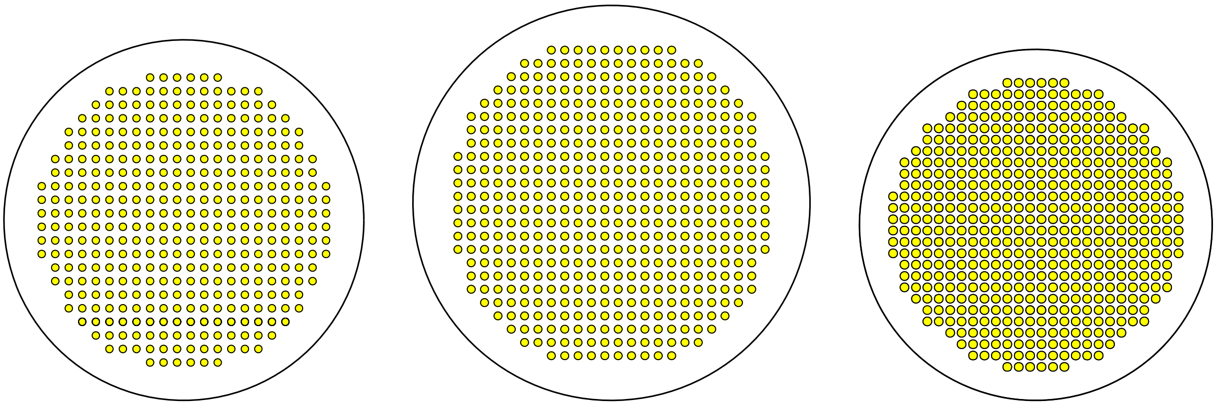
Table 1: Comparison of relevant CANDU reactor parameters.

At the same time, the similarity between the ACR-1000 and existing CANDU reactors is evident from the table, underlining the evolutionary nature of the ACR-1000. The following sections will describe a number of key enhancements of the ACR-1000 with respect to the traditional design.

2. Reactor Core

As in existing CANDU reactors, the reactor core of the ACR-1000 consists of a calandria vessel traversed by horizontal fuel channels.

Figure 1 shows comparisons of the core layouts of CANDU-type reactors in Table 1. It clearly demonstrates the compact design of the ACR-1000.



CANDU-6 Darlington ACR-1000

Figure 1: Comparison between the calandria dimensions of CANDU reactors and the ACR-1000

With respect to the traditional CANDU design, the ACR-1000 shows a large reduction of the volume of D_2O, mostly caused by the elimination of D_2O as a coolant, but also by the reduction of the moderator volume. This leads to a significant reduction in capital cost for the ACR-1000.

3. Basic Lattice Cell

3.1 Pressure and Calandria Tubes

Figure 2 shows the basic ACR-1000 lattice cell, including the arrangements of the fuel bundle, the H_2O coolant, pressure tube, calandria tube, and the D_2O moderator. The pressure tube of the ACR-1000 has been made slightly thicker than that of the existing CANDU. This is to reduce the creep and sag of the tube over its lifetime. Creep, or diametral expansion, of the pressure tube reduces the flow through the fuel, and thus the ability to cool the fuel.

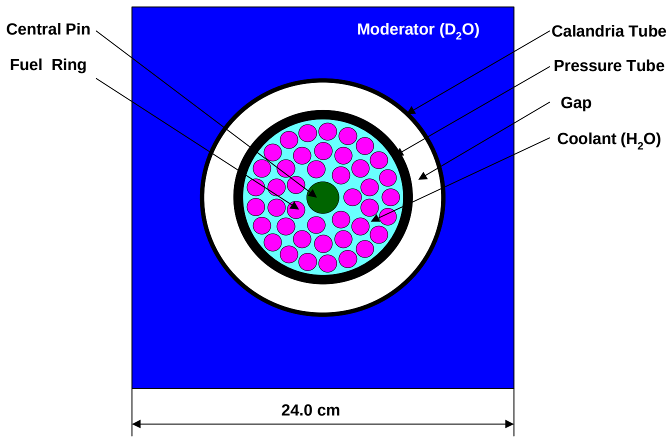


Figure 2: The basic ACR-1000 lattice cell

The gap between the calandria tube and the pressure tube has been widened in order to reduce the moderator-to-fuel volume ratio, which helps reduce the reactivity effect of coolant voiding.

3.2 Fuel Design

The fuel used in the ACR is uranium dioxide sintered in the form of cylindrical pellets and clad in zircaloy sheath using the 43-element CANFLEX-ACR fuel bundle design. The centre ring consists of one large diameter element, whereas the outer three rings consist of 42 elements with a smaller diameter. To reduce the coolant void reactivity during postulated accidents, dysprosium and gadolinium are blended, in a matrix of Zirconia, as burnable neutron absorbers in the centre element of the bundle, which does not contain any fissile material. The three outer rings of fuel elements contain low-enriched uranium (LEU) pellets. The fuel enrichment and the burnable poison concentration can be tailored to meet desired design targets of fuel burnup and coolant-void reactivity.

4. Reactivity Effects and Reactivity Control

4.1 Coolant Void Reactivity and Power Coefficient of Reactivity

CANDU fuel channel reactor designs have traditionally featured the advantages of small absolute magnitudes of reactivity coefficients. The flexibility of the ACR design allows these to be optimised. An important feature of the ACR-1000 core design is the selection of fuel parameters to achieve a small, slightly negative value of coolant void reactivity, achieved by the combined effects of a reduced lattice pitch and the addition of a neutron absorber to the central pin of the fuel element. A lattice pitch of 24 cm reduces the moderator-to-fuel ratio, while still maintaining all reactor-face maintenance activities, including the ability to replace single fuel channels. When combined with the light-water coolant, these features are chosen to deliver a small negative power coefficient of reactivity. Reactivity coefficients that are small and negative allow for an easy control of the reactor by slight adjustments of the control rod insertions. Small magnitude coefficients avoid the need for other large-scale emergency means of reactivity hold down such as boron injection into the coolant system as part of emergency coolant system action, and at the same time, render the consequences of accidents more benign.

4.2 Reactivity Control

All ACR-1000 reactivity and shut-off devices are located in the low-pressure environment of the moderator, and are hence not subject to large forces or stresses. Whereas traditional CANDU reactors employed light water as absorber in the zone control units, all reactivity devices (with the exception of the poison injection system) in the ACR-1000 are solid neutron absorbers of boron-carbide-filled stainless steel tubes sliding in zirconium alloy guide tubes. The cross section of the reactivity devices is a flat rectangle, to permit insertion in the tight lattice while retaining sufficient reactivity worth.

The ACR-1000 Reactor Regulating System (RRS) consists of zone-control units (ZCU) and mechanical control-absorber units (CAU). Each ZCU contains two independently-movable mechanical absorber elements, one driving in from above, one from below. The zone-control units perform the same bulk and spatial control function as the liquid-zone controllers in CANDU 6 reactors, but with the operational simplicity of solid absorber design. The power of each control region can be adjusted by varying the degree of insertion of the absorber in the corresponding unit. In this way, power levels across the core are maintained at design targets while individual fuel channels undergo refueling operations on line. The power in the centre of the core can be controlled by varying the degree of the overlap of the ZCUs in the centre. This enables fine control over peak fuel channel and bundle powers, thus increasing operating margins.

The ACR CAUs perform the same function as the mechanical absorber units in CANDU 6 reactors. The CAUs are designed to provide rapid controlled reductions of reactor power when power setback or stepback is initiated. The degree of insertion and the number of CAUs to be inserted are determined by the amount of power reduction required by the RRS.

5. Reactor Safety Systems

5.1 Shutdown Systems

ACR-1000 retains the two independent fast-acting shutdown systems (SDS1 and SDS2) present in other CANDU reactors. Each system is physically, logically and functionally separate from the other and from the RRS. Each of these shutdown systems is independently fully capable of rapidly shutting the reactor down in any postulated accident scenario, and to maintain the reactor in a shut down state.

The ACR-1000 Shutdown System 1 (SDS1) has vertical shutoff rods (SOR), which are designed to shut the reactor down quickly under emergency conditions. Shutdown is achieved by rapidly inserting neutron-absorbing elements into the reactor core. The insertion of the SOR is initiated by the control logic of the SDS1. The ACR SOR's are based on the shut-off rod technology employed in CANDU-6 units, with the adaptation to plate-type absorber cross section, in deviation from the previous cylindrical cross section.

The ACR-1000 Shutdown System 2 (SDS2) has horizontal liquid-injection nozzles, which cross the core at various locations. The SDS2 is designed to inject enough liquid poison (a concentrated gadolinium nitrate solution) to blanket the core within two seconds after actuation, sufficient to shut the reactor down rapidly during all postulated accidents. SDS2 contains enough poison to keep the reactor in the shutdown state under all foreseeable conditions. Again, the SDS2 delivery system takes advantage of components and features from the corresponding system on the CANDU 6 reactor design.

5.2 Regional Overpower Protection

Like the traditional CANDU reactors, the ACR-1000 is designed with a regional overpower protection (ROP) system, which triggers the shutdown systems when protective actions are warranted. The ROP system consists of two sets of self-powered platinum-clad inconel detectors distributed over the core. The sets are functionally independent from each other and are both further subdivided into four independent safety channels. The shutdown system will be activated when two out of four detectors of a system have registered a signal above a pre-set threshold value. Traditional CANDU reactors are equipped with a three-channel ROP system. With the four-channel system design of the ACR-1000, the chance of a spurious trip during testing of one of the safety channels is eliminated, simplifying operator testing. This further enhances the advantages of on-power fuelling, testing and maintenance that allow the ACR-1000 to achieve a three-year interval between maintenance outages.

5.3 Guaranteed Shutdown State

There is a strong desire from an operational point of view not to use an over-poisoned moderator to achieve a Guaranteed Shutdown State (GSS), due to significant activities and constraints required during outage operations. Additional absorber rods, of a mechanical design similar to the SOR's, are used to achieve a rod-based GSS without the need for poison addition in the moderator, except for start-up and fuelling ahead conditions. The reactivity depth of the GSS system is sufficient to keep the reactor in the guaranteed shutdown state indefinitely.

6. On-Power Fuelling

Like traditional CANDU reactors, the ACR-1000 has twelve fuel bundles per channel, which are replaced on-power at a rate that compensates for the reactivity loss due to the depletion of ^{235}U. The ability of on-power refuelling allows the fuelling engineer to maintain an optimised channel power shape, which ensures optimum power output at maximum bundle burnup, while adhering to the safety margins imposed on the channels powers. The on-power fuelling represents a safety feature as well, considering that no poison needs to be present in the moderator, nor does excessive core reactivity need to be held down by reactivity devices.

7. Burnup

The targeted discharge fuel burnup in the ACR-1000 is 20000 MWd/Mg(U), which is about three times the burnup in the current CANDU reactors using natural uranium fuel. Initially, the reactor will have a fresh start–up core and, after going through a fuel and core transition with fuel burnup of about 10000 MWd/Mg(U), will reach the equilibrium reference fuel and core configuration with fuel enrichment of up to 2.5% wt for ^{235}U and fuel burnup of about 20000 MWd/Mg(U).

The fuel-management flexibility of the ACR design allows further improvements in fuel burnup without requiring any modifications to the basic reactor design.

8. Alternate Fuels

The feature of on-power fuelling of individual fuel channels, combined with a flexible fuel bundle design, allows the ACR reactor to use a variety of fuel types and management strategies. Studies under way indicate that the ACR-1000 is adaptable to the use of recycled plutonium in the form of mixed plutonium and uranium oxide (MOX) fuels. In particular, the ACR-1000 design may enable the use of 100% MOX fuel in the reactor core without the need for costly reactor design changes or performance penalties. Furthermore, a number of very attractive options for establishing the thorium cycle in the ACR are being considered [2,3].

9. Conclusions

The ACR-1000 achieves substantial reduction in capital cost by using H_2O coolant, LEU fuel, in a compact D_2O-moderated lattice with respect to the traditional CANDU design. Full-core coolant void reactivity, as well as major reactivity feedback coefficients, are slightly negative under nominal operating conditions. The negative power coefficient aids in a smooth control of the power shape by the reactivity devices, and limits the speed and magnitude of power excursions following postulated accident scenarios. The flexibility provided by on-power fuelling and simple fuel bundle design enables the ACR to accommodate gradual increases in enrichment and hence in fuel burnup and to adapt to the use of various fuel cycles. This flexibility allows it to meet strategic energy requirements as they evolve over time, and to respond to changes in technology and resource availability.

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APPLICATION OF TECHNOLOGY NEUTRAL METHODOLOGY FOR ASSESSMENT OF DEFENCE IN DEPTH APPLICATION FOR GENERATION IV REACTOR SYSTEMS

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ABSTRACT

The Generation IV International Forum (GIF) Charter envisions the safe and reliable operation of nuclear systems as an essential priority in the development of next-generation systems. Safety and reliability goals broadly consider operations, improved accident management and minimization of consequences, investment protection, and reduced need for off-site emergency response. This emphasis on enhanced safety and reliability has been duly reflected in the Policy Group's selection of the system designs, as well as in recognition of the need to establish in 2004 a system cross cutting methodology working group on Risk and Safety (RSWG). The group was charged with the responsibility to promote a homogeneous approach to safety and quality in the design of Generation IV systems and define the framework for safety design and evaluation methodology which could be applied to all reactor systems. It was recognized that development of advanced and enhanced safety assessment methodologies might be needed for this purpose.

The paper provides an overview of the recent GIF Risk and Safety Working Group developments of technology neutral nuclear safety requirements and safety assessment methodologies. In particular, it addresses the applicability of "Objective Provision Tree" (OPT) methodology for verification of defence in depth application for new designs, as well as on the need to complement this methodology with traditional safety assessment techniques such as deterministic accident analysis and probabilistic safety assessments. The later are needed to judge on the adequacy of the safety provisions and relative importance of different provisions. The experienced gained with the pilot applications of the OPT methodology to different reactor designs will be also discussed along with the further effort needed to validate the applicability of this new proposed methodology for GIF reactor systems.

1. Introduction

The Generation IV initiative concerns the identification, development and demonstration of one or more new nuclear energy systems that offer advantages in the areas of economics, safety and reliability, and sustainability, and could be deployed commercially by 2030. Six innovative reactor system concepts are covered under the Generation IV International Forum (GIF) agreement, namely: Gas-Cooled Fast Reactor, Lead-Cooled Fast Reactor, Molten Salt Reactor, Sodium-Cooled Fast Reactor, Supercritical Water-Cooled Reactor and Very-High-Temperature Reactor System. All concepts potentially present a diverse set of design and safety issues. A number of these issues are significantly different from those presented by the earlier generations of light water reactors. The overall success of the Generation IV program depends, among others, on the ability to develop, demonstrate, and deploy advanced system designs that exhibit excellent safety characteristics.

In order to address nuclear safety concerns in a consistent manner throughout the GEN IV reactor systems and in order to provide the designers with safety concepts and methods that can help guiding their R & D activities towards improved safety, a system cross cutting methodology group on Risk and Safety (RSWG) was established in 2004. The primary objective of the RSWG is to assure a harmonized approach on long-term safety, risk and regulatory issues in development of the next generation systems. To this end, the RSWG focuses particularly on defining safety goals and evaluation methodology and advising and assisting the GIF Experts Group and Policy Group on interactions with the nuclear safety regulatory community, and other relevant interested parties including IAEA. The RSWG is comprised of representatives nominated from interested GIV Members and different System Steering Committees. The Euratom is represented in this group by the Institute for Energy of the Joint Research Centre, EC and VTT (Valtion Teknillinen Tutkimuskeskus) from Finland.

2. Objective Provision Tree

The Generation IV research and development program is guided by a GIF IV Technology Roadmap document (Ref. [1]) which identified three specific safety goals for Generation IV systems “to be used to stimulate the search for innovative nuclear energy systems and to motivate and guide the R&D on Generation IV systems”:

1. Generation IV nuclear energy systems operations will excel in safety and reliability.

2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

While the RSWG recognizes the excellent safety record of nuclear power plants currently operating in GIF member countries, it believes that advanced technologies and a coherent safety approach hold the promise of making Generation IV energy systems even safer and more transparent than this current generation of plants. In its first two years of existence, the RSWG has focused on defining the attributes that are most likely to help meet these Generation IV safety goals, and identifying methodological advances that might be necessary to achieve or demonstrate achievement of these goals. The results of the group developments are summarized in RSWG Report on the Safety of Generation IV Nuclear Systems [2].

One of the important issues addressed in this report is the need to apply the fundamental principle of defence-in-depth in a consistent manner from the very first stage of the reactor system design. Defence in depth is the key to achieve safety robustness, thereby helping to ensure that Generation IV systems do not exhibit any particularly dominant risk vulnerability. To meet these objectives the defence in depth has to be implemented in a way which is systematic, exhaustive, progressive, tolerant, forgiving and well-balanced.

To help GIV designers to correctly implement the defence-in-depth, to assess how well the latter has been applied for their reactor systems and identify areas which deserve further research, RSWG has suggested to utilize the Objection-Provision Tree (OPT) methodology complementing it with required traditional deterministic and probabilistic safety assessments.

The notion of OPT is first defined in the IAEA TECDOC 1366, Considerations in the Development of Safety Requirements for Innovative Reactors: Application to Modular High Temperature Gas Cooled Reactors [3]. In addition the IAEA Safety Series Report No 46: Assessment of Defence in Depth for

NPPs [4] presents a practical tool for inventorying the defence in depth capabilities of an operating NPP, including both the design features and the operational measures. For this purpose the definition of defence in depth and the guidance on its implementation agreed upon by international consensus (Ref. [5] & [6), have been combined into a logical graphical framework – Objective Provision Trees - that can be used for assessing the comprehensiveness and quality of defence in depth at a plant.

The OPT is a graphical representation of design safety architecture which identifies for each level of defence-in depth, with regard to each of the safety functions, of the provisions required to realize the required missions. All five levels of defence in depth are to be covered by the plant design.

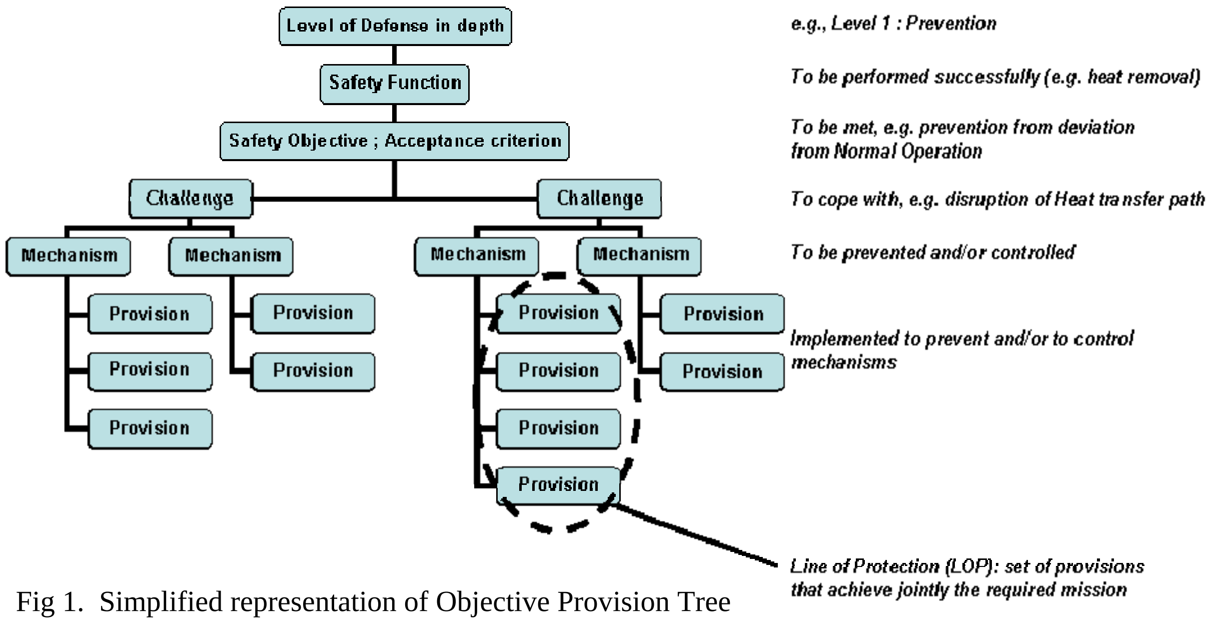

Fig 1. Simplified representation of Objective Provision Tree

In other words, for given objectives at each level of defence, a set of challenges[1] is identified, and several root mechanisms[2] leading to the challenges are specified. Finally, to the extent possible the comprehensive list of safety provisions[3], which contribute to prevent that the mechanism takes place, is provided. The broad spectrum of provisions, that encompass the inherent safety features, equipment, procedures, staff availability, staff training and safety culture aspects, are considered [4].

For easier understanding, the user-friendly application of the method, including the overview of all challenges, mechanisms and provisions for all levels of defence, is illustrated in the form of "objective provisions trees!"[4].

3. Application of OPT to GEN IV Reactor Systems

[1] Challenges: generalized mechanisms, processes or circumstances (conditions) that may impact the intended performance of safety functions; a set of mechanisms have consequences which are similar in nature.

[2] Mechanism: specific reasons, processes or situations whose consequences might create challenges to the performance of safety functions.

[3] Provisions**:** measures implemented in design and operation such as inherent plant characteristics, safety margins, system design features and operational measures contributing to the performance of the safety functions aimed at prevention and control of the mechanisms to occur.

[4] Objective provisions tree: graphical presentation, for each of the specific safety principles belonging to the five levels of defence in depth, of the following elements from top to bottom: (1) relevant safety functions; (2) safety objective of the level; (3) identified challenges; (4) constitutive mechanisms for each of the challenges; (5) list of provisions in design and operation preventing the mechanism to occur or achieving its control.

The application of the Objective Provision Tree (OPT) methodology to assess the implementation of defence- in depth concept for an operating LWR plant, and in particular for the WWER 440/V213 reactor units at Bohunice NPP [4] was considered and analyzed by the RSWG. In addition, a pilot study was conducted to assess the methodology applicability to the Japan Nuclear Cycle Development Institute Sodium Cooled Fast Reactor [7] as part of GEN IV reactor systems.

In the first case it was noted, that for LWR type of designs detailed OPTs are already developed by the IAEA [4] for 68 specific safety principles/ safety functions which are linked to the three fundamental safety functions. The user of the methodology needs simply to assess whether the identified provisions in the OPTs are present at his/ her plant. In some cases some alternative provisions might be identified by the methodology user, however the main difficulties are experienced in assessing the adequacy of the provisions. The need to use deterministic, probabilistic and other type of analyses for this purpose as well as additional research in some cases is clearly demonstrated.

The application of OPT methodology to innovative reactor designs needs, however more efforts to develop first OPTs which will have to evolve with the evolution of the design. For instance, at an early design/conceptual stage only very general OPTs could be developed for the fundamental safety functions, while at a more advanced stage OPTs have to be developed to address in detail the subsidiary safety functions(similarly to the 68 specific safety principles identify for the LWRs). While there might be some similarity with LWR, it is clear that GIV IV is deploying innovative technologies and concepts which will require new thoughts to be given on the way safety is ensured for these new type of reactors, possible new failure mechanisms, and why not to the safety principles themselves. Rigorous approach will have to be applied in development of the OPTs for each of the reactor systems in order to ensure that all levels of defense-in-depth have been addressed in comprehensive manner.

Difficulties could be expected in identification of all challenges and mechanisms and possible provisions for each of the GEN IV Reactor systems. The demonstration of the adequacy of the provision performance will be in its own a challenge in many cases. This will have to be done through applying of good engineering and performance of high quality research. The role of probabilistic safety assessment (PSA) to assess reliability of the safety provisions will also need an innovative approach since many of the design solutions deployed for GEN IV are not supported by any operational experience so far. It is however, RSWG belief that applying OPT can help designers to define their R&D plans in the most cost effective manner by focusing on the provisions and phenomena with high contribution to safety.

4. Conclusions

Both studies, the Bohunice NPP and JSFR, have demonstrated that there is a lot of potential benefits for the GIF reactor designers from the application of OPT methodology. It can help to ensure that at each stage of reactor system design adequate provisions are foreseen to ensure the application of all 5 levels of the DiD concept and identify topics where more research and development activities are needed to justify and prove this statement.

In order to facilitate the designer’s use of OPT methodology it will be important for RSWG to develop an application guide. This guide can be established in an electronic form to facilitate the building up of OPTs and provide predetermined options to be selected for safety objectives, functions, challenges, mechanisms, provisions (at least for the technology neutral ones). The reference to any available safety requirements for any of those items can also be incorporated and be available for designer consultations. The experience in building detail OPTs for LWR shall be repeated in the GIF reactors context for the new reactor systems. The aim would be to help designers to identify all necessary provisions to ensure safety and define R&D activities which are needed to demonstrate the adequate capabilities of selected provisions.

As any safety assessment methodology, the OPT has its limitations which are mainly related to the evaluation of the adequacy of the identified provisions and their prioritization or determination of their safety significance. It is clear that for these issues traditional deterministic (accident analyses) and probabilistic safety assessment will be needed to complement the OPT. A number of iterations of combined use of all these methods will have to be done to ensure that a comprehensive and systematic assessment has been performed for each of the GIF reactor systems.

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THE BALANCED SAFETY APPROACH OF EPR

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ABSTRACT

General safety objectives taken into account for EPR design considered both harmonized safety requirements of Safety Authorities and safety experts of France and Germany and the European Utility Requirements (EURs). This led to design an "evolutionary" PWR that has benefited from the experience feedback of LWRs in operation. However innovative features were also considered and included in the design where necessary.
The safety features were defined to ensure with a very high level of reliability the safety functions. Reliability target is achieved through an adequate combination of redundancy, segregation and diversity. The selection and implementation either of active safety features or passive and inherent safety provisions depend upon the advantages and drawbacks of the various design options.
Such a balanced safety approach allows to meet most of the requirements of the Nuclear Safety Authorities of western countries, without the need for significant design variations, thus allowing an international generic design to be developed with the benefits of standardization.

1. Introduction

AREVA's Evolutionary Power Reactor (EPR) results from a French-German cooperation set up to develop this large 4-loop PWR of the latest generation [1]. EPR is presently under construction in Finland for TVO and in France for EDF. In different countries, utilities are considering EPR as a major option for their new builts; one of the reasons is the wide acceptability of the design due to the balanced safety approach that ensures a very high level of safety.
This paper aims at giving a synthetic view of this approach; it is not intended to address here all aspects of the safety case but rather to focus on the mitigation of the faulted conditions; other important topics such as improving radioprotection or limiting the production of radioactive waste are not here dealt with.

2. The general safety objectives

French and German Safety Authorities and safety experts worked closely together during the EPR basic design phase and, in 2000, the "Technical guidelines for the design and construction of the new generation of pressurized water reactors"(TGs) [3] were endorsed. The TGs require significant safety improvement to be incorporated in the design of the next generation plants, in comparison to NPPs presently in operation, however in harmonising the requirements of France and Germany a balanced approach was sought rather than simply a summation of requirements identified by each country.
By this time, the European Utility Requirements (EURs) were defined by utilities from different countries with very diverse approaches to NPP regulation and licensing. The utilities wished to promote standardisation of the designs of new NPPs, and therefore also sought harmonisation of design rules, in particular with regard to nuclear safety.
EPR design was developed in accordance with both the TGs and the EURs.

3. The "evolutionary" design

The evolutionary design of EPR is in line with the TGs requirement as well as the wishes of many European Utilities. The reactor benefited from development, design, construction and operational

experience of the hundreds of LWRs in operation worldwide and took the best key technological features of the French N4 and of the German Konvoi PWRs.
The designers implemented an exhaustive, progressive and robust defence to guarantee a very high level of protection to investment, persons and the environment.

- Exhaustiveness is based on a deterministic approach supplement by safety assessment, leading to comprehensive analyses of design bases events, design extension conditions, internal and external hazards.
- Progressiveness starts with the implementation of surveillance and limiting functions which will react should the control system failed and avoid to actuate the protection functions and goes as far as measures to preserve the integrity of the containment and avoid significant radioactive releases should a severe accident occur.
- Robustness is ensured by seeking reliability of the safety relevant features through redundancy, segregation and diversity and the use of known materials and technologies for design measures whenever possible.

3.1 Deterministic and probabilistic safety assessment

The safety principles and criteria against which the EPR was developed are based on a strong deterministic defence-in-depth safety concept complemented through a probabilistic safety assessment (PSA). Probabilistic consideration was incorporated from the outset into the design process in order to identify accident sequences capable of leading to severe core damage or significant releases of radioactivity, to evaluate their probability of occurrence, and to assist in implementing design features to reduce the contribution of such sequences to the overall risk.
The early use of PSA provided also a basis for assessing the relative advantages of different design options, while verifying design compliance with initial project objectives.

3.2 Design Basis Conditions and Design Extension Conditions (Risk Reduction Categories)

Faulted conditions taken into account at an early stage of the design were extended in comparison with the previous reactor generation. The whole plant life and all operating modes were considered, in particular the shutdown states are explicitly addressed in the deterministic and probabilistic fault analyses, they should not contribute predominantly to the core melt frequency.

First, Design Basis Conditions, i.e. postulated event initiated by the failure of one component or one of the I&C function, were used primarily for designing and sizing the Protection and Safeguard systems. Anticipated Operational Occurrences (Condition 2) were addressed for designing the surveillance and limitation features which aim at avoiding that small deviation from normal operating condition could evolve towards more adverse conditions, the benefit provided by such surveillance and limitation feature is assessed through the PSA. However it is also determistically checked that, should these limitation provisions failed, the protection and safeguard systems ensure that criteria for core integrity are met.
Infrequent accident (Condition 3) and Limiting accident (Condition 4) are analysed with a conservative approach to give an adequate degree of confidence in the defence efficiency.

Second, three types of Design Extension Condition were considered:

1) The "complex sequences" that could lead to core melt due to multiple failures. Such sequences are derived from PSA analysis, they result either from a complete loss of a safety function after occurrence of a design basis initiating fault, e.g. Anticipated Transient Without Scram, loss of off-site power and failure of the four emergency diesel generators (station blackout)… or from combination of independent events. The deterministic analysis of the complex sequences is aimed at demonstrating the effectiveness of the safety measures implemented to reduce the risk of core melt to a very low, thus acceptable, level (e.g. additional small diesel generators to cope with the station blackout).
2) Severe Accidents (SA) are analyzed to assist the design features for preventing large early releases in case of a postulated core melt; this is described in a next paragraph.

3) Specific studies address fault analysis of event that have been excluded from the design basis due to probabilistic or deterministic reasons but are nevertheless analysed in order to check the absence of any cliff-edge effect in the plant safety demonstration, or to introduce additional safety margins in the design of certain systems and components, if necessary, to avoid this effect. For instance, the double–ended break on the main coolant lines (2A-LOCA) is analysed, despite the application of the break preclusion concept to the main primary systems, in particular it is verified that the pressure and temperature in the containment building remain lower than their design values.

3.3 Severe accident

The emphasis given to the reduction of the core melt frequency is backed up by taking into account severe accidents in a deterministic way, and mitigation measures are designed so that the associated maximum conceivable releases would require only very limited protective measures [4].

Core melt sequences which would lead to large early releases are "practically" eliminated; provisions against such a scenario have a reliability so high that this kind of event can be excluded; for instance dedicated diverse valves are implemented to supplement the three pressurizer safety valves, thus avoiding any high pressure core melt sequence.

To deal with low pressure core melt sequences, specific features are provided for retaining the melt within the containment to prevent penetration of basemat by corium-concrete interaction and to keep the confinement integrity without the need for any containment venting.

The effectiveness of the mitigation features was comprehensively verified for different possible accident scenarios. As there is significant amount of uncertainty surrounding the main physical phenomena involved in the demonstration of in-vessel corium retention by outside flooding of the Reactor Pressure Vessel, cooling the corium on a dedicated spreading is seen as a more robust solution for large PWRs.

3.4 Internal and external hazards

External and internal hazards that could affect the plant are identified on a generic basis, and provision made to ensure that the risk from the hazards is commensurate with the overall frequency and release targets. A deterministic analysis aims at ensuring that the safety functions needed to bring the plant in a safe shutdown state and to limit radiological releases are not unacceptably affected by hazards.

Hazards are also covered by the Probabilistic Safety Assessment to verify that they do not contribute predominantly to the risk of core melt or large radioactivity releases.

The layout configuration and structural technology is chosen in order to provide a high degree of robustness of the whole facility in relation to internal and external events, and also to accommodate unanticipated events. The plant layout contributes to the reliability of the safety functions by protecting the relevant equipment, in particular by bunkerisation and segregation.

4 Achieving reliable safety functions

Safety features were defined and designed to achieve the safety functions, with a very high level of reliability through an adequate combination of redundancy, segregation and diversity. The challenge for the designer was to defined an optimal mix between largely proven solutions derived from the large experience basis and innovative features needed to meet new requirements.

Innovative features were included in the design where necessary for providing cost-competitive solution while ensuring a significantly higher level of safety (especially to prevent and mitigate severe accidents, or to reduce the risk of radioactive release by mitigating any Steam Generator Tube Rupture) relative to previous generation of PWRs.

For design, selection and validation of these options, a lot of R&D work were performed, mostly in France, with a significant support of the Commissariat à l'Energie Atomique (CEA) and in Germany.

4.1 Redundancy

The main safeguard systems (e.g. Safety Injection System, Emergency Feedwater System) are arranged together with their associated control and support systems in a 4-train configuration. This arrangement

leads to a simple design concept for the fluid system: each train is connected to one of the reactor loop. This architecture makes it possible for a system to fulfil its function even if one train is unefficient because of the impact of the postulated initiating event, a second train is affected by a single failure while another train is unavailable due to preventive maintenance.
However, when a four-train configuration is not necessary, a twofold configuration is adopted. For example the Extra Boration System or the Containment Annular Space Ventilation System are needed to mitigate postulated events which do not impact the efficiency of one train, and preventive maintenance on these systems is not scheduled during power operation. Therefore the safety function can be fulfil with a 2-train system, assuming a single failure on one train.

4.2 Segregation

Due consideration is given to the possibility of common cause failures that limits the benefit provided by adding identical train. The likelihood of common mode failure due to internal hazards is minimized by physical and spatial segregation of the redundant trains of the safety systems. There are four safeguards buildings, every train is located in a different building with its support systems.
These buildings are also protected against external hazards such as a large commercial airplane crash either by very thick outer walls (for building #2 and #3) or by physical separation; building #1 and building #4 are on both side of the reactor building, only one of these buildings could be affected by the hit, the other remaining operable.

4.3 Diversity

To ensure that a diverse means can be used as a backup whenever the total failure of a safeguard system induces a significant risk of core melt or radioactive releases - the event sequences that fall into this category being identified by probabilistic Safety Assessment - any safety grade system function can be backed-up by another system or a group of systems. The drawbacks due to diversity such as the use of more complex system or additional maintenance burden are outweighed by the risk reduction provided by such a design.
Some example of diversity implementation are:

- Two small diesel generators, diverse from the four main diesel generators supply power to two safeguard trains in case of a station blackout.
- A diverse I&C channel actuates reactor and turbine trip in case of failure in the Protection System; this channel is implemented in the Process Automation System outside the Protection System with adequate functional, equipment or software diversity between the digital I&C functions. Apart from being implemented between two hardware platforms, diversity is also introduce within the Protection System in order to prevent the occurrence of common mode failures, and two diverse means are used to switch off the control rods power supply in case of a reactor trip demand: trip breakers for interruption of power to all rods and trip contactors dedicated to every bank of four rods.

4.4 Inherent safety provisions, active and passive safety features

In any PWR a mix of inherent characteristics, passive and active features is used. For EPR, a balanced and comprehensive approach was sought with regard to the extent to which inherent safety provision should be sized and safety functions should be achieved by passive systems. The selection depend upon the efficiency, reliability, availability and balancing cost and productivity of the various design options.
For EPR, inherent characteristics provide additional design margins and extended grace periods for operator actions thanks to components with large water inventories such as the pressurizer and the steam generators.
Passive systems have technical assets and they may be seen as an help for communication with the public; however, passive systems do have failure modes; they may need an active triggering, they work well only if they are correctly aligned while dormant, they are not always easy to test and they may request specific system architecture and layout which make more difficult and/or costly protection against hazards.
As any PWR, EPR relies on passive features like accumulators for safety injection, safety valves for over-pressure protection, gravity-driven control rod insertion…EPR designers addressed the potential inclusion

of passive features and many of them were evaluated at the beginning of the conceptual phase [5], but only few of them were included in the design.
Mitigation of Design Basis Condition relies mainly on active safeguard systems which are preferred for their proven design, their versatility and their ability to allow operators to keep their hands on the plant during perturbed situation. Their main weakness is the need of electrical power, it is counteracted, in case of a station black out, by the redundancy and the diversity of the emergency electrical sources. This allows the safety function to be achieved by active systems with a extremely high reliability level.

Some passive features were considered valuable to mitigate the consequence of a core melt accident and to preclude significant radioactive releases. Use of passive components and means in the early phase following the core melt allows to implement a simple and robust mitigation strategy which makes proper allowance for the plant state in this extreme conditions
Should a core melt occurs, passive hydrogen recombiners avoid a global hydrogen detonation that could challenge the containment integrity. The retention, the spreading of the corium and then the flooding and cooling of the spreading area by draining water from the IRWST (In-Containment Refueling Water Storage Tank) located inside the reactor building, are fully passive at all stages for at least twelve hours. The large heat capacity of the containment building makes not necessary an active heat removal from the containment during at least the first twelve hours that follow a core melt, thus providing plenty of time for recovery.

5 Outlook

The safety principles and criteria against which the EPR was developed were intended to result in an international standard design that should meet most of the requirements of the Nuclear Safety Authorities of western countries.
The evolutionary approach followed by the EPR designers led to an exhaustive, graduated and robust defence based on an optimized mix between largely proven solutions derived from a large operating experience and innovative features where needed to meet new requirements. Such a balanced safety approach achieves compliance with the harmonized safety requirements. It provides wide acceptability without the need for significant design variations, thus allowing an international generic design to be developed with the benefits of standardisation. This approach protects against licensing, construction and technical risks and their economics impacts.
This harmonized safety philosophy makes the EPR a major solution to lead the international nuclear renaissance which is raising.

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Session 18.1.1:

Generation IV rectors

EUROPEAN RESEARCH ON THERMAL HYDRAULICS FOR HEAVY LIQUID METAL ADS APPLICATIONS

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ABSTRACT

The objective of the European 6th framework project EUROTRANS is to demonstrate the technical feasibility of transmutation of high level nuclear waste using Accelerator Driven Systems (ADS). Within this objective the design of a European experimental ADS should demonstrate the technical feasibilities to transmute a sizeable amount of waste and to operate an ADS safely. This ADS will be a subcritical reactor system having liquid lead-bismuth eutectic (LBE) as coolant. The liquid LBE is also intended to serve as target material for the spallation reaction which forms a crucial part to the subcritical reactor core. Since LBE is used as core coolant and spallation material, knowledge of the thermal hydraulic behaviour of LBE is essential. Within the DEMETRA domain of the EUROTRANS project, basic thermal hydraulic studies in order to support the design and safety analysis of XT-ADS components and the development of measurement techniques have been started.

1. Introduction

The objective of the European 6th framework project EUROTRANS [9], sponsored by the European Commission, is to demonstrate the technical feasibility of transmutation of high level nuclear waste using Accelerator Driven Systems (ADS). Within this objective, the design of a European experimental ADS (XT-ADS) should demonstrate the technical feasibilities to transmute a sizeable amount of waste and to operate an ADS safely. Besides that, the conceptual design of a European Facility for Industrial Transmutation (EFIT) is foreseen. Both systems will be subcritical reactors having liquid lead-bismuth eutectic (LBE) and lead as coolant, respectively. This liquid metal is also intended to serve as target material for the spallation reaction which forms a crucial part to the subcritical reactor core. Since liquid metal is used as core coolant and spallation material, knowledge of the thermal hydraulic behaviour of liquid metal is essential. Due to the functional similarity between the XT-ADS and the so-called MYRRHA Draft 2 concept (shown in figure 1) as developed by the Belgian nuclear research institute SCK•CEN (Aït Abderrahim, 2005 [1]), this design was chosen as a starting point for the design of the XT-ADS.

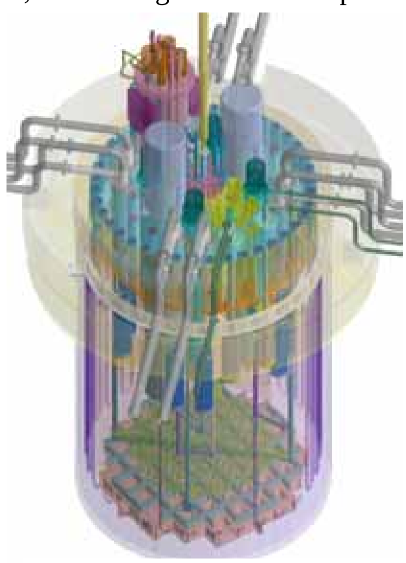

Fig. 1: Overall configuration of MYRRHA Draft 2 which serves as basis for the XT-ADS.

Within the DEMETRA domain (Fazio et al., 2006 [7]) of the EUROTRANS project, basic thermal hydraulic studies in order to support the design and safety analysis of XT-ADS components and the development of measurement techniques have been started. In particular, the work focuses on:

- Characterisation of the free surface flow for the windowless spallation target design.

- The interaction of LBE with water as secondary coolant.
- The development of measurement techniques for heavy liquid metal (HLM) flows.

The work on the characterisation of the free surface flow for the windowless spallation target is directly linked to the design of the windowless spallation target for the XT-ADS within the DESIGN domain of the EUROTRANS project. The interaction between LBE and water as secondary coolant has an impact on the design selection and safety considerations of the heat exchanger of the ADS. These studies are also used to prepare a large scale integral experiment which is foreseen within the DEMETRA domain in the CIRCE facility at ENEA (Fazio et al., 2006 [7]). Since a large number of European lead/LBE experimental facilities are involved, this work is also closely linked to the European Commission Integrated Infrastructure Initiative VELLA (Virtual European Lead LAboratory [4]). Furthermore, as the lead and LBE technologies developed within EUROTRANS are also applicable to a lead cooled fast reactor (LFR), this work is strongly related to the European Commission Specific Targeted Research Project ELSY (European Lead-cooled System [10]).

2. Characterisation of the Free Surface Flow in the Windowless Target

2.1 Windowless Target

As outlined before, due to the functional similarity between the XT-ADS and the MYRRHA Draft 2 concept, the latter has served as starting point for the design of the XT-ADS. Therefore, also the design of the spallation target of the XT-ADS (Schuurmans et al., 2006 [14]) is based on the windowless spallation target design of the MYRRHA Draft 2 concept. The limited space available for the external neutron source in the core of the XT-ADS and the high proton current, lead to very high proton beam densities. At present, no structural material is expected to withstand such extreme conditions at the operational temperatures foreseen for the XT-ADS during a reasonable lifetime of the spallation target of at least one year. Therefore, an LBE windowless spallation target is chosen in which there is direct contact between the proton beam from the accelerator and an LBE free surface flow. This results in a challenging task for the design of the spallation target. The design of the target nozzle has to be such that an LBE free surface flow is created within the geometrical constraints imposed by the compact sub-critical core which is adequate to remove the heat deposited by the proton beam. Furthermore, the design has to be compatible with the vacuum requirements of the beam transport system. These constraints lead to a design of the windowless spallation target with a vertical confluent flow as presented in figure 2.

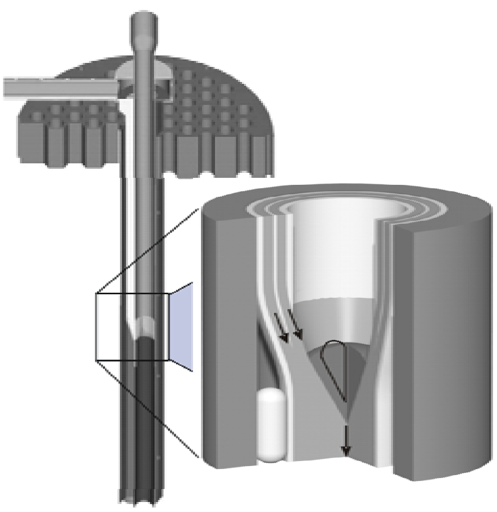

Fig. 2: Schematic view of the vertical confluent flow design of the MYRRHA Draft 2 design of a windowless spallation target.

2.2 Numerical Model Development

As no experiment can demonstrate the ability to transport the deposited heat in a windowless spallation target adequately, validated numerical methods are required. For this purpose, computational fluid dynamics (CFD) simulation methods are the most appropriate to capture the specific three-dimensional local effects of the LBE free surface including the heat deposition. This requires sufficiently accurate free surface modelling, predicting a unique (sharp) interface between LBE and beam vacuum in combination with adequate turbulence modelling. In the European 5th framework project ASCHLIM (Arien et al., 2004 [2]), it was demonstrated that sufficiently accurate CFD modelling of such free surface targets was not possible with the state-of-the-art methods available at that time. This is confirmed in other papers concerning this subject, e.g. Van Tichelen et al., 2000 [15], Van Tichelen et al., 2003 [17] and Fazio et al., 2005 [6]. Within the EUROTRANS project, the development of CFD methods for the simulation of the removal of deposited heat in the LBE windowless target has been envisaged. Different methods are assessed by NRG, FZK, and AAA and qualitatively compared to existing real size water flow experiments performed at UCL (Van Tichelen et al., 2001 [16]), mercury experiments performed at IPUL (Van Tichelen et al., 2001 [16]), and LBE flow experiments at FZK (Schulenberg, 2005 [13]). Table 1 summarises the different numerical methods assessed by the different partners.

Numerical Method	CFD Code	Institute
Volume of Fluid (VOF)	STAR-CD	FZK
VOF + Cavitation Module	STAR-CD	FZK
Euler-Euler	CFX10	NRG
Moving Mesh Algorithm (MMA)	STAR-CD	AAA

Tab. 1: Evaluated numerical models

First assessments have been made for the isothermal situation, i.e. without taking into account the heat deposition of the proton beam (Batta & Class, 2007 [3], and Roelofs et al., 2007 [12]). It is concluded that application of the VOF model in combination with the cavitation module in STAR-CD and application of the Euler-Euler model in CFX10 lead to promising results, although both models still require improvements. Furthermore, it is concluded that the VOF model without cavitation model does not lead to realistic results. The MMA method is still under evaluation. First results are expected by the end of 2007.

2.3 Experimental Campaign

An experimental campaign has started for the improvement and validation of the developed numerical models. This campaign foresees experiments in a water loop of UCL, see figure 3, and in a lead-bismuth loop in the KALLA laboratory of FZK. First experiments in the water loop at UCL have already been performed using the MYRRHA Draft 2 spallation target design assessing the influence of adding a mild swirl to the annular feeder flow on the behaviour of the free surface. Preliminary experiments have shown that adding a swirl of about 10% leads to an unacceptable vacuum core vortex in the central downcomer of the spallation loop. This confirms the conclusions from numerical simulations performed by FZK and NRG.

Fig. 3: Water loop at UCL

3. LBE-Water Interaction

3.1 Experimental Campaign

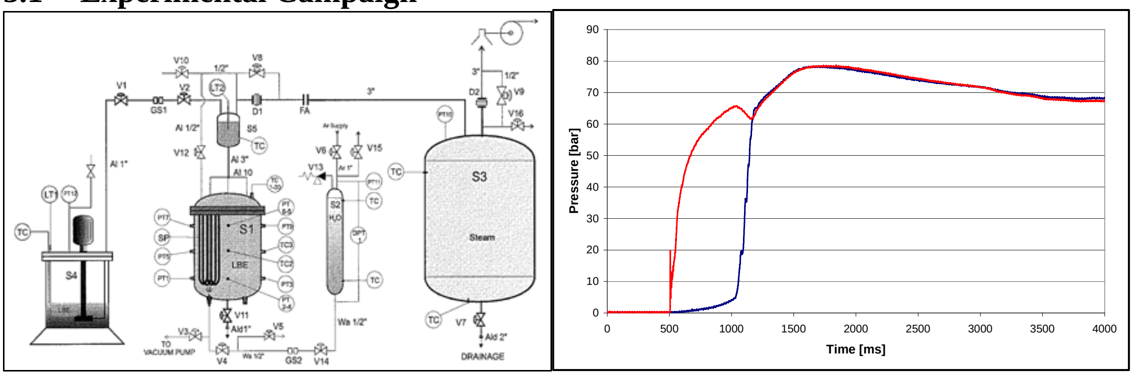

Fig. 4: LIFUS 5 test facility and pressure evolution in the reaction vessel (S1 - red line) and the buffer tank (S5 - blue line) during campaign 1

The XT-ADS and EFIT design foresee the presence of heat exchangers or steam generator modules placed inside the main vessel. This allows direct contact between LBE as primary coolant and water as secondary coolant in the case of a tube rupture. Since the probability of a tube rupture cannot be neglected, the consequences of such an accident have to be assessed. The experimental campaigns, which are performed in the LIFUS-5 facility of ENEA in Italy, aim at assessing the physical effects and possible consequences related to the interaction of LBE and water in representative conditions.

For this purpose, a steam generator mock-up is placed in a reaction vessel (S1) filled with LBE. Water is injected near a steam generator mock-up into the LBE. Fast pressure transducers and thermocouples at various locations register the pressure and temperature evolution during the experiment. A first experimental campaign aimed at obtaining first of a kind data of LBE-water interaction, has been performed successfully injecting pressurised water at 70 bar in the reaction vessel of LIFUS-5 containing LBE at 350 °C (Ciampichetti, 2007 [5]). Figure 4 shows the LIFUS-5 facility and the pressure evolution detected in the reaction tank S1 and buffer tank S5 during Test n.1. For the applied conditions, a pressure increase of about 10 bar above the injection pressure (70 bar) was observed and no steam explosion occurred due to the fast pressurisation of the system.

3.2 Numerical Program

The experimental data are also used for the validation of the SIMMER III code by ENEA/UNIPI and CEA. The SIMMER III code is a general fluid dynamics code coupled with a space-time and energy-dependent neutron transport kinetics model (Tobita et al., 2006 [18]). First simulations with a two-dimensional model performed by ENEA/UNIPI show a reasonable comparison between the simulation results and the experimental values. The pressure increase above the injection pressure was predicted correctly. However, the exact value of the pressure peak and the time evolution give reason for improvement of the numerical model by extending the model to three dimensions and by a more accurate geometrical representation of the steam generator mock-up.

4. Development of Measurement Techniques

Measurement techniques are developed for thermal-hydraulics experiments and for operational techniques in the XT-ADS and EFIT reactors. These techniques are tested within the laboratories of FZD, FZK, and SCK•CEN. The focus is on local velocity meters, integral contactless flow meters, and free surface level sensors. Two types of flow meters and two types of free surface level sensors will be described hereafter.

4.1 Flow meters

Contactless electromagnetic flow meters (EMFM) based on different principles are developed in parallel. One EMFM is based on the principle of phase shift and is developed by FZD, see Priede et al. (2006) [11]. This EMFM is validated against a commercial flow rate sensor as well as local velocity measurements using Ultra Doppler Velocimetry (UDV) in a GaInSn-loop at FZD. Furthermore, the EMFM measuring device is made resistant against temperatures up to 800°C. Figure 5 shows two developed devices which are ready for further testing in existing liquid metal loops. The device on the left is able to measure flow rates in channels up to 85 mm. The other device can be attached to a channel using a clamb. The latter system can be used in channels up to 34 mm and temperatures up to 800°C.

Another EMFM under development by FZK is based on the principle of dragging magnetic field lines. This flow meter is able to detect the flow direction. Besides that, a self calibrating method is developed for this type of flow meter. First successful tests have been performed in the KALLA laboratory at FZK.

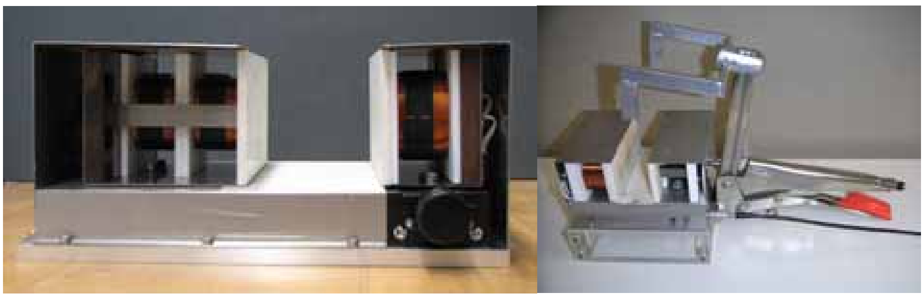

Fig. 5: Developed flow meters based on phase shift

4.2 Free Surface Measuring Techniques

Free surface measuring techniques are required for experimental as well as operational purposes. Concerning the experimental purposes, the measuring technique requires accurate measurement of the free surface shape and position. For this purpose, FZK is developing a non-invasive detection method based on the double layer projection technique (DLP). The proof of principle is demonstrated on a static and a rotating mirror. Further validation is foreseen on a circular hydraulic jump experiment (Hillenbrand et al., 2007 [8]).

Concerning the operational purposes, the measuring technique has to fulfil different requirements. During operation of the XT-ADS, accurate and frequent knowledge about the position of the free surface is required for reactor and beam control. In combination with the not readily accessible location of the free surface in the core of the reactor this leads to very stringent requirements for the technique under development: the distance between sensor and surface is about 10 m, the accuracy should be lower than 1 mm, and the measuring frequency should be about 1 kHz. SCK•CEN has selected a time of flight (TOF) technique for this purpose.

5. Summary

This paper summarises the ongoing work performed within the framework of the ‘advanced thermal-hydraulics and measurement techniques’ workpackage of the DEMETRA domain of the European integrated project EUROTRANS. This work focuses on the characterisation of the free surface flow for the windowless spallation target design, the interaction of LBE with water as secondary coolant, and the development of measurement techniques for heavy liquid metal (HLM) flows. Main achievements are:

- Development of numerical methods for the simulation of the isothermal windowless target;
- Determination of the influence of adding a mild swirl in a windowless target water loop;
- Performance of a first of a kind LBE-water interaction experiment which shows a pressure increase above the injection pressure and no occurence of a steam explosion;
- Reasonable results for the simulation of the first campaign of the LBE-water interaction experiments using a two-dimensional model in the SIMMER III code;
- Development of EMFM devices for the contactless measuring of HLM flow rates;
- Development of DLP free surface measuring technique for determination of free surface shape and position in experiments.

Acknowledgments

The authors would like to thank all participants (AAA, ANS, CEA, CRS4, ENEA, FZD, FZR, NRG, SCK•CEN, and UCL) involved in the workpackage ‘advanced thermal-hydraulics and measurement techniques’ of the EUROTRANS project. The work described in this paper was carried out in and supported by the FP6 EC Integrated Project EUROTRANS No. FI6W-CT-2004- 516520.

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ELSY : NEUTRONIC DESIGN APPROACH

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ABSTRACT

ELSY, European Lead-cooled System, aims to fulfil European Requirements of Minor Actinides burning and the GEN-IV strategic goals. It is a 600 MWe lead-cooled reactor (see a second ELSY paper in this ENC-2007). Two different core types are being preliminarily considered: the core made up of hexagonal wrapped assemblies, as used in sodium-cooled reactors, and wrapperless square assemblies, as common in PWR's. ELSY is conceived as "adiabatic reactor", i.e. it shall feature a unitary Conversion Factor and burn its self-generated MA. Consideration is also being given to the transmutation of a larger amount of MA, to address the issue of the MA legacy. A key-point of the core design approach is the small delta-T between coolant mean outlet temperature (480 °C) and allowable cladding temperature (now about 560°C), that requires a rather flat radial power distribution. This implies a core sparsely fitted with Control Rods, but offering the required worth.

1. Introduction

This paper deals with the preliminary neutronic design approach of ELSY (European Lead-cooled SYstem, see a second ELSY paper in this ENC 2007 Conference): a MOX fueled 600 MWe pool-type fast reactor, aiming to fulfil the GEN-IV strategic goals. Two different schemes of core are here considered: the first one made by conventional wrapped assemblies in hexagonal lattice and a second one made of wrapperless assemblies in square lattice with pins arranged in square bundle as well.
Beyond the self-evident qualitative differences rising up from the presence or not of the hexcan steel and from the quite different thermohydraulic conditions, it is mandatory to explore, evaluate and compare how much all these differences are impacting on the most important reactor parameters.
For this aim a number of parameters are assumed as a common basis, while the parameters able to exploit and show the differences in the two concepts are assumed as freedom degrees. The comparison is carried out paying attention, and weighting, to the extension in fulfilling the goals of sustainability, economics, safety and proliferation resistance.

Both the concepts assume in common the power plant, the type of fuel, the fuel residence time, the BU peak, the cladding max temperature and damage dose, the coolant temperatures, pointing to an unitary Conversion Factor.
On the other hand each concept optimizes, according to its own design, the pin diameter, the active height, the core diameter, the fuel enrichment, the volumetric fractions, whether to use or not the axial blanket for reaching the required Conversion Factor, the number and position of the Absorber Rods for the reactivity compensation, control and safety.
Minor Actinides burning is a key point. The ELSY should be an "adiabatic reactor" in the sense that it produces its own new fuel (Conversion Factor = 1+ reprocessing losses) and burns its own self-produced MA, without any material exchange with the environment, except loading natural or depleted Uranium and unloading fission products. Nevertheless to cope with the MA legacy it will be able to burn in addition even MA coming from other nuclear plants.

2. Specifications and aimed performances

The main aimed specifications of the ELSY core to kept in common in the two schemes are collected in the table 1 (see other paper of L. Cinotti et al.).

ELSY PLANT AREA	TENTATIVE PARAMETERS
Power	**About 600 MWe (1500 MWth)**
Thermal efficiency	**About 40 %**
Primary coolant	**Pure lead**
Primary system	**Pool type, compact**
Primary coolant circulation (at power)	**Forced**
Primary coolant circulation for DHR	**Natural circulation + Pony motors**
Core inlet temperature	**~ 400°C**
Core outlet temperature	**~ 480°C**
Fuel	**MOX with assessment also of behaviour of nitrides and dispersed minor actinides**
Fuel handling	**Search for innovative solutions**
Main vessel	**Austenitic ss, hanging, short-height**
Safety Vessel	**Anchored to the reactor pit**
Steam Generators	**Integrated in the main vessel**
Secondary cycle	**Water-supercritical steam**
Primary Pumps	**Mechanical, in the hot collector**
Internals	**As much as possible removable (objective: all removable)**
Hot collector	**Small-volume above the core**
Cold collector	**Annular, outside the core, free level higher than free level of hot collector**
DHR coolers	**Immersed in the cold collector**
Seismic design	**2D isolators supporting the main vessel**

Tab. 1 ELSY main specifications

2.1 Fuel

At first step, an oxide fuel without MA is proposed to be assumed: UO_2-PuO_2 MOX with a maximum allowed enrichment of 35 wt % of reactor grade Pu in heavy metal and with 95 % of theoretical density (TD). Nitride fuel will be considered later as an option.
At the second step, oxide fuel containing minor actinides (MA: Am, Cm and Np) up to about 5 at.% on HM will be considered.
At this stage of the ELSY conceptual design, the actinide isotopic vectors chosen in IP EUROTRANS [4] are proposed to use for minor actinides. Theses vectors were obtained in the result of mixing of MA coming from the spent UO_2 fuel (90 %) and the spent MOX (10%) of a typical PWR unloaded at the burnup of 45 MWd/kgHM, then cooled down and kept in storage for a period of 30 years. Plutonium is extracted from the same spent UO_2 but with the storage period of 15 years [5]. Depleted

U remaining in production of UO_2 fuel for LWRs or U obtained in reprocessing of LWR spent fuel is usually used for industrial MOX production.The isotopic vectors of the Pu is in Table 1.The relative atomic content of neptunium, americium and curium in total MA can be expressed as follows: Np:Am:Cm = 3.88 : 91.82 : 4.30 at.%.

Reactor grade PLUTONIUM			
Isotope	Molar mass g/mol	Content wt.%	Content at.%
^{238}Pu	238,0496	2,332	2,348
^{239}Pu	239,0522	56,873	57,015
^{240}Pu	240,0538	26,997	26,951
^{241}Pu	241,0569	6,104	6,069
^{242}Pu	242,0587	7,693	7,616
Pu	239,6493		

Tab. 2. Isotopic vector of plutonium.

In order to assure a high efficiency of the reactor operation and the requirements of non-proliferation, a long fuel life-time in the core should be aimed. In ELSY, where liquid lead is used as coolant, the corrosion of the pin cladding will play a major role. The existing laboratory studies on compatibility of different structure materials with molten lead show that some of them can resist a corrosion attack of the liquid metal flow with a velocity of 1.8-2.0 m/s and temperature of 560 °C during 12000 hours under oxygen control conditions. A small thickness of the formed corrosion layer (4-5 microns) allows making a prognosis that their operation time at this temperature can be extended to 50 000 hours (more than 5 years) [1]. However, there is no similar experience under in-pile conditions. New advancements in the development of the corrosion resistant steels and protective layers (i.e. GESA treatment) indicate that longer operation periods can be envisaged in a near future [2].
Thus a fuel life-time in the core can be assumed to be 5 years as a realistic option and, tentatively, 10 years as a futuristic option.
Of course the residence time of the fuel in core is ruled even by the allowed fuel burnup and allowed cladding damage.

2.2 Cladding

The choice of cladding material is of critical importance both from economic and safety viewpoints. Ferritic-martensitic steels (FMS) and austenitic steels (AUS) are within the best candidates for the cladding material. FMS show a lower swelling rate and embrittlement under irradiation at T > 350 °C and higher resistance to dissolution in the oxygen-free Pb and Pb-Bi(e), compared to austenitic steels. However, they have a higher corrosion rate in the presence of oxygen [7].
The existing experience of LMFR operation and the performed irradiation studies of the cladding materials for fast spectrum reactors demonstrate that optimized austenitic steel AIM1 (15-15 Ti mod Si) can withstand typical LMFBR operation conditions (sodium, 400-550 °C) up to the peak dose of 115 dpa [3]. At these temperatures, ferritic martensitic steels with 8-12 % chromium (such as T91, EM-10, HT9, F82H, …) are even more resistant (~200 dpa).
At this stage, it is proposed to choose FMS T91 as the first candidate for cladding material, taking into account its better irradiation resistance and ongoing R&D on technology of its protection against corrosion. The well-known austenitic steels AIM1, AISI 316 L and few Russian steels (EP823, EI852) are kept as a backup solution.

The cladding damage and the fuel burnup are correlated. In the neutronic modeling of the ADS MYRRHA, loaded with 30 % Pu MOX and cooled by liquid Pb-Bi eutectic, the mean damage of 25 dpa (the peak damage of 32 dpa) was obtained for the T91 cladding in the hottest rod at the average burnup of 28.68 MWd/kg HM in the hottest assembly (30.06 MWd/kg HM in the hottest pin) [8]. So, for an approximate estimation of the cladding damage rate, one can use a value of 0.83 dpa per 1 MWD/kgHM of fuel burnup. Then the aimed average burnup of 10 at % hma will result in a damage dose of about 88 dpa (validity of this simple relationship for ELSY has still to be confirmed).
In order to keep freedom in selection of cladding material, it is proposed to limit the cladding damage dose to 100 dpa, taking into account that synergy can exist between corrosion and irradiation effects. The optimistic limit can be 200 dpa, assumed to be reached in a near future with FMS and ODS steels.

2.3 Other core specifications

Available experience with Pb-Bi(e) and Pb coolants shows that the bulk velocity has to be limited to 2 m/s in order to avoid erosion problems during long-term operation [9].
A supplementary parameter, i.e. a core Conversion Factor of about 1, has o be introduced in the core pre-design specifications to achieve the sustainability. Moreover this condition is important for reduction of the number of intermediate reactor shutdowns needed for the core reconfiguration. It seems reasonable to do it not earlier than after every year of operation.
Following a passive safety approach, it is proposed that RDH will be removed by natural thermal convection; an effective height of 4 m between the heat exchanger and the core is foreseen.
Table 3 summarizes the specifications, based on arguments presented above, for the preliminary design of the ELSY core.

Characteristic	Value		
	real	future	
Thermal power	1500		MW
Core conversion factor	~1		
Minimum sub-cycle duration	1	5	y
Fuel residence-time (aimed)	5	10	y
Hottest assembly discharge burnup (aimed)	100	150	MWd/kgHM
Maximum allowed Pb bulk-velocity	≤ 2.0		m/s
Maximum allowed clad temperature	550	620	°C
Peak clad damage	100	200	dpa
Residual heat removal mode	natural convection+RVACS		
HX-core levels difference	4		m

Tab. 3 Other core specifications

3. Hexagonal wrapped assemblies core

3.1 Fuel

For the Hexagonal Wrapped scheme the following preliminary parameters have been precalculated. They need to be verified and optimized, along the core characterization, before making the selection vs the Wrapperless Square concept.

Estimated characteristic	Value	
Pellet outer diameter	8,88	mm
Central hole diameter	2,0	mm
Pellet height (postulated)	12,5	mm
MOX density (STP)	10550	kg/m³
Maximum allowed linear heating rate (EOC)	390	W/cm
Maximum allowed fuel power density (EOC)	630	W/cm³

Tab. 4. Estimated and pre-selected parameters of fuel pellet

Fuel rod :	Variant 1	
Pellet diameter	8,88	mm
Hole diameter	2,00	mm
Pellet height	12,50	mm
Clad inner diameter	9,18	mm
Clad outer diameter	10,58	mm
Fuel column height	900	mm
Supplementary breeding/burning segments	300	mm
Insulation pellets	10+10	mm
Gas plenum length	960+240	mm
Caps	50+50	mm
Fuel rod length	2520	mm

Tab. 5 Estimated and pre-selected parameters of fuel rod

The first estimate of the pitch of a rod bundle (defined as the "centre-to-centre" distance between the neighbour fuel rods) was obtained from the thermal balance. Then it was optimised in order to respect two other conditions: the coolant bulk velocity < 2 m/s and the pressure drop on the core < 0.12 MPa. Finally, the rod pitch of l_{pitch} = *15.5 mm* (x_{pitch} = *1.465*) has been fixed. The number of the rods in the hexagonal bundle was chosen to be 169: 168 fuel rods and 1 central bar for assembly manipulation. Then the inner plate-to-plate width of 203.0 mm was obtained for the assembly hexagonal wrapper. A wall thickness of 4.0 mm and a clearance of 5.0 mm between the neighbour assemblies were fixed taking into account the experience of the EFIT pre-design [10]. A schematic view of the radial cross-section of the proposed fuel assembly is presented in Figure 1 below, while Table 6 hereafter summarises the main geometrical parameters of this variant.

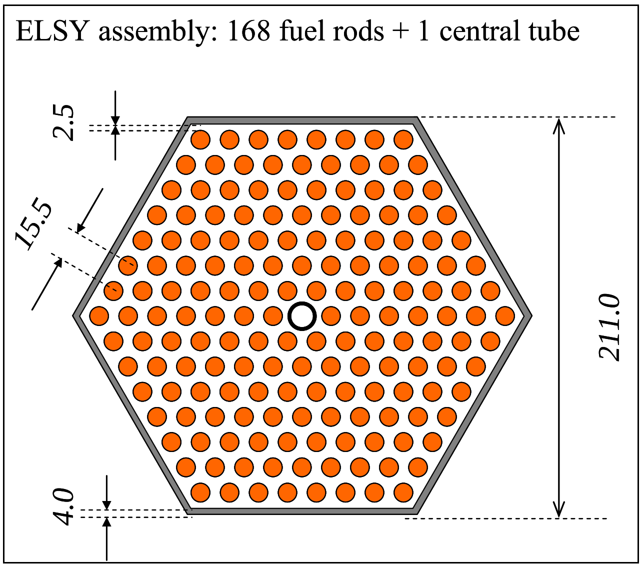

Fig. 1. Assembly radial cross-section at midplane

Assembly (hexagon):	Variant 1	
Fuel rod pitch (center-to-center)	15.50	mm
Rod pitch ratio	1.465	
Number of fuel rods in assembly	168 + 1 dummy rod	
(rod-row number)	15	
Wrapper inner width	203.0	mm
Wraper wall thickness	4.0	mm
Wrapper width	211.0	mm
Clearence between assemblies	5.0	mm
Assembly pitch (center-to-center)	216.0	mm
Total assemnly length (EFIT based)	3800	mm
Fuel mass in assembly	118.6	kg

Tab. 6 Hexagonal wrapped fuel assembly data

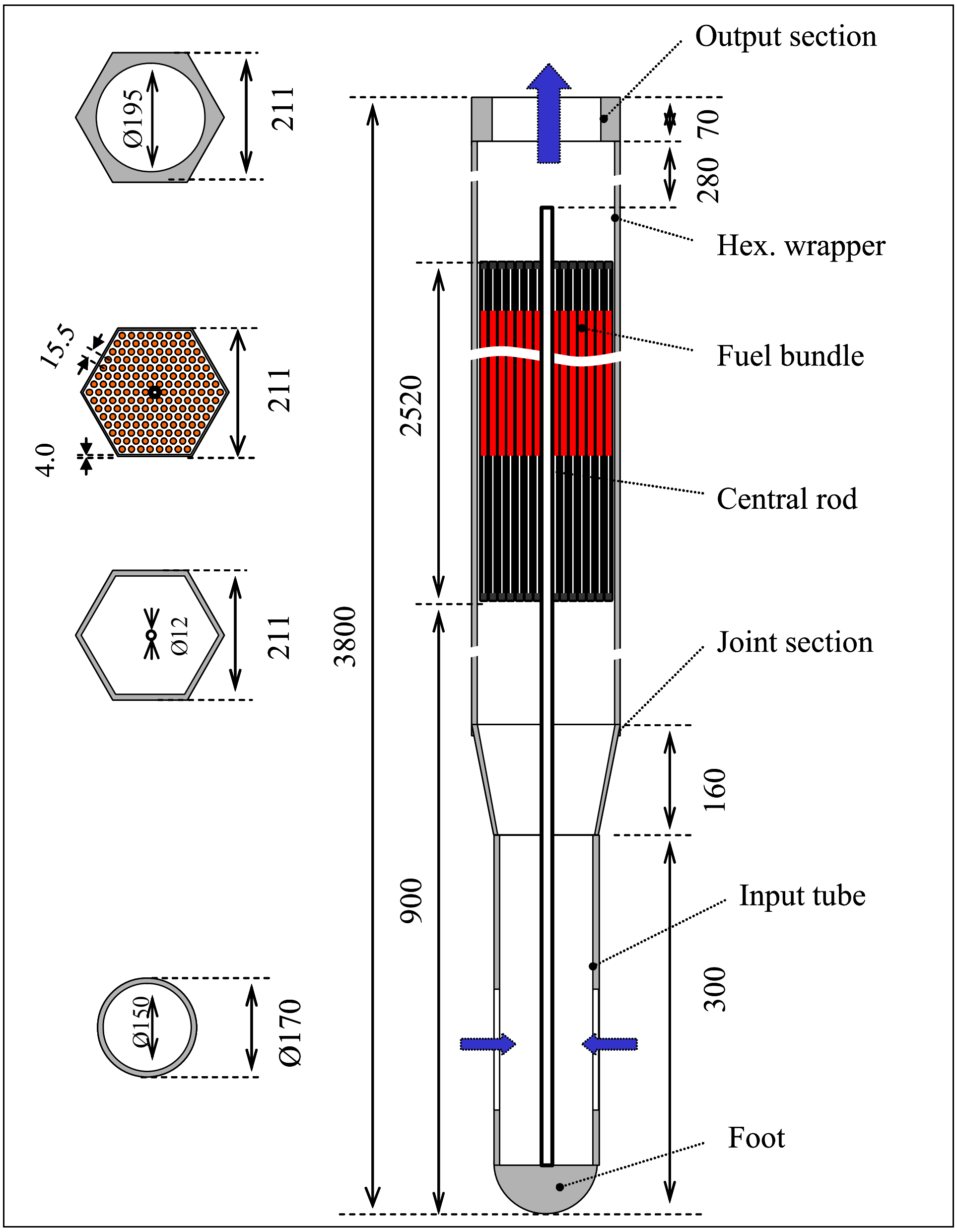

Fig. 2 Axial schematics cross-section of the fuel assembly.

At the initial pre-design stage, it was proposed to base the axial schematics of the hexagonal fuel assembly on the mechanical design of the EFIT fuel assembly done by Ansaldo Nucleare [10], but making it shorter and simpler where possible. A tentative axial schematic of the proposed assembly is presented in Figure 2.

3.2 Control rods

A reactivity compensation element has about the same design as fuel assembly, except the fuel rod bundle which is replaced by the bundle of 127 rods with a diameter of 15.76 mm. Each of the rods is filled with B_4C pellets of Ø 14 mm over the height of 1200 mm. Other elements and axial schematic of the absorber rod is the same as that of the fuel rods (Fig. 2). Main measures of the absorber assembly are given in Table 7.

Absorber rod :		
Pellet diameter	14.00	mm
Hole diameter	no	
Pellet height	20.00	mm
Clad inner diameter	14.36	mm
Clad outer diameter	15.76	mm
Column height	1300	mm
Insulation pellets	10+10	mm
Gas plenum and spring chamber lengths	560+240	mm
Top and bottom plugs	50+50	mm
Absorber rod length	2520	mm
Absorber assembly (hexagon):		
Fuel rod pitch (center-to-center)	17.70	mm
Rod pitch ratio	1.123	
Number of absorber rods in assembly	126+1	
Wrapper inner width	203.0	mm
Wrapper wall thickness	4.0	mm
Wrapper width	211.0	mm
Clearance between assemblies	5.0	mm
Assembly pitch (center-to-center)	216.0	mm
Total assembly length (EFIT based)	3800	mm

Table 7 Main geometrical measures of the absorber element

3.3 Core

A variant of the ELSY core configuration with the fuel and absorber elements described above is presented in Fig. 3. The main parameters of this core, estimated on the basis of simplified thermal and hydraulic calculations, are presented in Table 8.

Neutronic modeling followed by thermohydraulic and thermomechanical calculations has to be performed to estimate the key performances of this core. The results of these calculations will be used for optimization of the designs of the fuel rod, assembly and core.

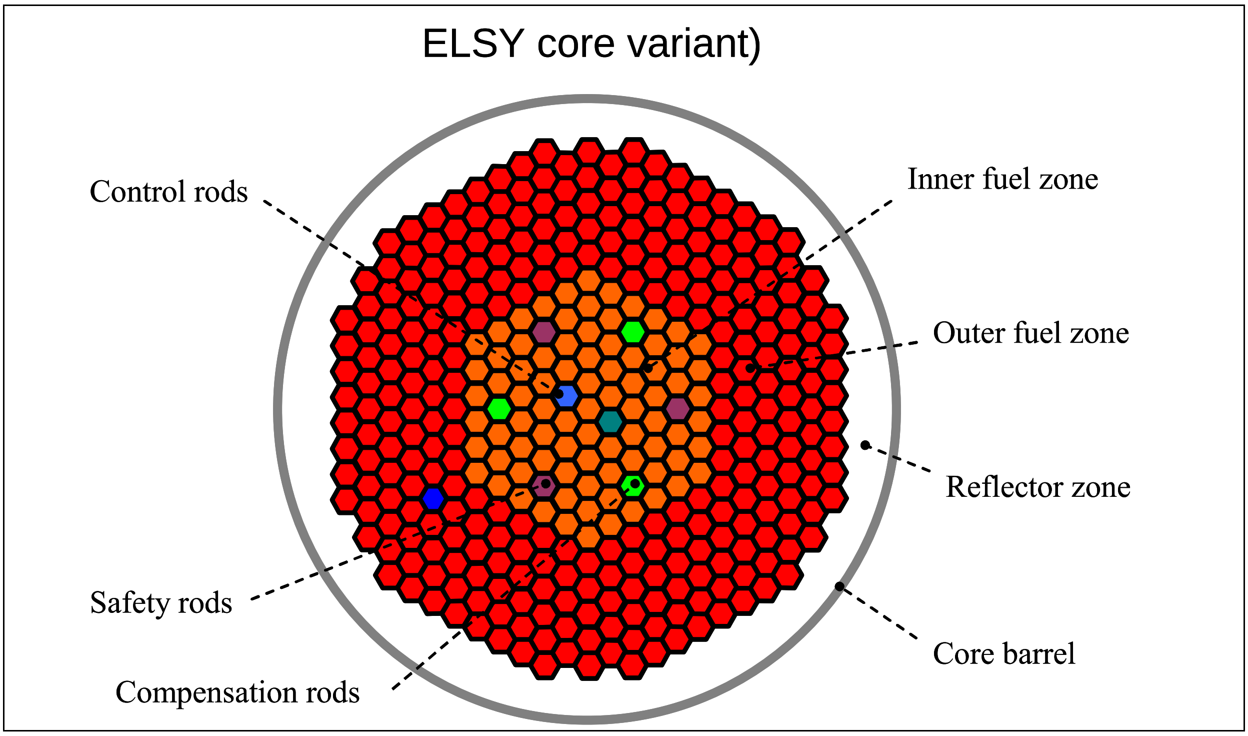

Fig. 3 A variant of configuration of the hexagonal wrapped ELSY core.

Normal regime	**ELSY-600**	
Core:		
Thermal power	1500	MW
Core diameter	4,54	m
Core height (driver+special)	1,20	m
H/D	0,26	
Fuel mass	38,24	t
Number of fuel assemblies (expected)	323	
Mean assembly power	4,65	MW
Mean rod power	27,7	kW
Radial power form-factor (expected)	1,30	
Axial power form-factor (expected)	1,30	
Colant inlet temperature	400	°C
Coolant average outlet temperature	480	°C
Lead mass flow-rate (maximum)	128,3	t/s
Lead maximum bulk-velocity	1,90	m/s
Pressure drop on core	0,12	MPa
Clad maximum temperture	540	°C

Tab. 8 Estimated parameters of the ELSY-600 core

4. Square Wrapperless assemblies core

The main aim of this first approach is to verify the Control Rods (CR) worth, the Conversion Factor (CF) and the Minor Actinide burning or build-up attitude.

4.1 Fuel element

At this step no specific design of the fuel element has been done yet. Nevertheless the data have been derived from the EFIT pre-design [10], in such a way to be confident to respect the cladding limit temperature with a Pb coolant velocity of some 1 m/s. The volumetric fractions are tuned on an average linear power rating of 220 W/cm. The related pressure drop through the core is expected to be of some 0.1 MPa.

The postulated wrapperless fuel element is sketched in Figure 4. It contains 285 fuel pins and 5 solid steel pins for structural needs

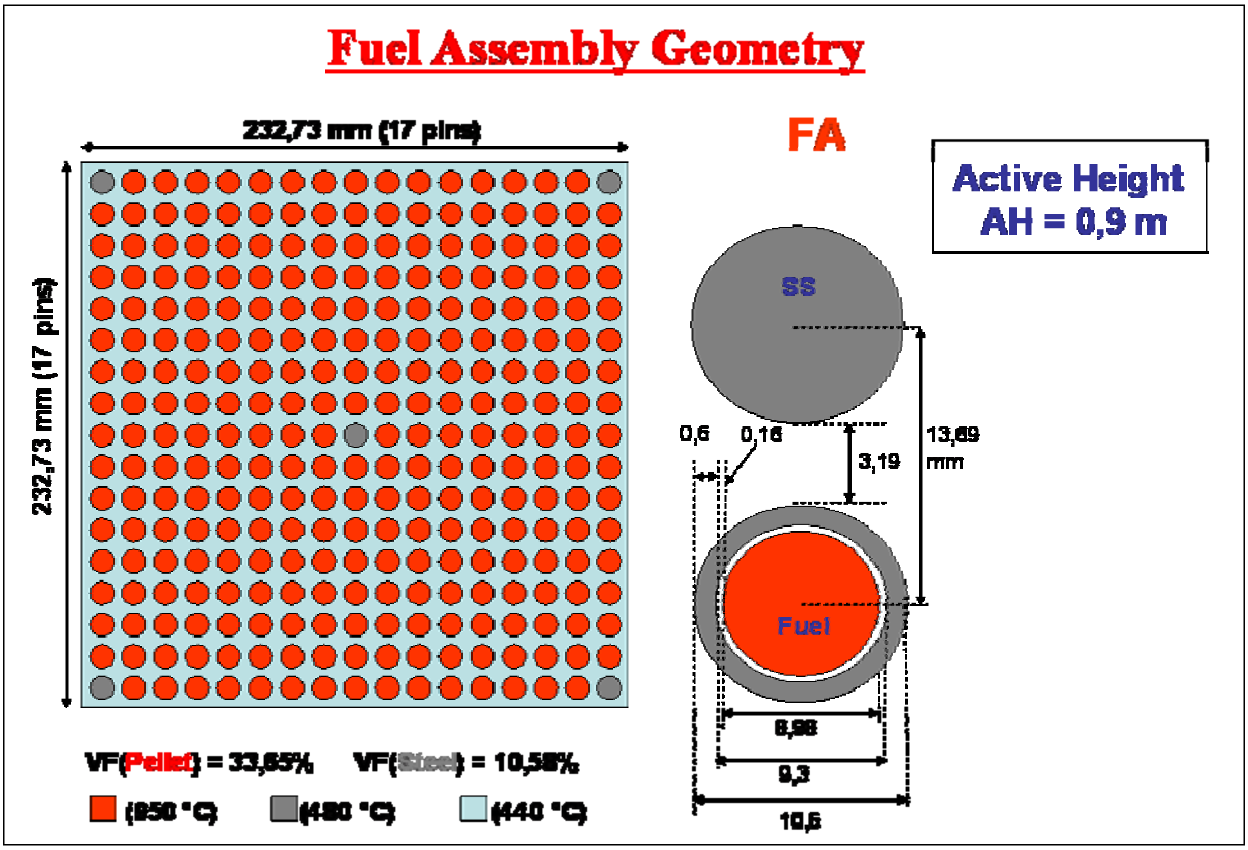

Fig. 5. Cross section of the wrapperless fuel element

4.2 Core

The arrangement of the core has been driven from the important key point of the sustainability, i.e. a unitary Conversion Factor, possibly without blanket (for proliferation resistance). The requirement on the Conversion Factor implies a unique ratio between Pu and U in the fuel, i. e. the enrichment. So fixed dimension and composition of the assembly, a suitable core has been laid down to reach the required reactivity.

The core is made by 259 square wrapperless fuel element (fig. 6), divided in 3 zones with different enrichments (whose average accounts for a unitary CF) to achieve a rather flat radial power distribution.

Taking into account two shielding rings the diameter of the barrel has to be about 5.4 m.

4.2.1 Control Rods

Some attempts of placing control rods surrounding the active core, to avoid spoiling the power radial distribution, have not shown a sufficient worth. Placing only a huge control rod in a devoted location at the centre would account for some 1000 pcm, while surrounding completely the active core by as many as 60 Control Rods would account for not more than 3000 pcm.
So at this step an overall of 12 Control Rods are foreseen, here not yet distinguished as their functions.
The next sketch in Figure 6 shows the cross section of the wrapperless square core, the three enrichment zones and the locations of the control rods.

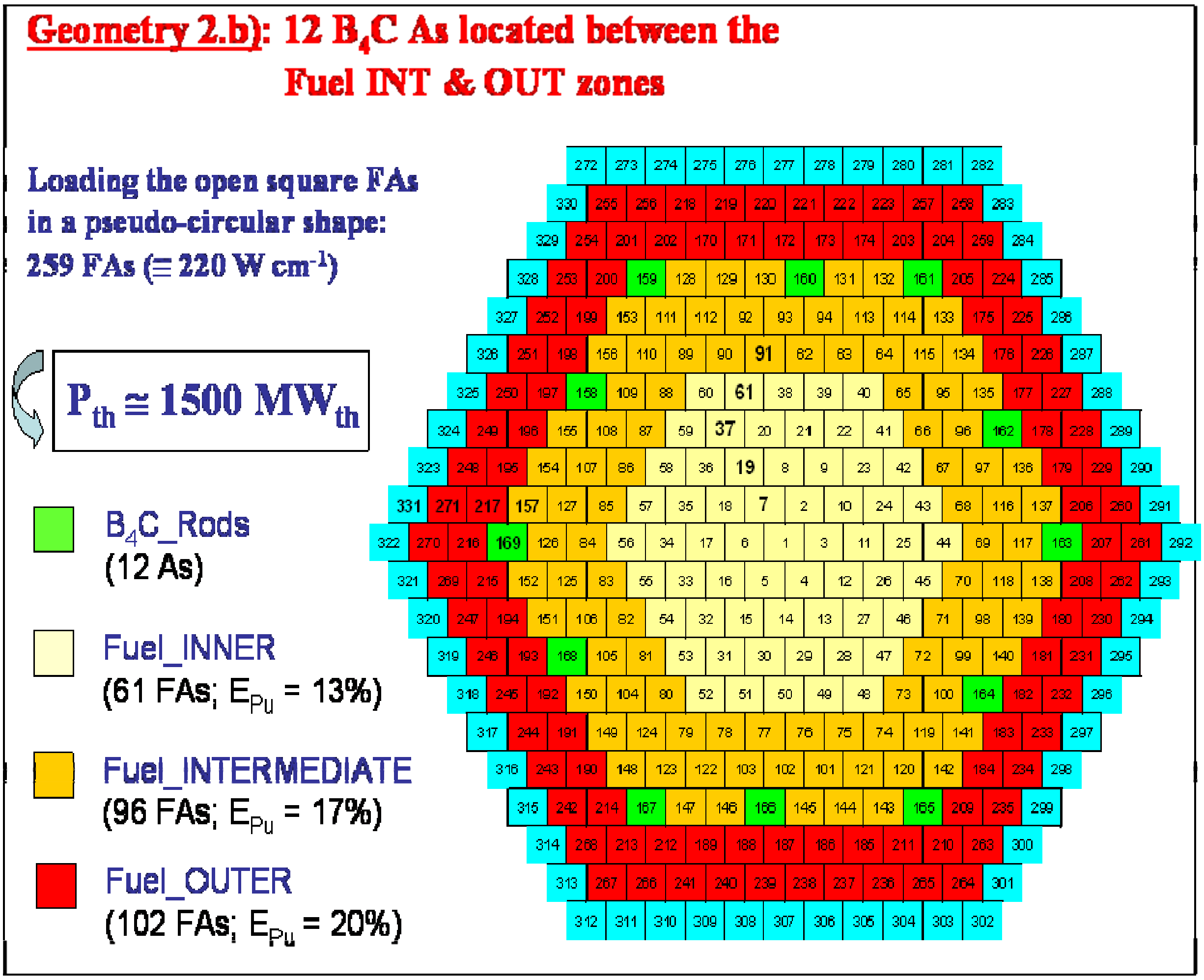

Fig. 6. Wrapperless square ELSY core

4.3 Control Rods worth

The Control Rods worth has been calculated in a cylindrical schematization. The CRs have been represented as circular ring, whose composition is the average of the whole hexagonal ring where they are (12 CRs + 30 Fuel elements). To have an idea of some compensation margin, 3 positions have been calculated: CRs out, CRs entirely inserted, and CRs inserted 30 cm. The overall worth of the 12 CRs is rather high: about 9000 pcm.
The next figure 7 shows the calculation scheme and the geometrical data of the cylindrized core.

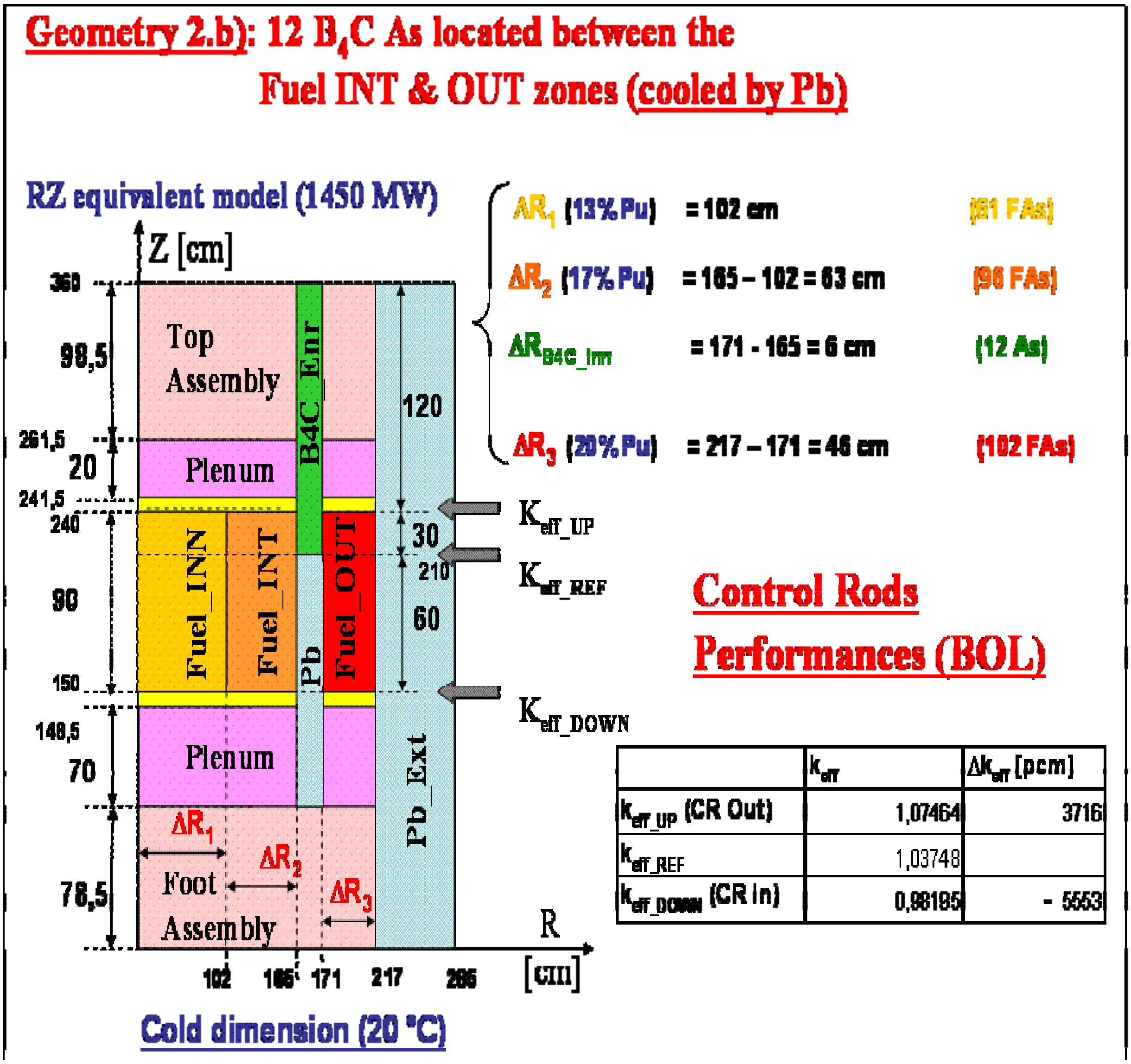

Fig. 7. Core and Control Rods calculation scheme

Since the peripheral surrounding set of 60 CRs has shown a poor effectiveness, a different surrounding has been tried. Taking profit from the small height of the core, some absorber has been placed over the fuel element heads (placing the plenum in the lower part). Its effect has been proved to be not negligible: using volume fractions in such a way to allow the coolant flow, a worth of some 3000 pcm has been calculated. Figure 8

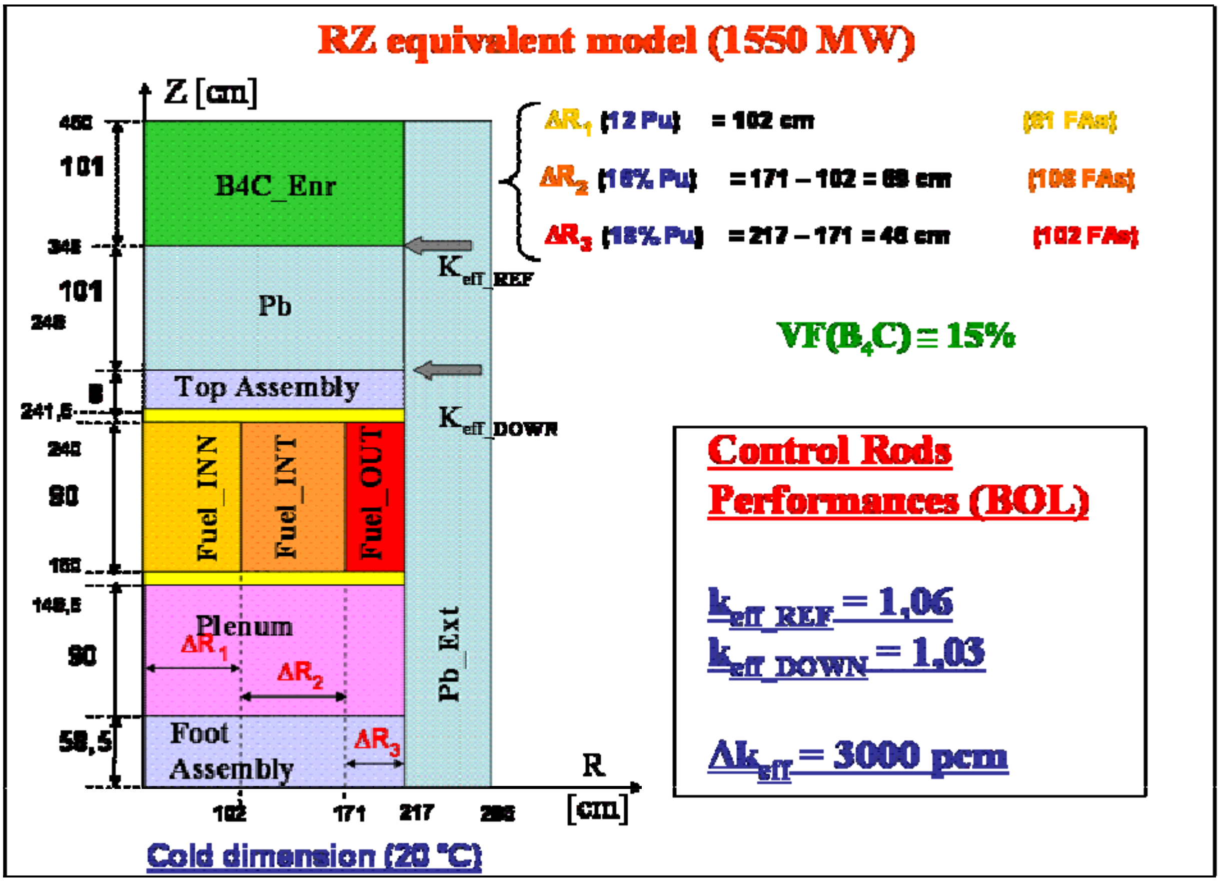

Fig. 8 Effectiveness of absorber placed over the fuel element heads

4.4 Cycle and breeding.

No specific cycle calculations have been done to this step, except the evaluation of the loss of reactivity in the time. It is evaluated as about 1000 pcm/year. The Conversion Factor, simply expressed as ratio between odd isotopes of the Pu, results 0.99.

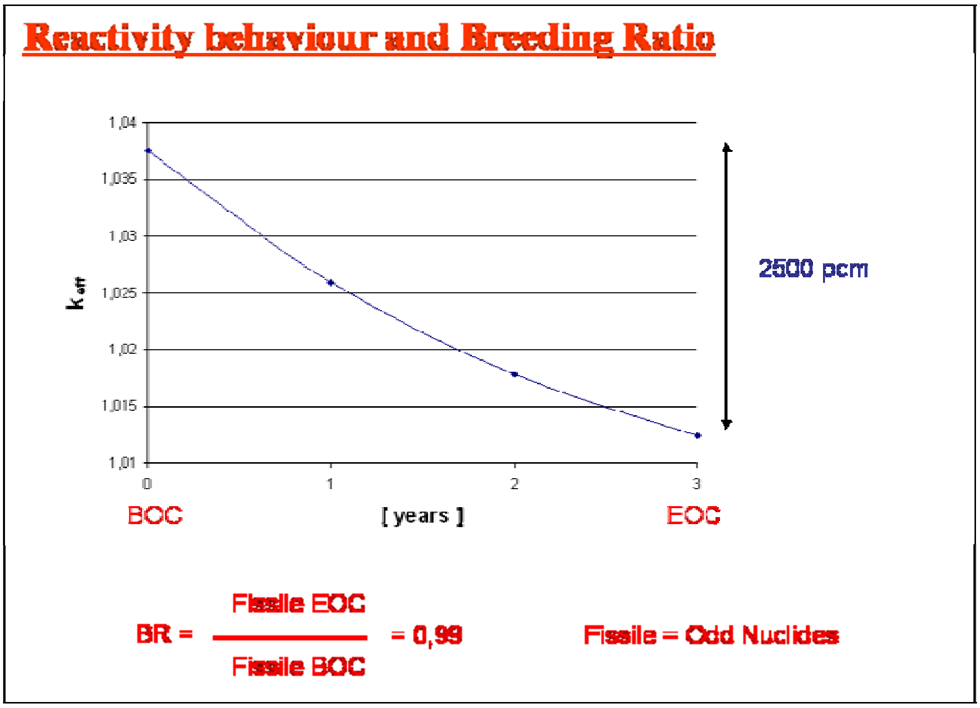

Fig. 9 Reactivity loss during the cycle

4.5 Mass balances

The total core inventory is 37 ton of fuel, 30 ton of depleted U and 7 ton of Pu. The sustainabilty target is quite well reached: in fact the Pu mass does not change significantly (-3.14 kg/Twh$_{th}$), while the large part of the energy is coming, both directly and indirectly, from the fission of the U (-40.96 kg/TWh$_{th}$).

4.5.1 Minor Actinides

Since the core studied has been loaded without any MA, they have been produced during the cycle. They have an equilibrium concentration in the fuel matrix and, more or less, they point toward this concentration by an exponential law. From the results reported in figure 10, it is possible to deduce the exponential law, the equilibrium amount and the time constant. The MA equilibrium content is about 210 Kg in the whole core, i.e. 0.57% in the heavy metal or 3% if referred to the only Plutonium. The time constant of the exponential is 5-6 years, that accounts for some 30 years to reach “naturally” the equilibrium.

In any case it means that having some 0.5-0.6% of MA, ELSY acts as an “adiabatic reactor”, without any exchange with the environment except for the Uranium loaded and the Fission Product unloaded for each refueling.

At this equilibrium concentration a worsening is expected for the Doppler effect by 5%, a worsening for the Void effect by 1-2 % and a reduction of the Delayed neutrons yield by 1% [11],

Should the MA content greater, for example 2% in hm, ELSY would show a net burning of MA coming from other nuclear plants, in addition to its self produced. For this content in MA the safety parameters worsening would be 20% for the Doppler effect, 5% for the Void effect and a reduction of 3% for the Delayed neutrons yield [11].

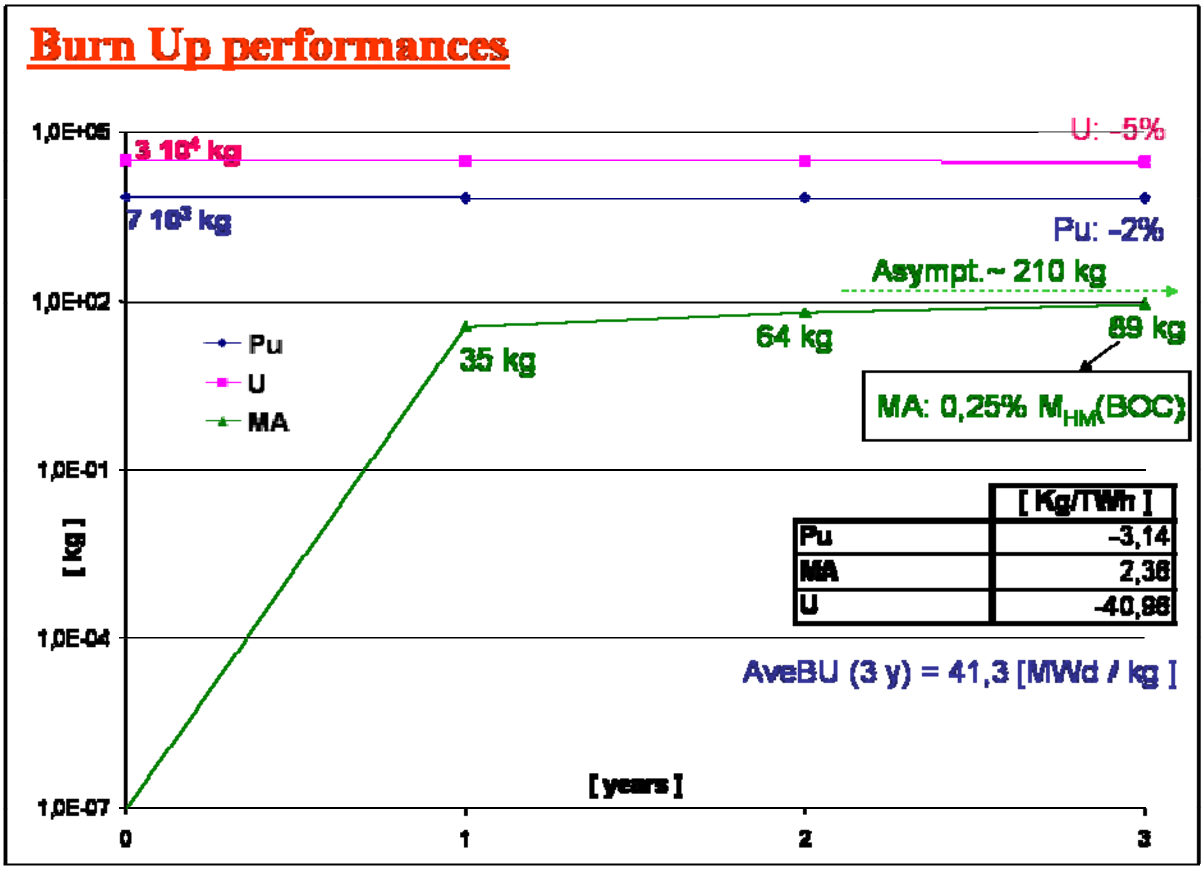

Fig. 10. Mass balances in the cycle: Uranium, Plutonium, Minor Actinides

5. Conclusions

Even if the studies are still in the first stage and is not yet selected the most promising variant, wrapped hexagonal vs wrapperless square, it is clear how ELSY can fulfil the requirements of sustainability and proliferation resistance (for other requirements see other ELSY paper in this ENC 2007 Conference, L. Cinotti et al.).
Quite interesting is the capability of reaching the unitary Conversion Factor without using neither radial blankets nor axial (at least in the wrapperless square variant), that increases essentially the proliferation resistance.
The possibility of burning its own MA without a significant worsening of the safety parameters (worsening at equilibrium: Doppler by 5%, Void by 1-2%, beta by less of 1%) makes ELSY a very promising “Adiabatic Reactor”.
Moreover ELSY can acts as a net burner receiving in addition other MA to be burnt.

ACKNOWLEDGMENTS

The authors thank the Partners of the IP-EUROTRANS project for their fruitful contribution to the project. Special thanks to the European Commission for the financial support trough the FP5 and FP6 programmes.

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FRENCH PROGRAM TOWARDS A GENERATION IV SODIUM COOLED FAST REACTOR

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ABSTRACT

Sodium-cooled fast reactor is in France a candidate for a prototype of 4th generation system to be built by 2020. A detailed working program has been defined to identify by 2012 the potential improvement tracks for later industrial development of these reactors. The goals for innovation are first identified: Progress of the safety with a special attention to severe accidents risk minimization and mitigation (defence in depth approach); Economic competitiveness of the system mainly by reducing the capital cost, the investment risks by enhancing in service inspection and repair capacities, and raising the availability; Consideration of advanced energy conversion system with particular emphasis on the reduction of risks linked to Na reactivity; Sustainability with fissile material management while minimizing the proliferation risk; Capacity for long-lived waste transmutation. The detailed content of the CEA, AREVA and EDF coordinated program is then described.

1. Introduction: a strategy in France

In March 2005, an inter-departmental committee stated that France should study sodium-cooled (SFR) and gas-cooled (GFR) fast reactors for the long term deployment of its nuclear fleet, together with Hydrogen production using high temperature reactors. In January 2006, the French president requested for the design of a generation IV system to be operated by 2020. In June 2006, the parliament voted a law to specify the future management of radioactive waste, and included the necessity to study by 2012 the options of future nuclear systems and to start operate a prototype in 2020. Finally, in December 2006, a second inter-departmental committee agreed on a technical roadmap for SFR, GFR and fuel cycle studies leading in 2012 to gather data to choose future options. This strategy is based on the necessity to save uranium and to reduce ultimate waste in the future. In this frame, a coordinated program has been launched by the CEA, AREVA and EDF to develop innovative SFRs. This program is presented hereafter

2. Specific goals for innovation

The research goals for SFRs able to be deployed by industry in 2040-2050, are derived from the generic objectives of the Generation IV International Forum. At first, the safety specifications are set up at a level at least equivalent to the one for generation III light water reactors (EPR). SFRs specificities (sodium and fast spectrum) are taken into account for behavior under severe accident conditions and for sodium risks. The occurrence of failing of critical components will be pushed back by extended and performing In Service Inspection (ISI).

The economy of the system should be optimized to achieve a plant cost acceptable for the industry. If the investment cost is traditionally high for a nuclear plant, it is still higher for a SFR. The financial risk engaged by the utility that buy the plant must be comparable to traditional power plants. So innovations to decrease the investment cost are to be searched for together with excellence in the availability of the plant.

The interest of fast neutrons is to allow an optimized management of nuclear matters. First, the reactor should be able to breed Pu with a breeding ratio at least equal to zero and must have the possibility to raise this ratio in the positive values in order to allow the deployment of a fast reactor fleet by producing a sufficient mass of Pu. This implies too the deployment of an industry that closes the fuel cycle and allows recovering Pu in a non-proliferant way.
The same reactor must have the capacity to transmute long lived nuclear waste. The core must accommodate the quantity of minor actinides (Americium, Neptunium and Curium) coming from its own closed fuel cycle and transmute most of it. It should be able, where needed, to transmute also wastes accumulated in the spent fuel of light water reactors. This implies the management of the minor actinides in the fuel cycle process.
Finally, the weak points of the sodium technology must be improved. In that domain, in service inspection is concerned together with an easy operation, repair and dismantling. One should think to the availability of the plant on a 60 years period while proving periodically that safety margins are sufficient.
A coordinated research program between the CEA, AREVA and EDF was launched to answer those previous objectives. It is organized in four main items which are reviewed in the four next paragraphs.

3. An efficient Core with enhanced safety

3.1 Reduction of the sodium void effect

The core performance in terms of fissile fuel management is to produce at least the same quantity of fissile Pu than the one burnt. A basic objective is to get a zero breeding gain (IBG = production/consumption -1) for the fissile core, to ensure that later optimization including blankets could easily reach positive breeding gain to allow the development of a fast reactors fleet. At that time, the blankets should be designed to avoid easy Pu separation from other actinides in a proliferation resistant way. Still for fissile material management, it is necessary to limit the mass of Pu that is needed per electrical MW. This objective to get a zero breeding gain brings too the safety advantages of a core with a low reactivity loss.
In addition, future SFRs may have enhanced safety. The sodium void effect, that induces a positive reactivity for an industrial size of the core, must be significantly reduced. The objective is to compensate it by negative effects from the Doppler and from other feedbacks like dilatation of the materials.
Several items will be looked at. The impact of the fuel element geometry is studied while comparing pin to plate. The geometry (height/diameter, cylindrical, annular, modular) of the core is an open parameter as are size, total power and power density, the relative volume of the components (fuel, structure and coolant) and the presence of moderator to sweeten the neutron spectrum. The sub-assembly geometry is also open to modifications: diameter of the wire spacer, impact of the sodium hydraulics, presence of a sodium plenum at the outlet of the fissile length (that voids as soon as sodium boils in the core, inducing important neutron leaks).
The nature of the fuel material is studied to compare advantage and drawbacks of oxide, carbide, metal and nitride. One needs to design a specific fuel element adapted to each fuel material in order to optimize each core. The development of such cores is conducted with a three steps program. First, neutronic calculations give reasonable materials repartition in the core. Then the fuel element and the sub-assembly design and technology are assessed. Finally, a detailed design of the core is used for several transient calculations that verify the performance and the safety of the given core. More details are given in [1].

3.2 Compaction risk management

Fast neutron cores are very sensitive to compaction. A special effort must be made to enhance their resistance to compaction, due to a seismic stress for example. The cores of the French SFRs (Phenix, Superphenix) are free to expand and compaction is limited by the contact at the pad level between sub-assemblies and especially dedicated stiff ones located at the core periphery. This effect could be enhanced.
The performance of a ringed core will be evaluated comparatively. Finally, the dynamic behavior of the core when a mechanical stress is applied will be studied using for instance the Symphony past experiments realized on a shaking table at CEA/Saclay. Modeling improvements either in static and dynamic situation will support these studies.

3.3 Core instrumentation

Taking advantage of the most recent technology evolution, the core instrumentation can be revised to develop new systems with a better efficiency and a higher dynamics.
The possibility of monitoring of the power distribution through in-core high dynamic fission chambers will be assessed.

For different applications such as temperature (measurement of the temperature inside the sub-assembly head, avoiding the discrepancy coming from the mixing jets at the core outlet), detection of boiling, presence of gas, ultra sonic detectors are being studied to be used under hot sodium conditions. Difficulties are to solve wetting of the transducer by sodium in order to improve sensitiveness, and the question of a piezzo-electric material able to sustain high temperatures during long times.
The clad rupture detection system can be optimized in order to reduce its response time delay. The clad rupture localization system is an expensive system with tubes that cross the closure slab. Reconsidering the complete system could allow a better performance, simplification and enhanced security.

3.4 Core performance

One point of the core performance is the fuel burn-up. The first limitation to the lifetime of the fuel elements comes from the dose rate on the structure materials (clad, canister). A specific program, on the long term, is envisaged to improve the dose rate acceptable on the clad.
First, the present optimized austenitic steel AIM1 will be confirmed able to reach 120 to 130 dpa using recent irradiations in Phenix. To go beyond, a new cladding material ferritic-martensitic strengthened by oxide dispersion will be developed, while advanced austenitics track will be kept as a backup. The hexagonal canister should withstand the same dose than the clad and possibly a higher temperature than presently. A ferritic steel like the T92 grade is foreseen.
In order to reduce the total diameter of the core, a compact lateral neutron shield must be developed. Coming from the EFR studies, a specific sub-assembly with neutrons absorber will be developed. Two problems are to be solved: the extraction of the heat produced in the material while the filling up density must be maximized, and the stability of the compounds on a sufficient long time period.
The lifetime of the control rods will be extended progressively.
As to fuel, it is anticipated that the core of the prototype mentioned in the introduction, will be an oxide core, the only one that allows for sufficient knowledge in the time scale of this reactor, including in accidental conditions. Innovation on the oxide fuel will be introduced as it is anticipated to be issued from so-called COEX^{TM} process. This process allows for coprecipitation of actinides and avoids handling of separated Pu, while allowing for simplified pelletization process. Qualification will be addressed in the timescale of the Prototype, and for that purpose an irradiation in Phenix (COPIX) is currently being prepared.

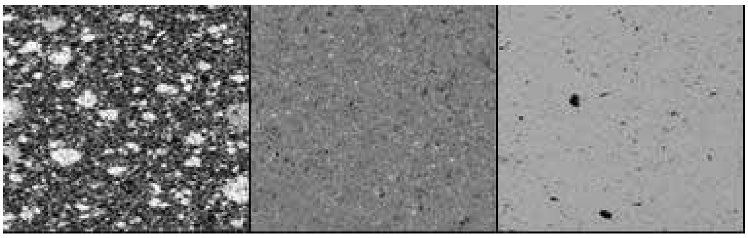

Fig 1. Mixed (U,Pu) oxides obtained with different fabrication porocesses. From the left to the right : 11% Pu , MIMAS process, 6%Pu COCA process and 27,5%Pi COEX process

A R&D program in a more long term will be pursued on dense fuels, especially carbide. Clearly, the prototype will be used in the frame of this program for experimental irradiations of such fuels, but application is foreseen in a longer timescale with the view at industrial deployment.

3.5 Minor actinides transmutation

The future cores of the GenerationIV SFR should be able to transmute minor actinides to reduce the quantity of ultimate waste. Several technical options are available. Comparative studies of the global efficiency of these options are yet underway for various options of the future French nuclear fleet.
Minor actinides can be mixed with the driver fuel, in a homogeneous way. In such a case their relative volume in the fuel is from 1% to 5%. They are quite easy to fabricate and handle, but all the fabrication process of the fuel needs additional adaptation and protection against radiation.
They can also be put in specific sub-assemblies with a high concentration, from 10% or more in volume. This is the heterogeneous way. In that case the number of sub-assemblies to manufacture requires a separate facility from the one for the

driver fuel. The choice of the matrix that contents the M.A. is an essential part of the research. Most of the experiments launched in the past were devoted to inert matrix, either ceramic or metal. Specific irradiation experiments are yet underway or under post irradiation examination with various materials and fuels. A new program is starting with a MA bearing UOX matrix featuring radial blankets that could be placed around the core. Such a solution opens the way to proliferation resistant blankets.
Other special concerns are on the one hand the fuel handling, according to the level of residual power even for a "fresh" fuel (especially in the case of the heterogeneous route), and on the other hand the detection of a clad failure of a sub-assembly containing M.A. as release of delayed neutron emitters has to be verified.

3.4 Simulation tools

Tools for core simulation include neutron physics, thermal-hydraulics, mechanics & fuel behavior. For neutron physics, the reference tool is ERANOS that has been validated in a wide range of situations. For thermal-hydraulics several tools are available, some are commercial like STAR-CD and others are specific like TRIO_U at CEA. For their use with sodium, they need to be validated on existing experiments from the 1980's. Core mechanics will rely on the HARMONIE tool that describes the static equilibrium of a core depending on limit conditions imposed at its boundary. The fuel behavior is simulated with the GERMINAL code. This tool was widely validated for a Phenix-like geometry. It will be extended to various future geometries and transferred into the PLEIADES platform, the French reference for nuclear fuel behavior simulation. GERMINAL will take advantage of the thermal-mechanics models yet available in PLEIADES. A special effort is intended on the long term to take into account and to validate the behavior of fuel loaded with large quantities of M.A.
For all the tools, the existing experimental basis will be revisited and a set of database will be developed.
On the medium term, coupling methods between the various physics will be introduced to simulate more precisely the core transient behavior.

4. Resistance to severe accidents and external hazards

4.1 Safety approach

Generation IV systems require an enhanced safety. Globally, the SFR safety will be of the same level as the one of the third generation LWRs. The EPR is taken as a reference and its general objectives are already very ambitious and guarantee a very high level of protection to persons and the environment. The defence-in-depth method is adopted as the basic principle to cover the risks and uncertainties inherent in this concept. Additional requirements provide both a real and demonstrable benefit and a greater degree of assurance in the safety demonstration and therefore in its robustness. Four topics are studied:
- Allowance for degraded situations and the "practical elimination" approach,
- The robustness of the demonstration adapted to the system,
- Consideration of the specific aspects of the sodium-cooled system,
- Minimization of impacts concerning radiological protection and the environment (discharges, wastes, dismantling actions).
A complete overview of this approach is given in [2].

4.2 Scenarios and transients studies

The innovative design that are envisaged may lead to specific scenarios, somewhat different from the one studied before in SPX or EFR. For instance, the fourth category transients are different whether you consider a loop-type or pool-type reactor. The presence of a power conversion system driven by gas will also induce new studies (impact of large quantities of gas under pressure in case of suppression of the intermediate circuit for instance). A special emphasis will be put too on the long term influence of M.A. in the sequence of events, a field of research quite new.

4.3 Sub-Assembly design for core melt-down management

Concerning severe accidents and core degradation, a threefold strategy must be implemented that includes <u>prevention</u> against melting initiators (hydraulics allowing a better mitigation of a local temperature raise, design of a boiling zone capable to favor natural convection and neutrons leakages…), enhancement of <u>protection</u> systems (passive 3rd level shutdown level), and at the end, in case of core melt, <u>provisions to minimize the consequences</u> especially the risk of significant energetic re-criticality. As a matter of fact, a major difference between LWRs and SFRs is the core reactivity. For a fast neutron spectrum,

a compaction of the core induces a reactivity step; that is not the case in a LWR. So a specific risk linked to hypothetical core degradation is the possibility to have a recriticality that can develop a mechanical energy release.
Globally, two routes will be studied, one consisting in the dispersion of the molten core by introducing some discharge channels either among the sub-assemblies or using neutron absorber channels, the other consisting in releasing absorber material in the molten fuel to reduce its reactivity; the two solutions being potentially combined. Fig. 3 describes a qualitative scenario of core melt-down up to corium recovery with the objective to significantly reduce re-criticality risks or their consequences.

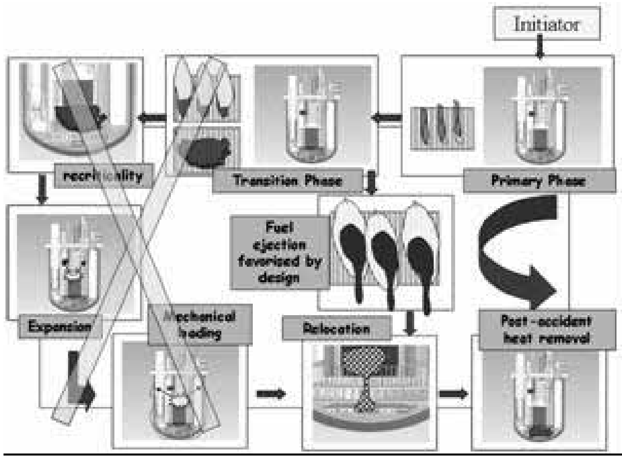

Fig 2. Severe accident sequence of events

Finally, to analyze the various modes of degradation, several initiators will be taken into account to study the possible degradation of the core. This item contributes to the robustness of the demonstration. One should try to cover a wide range of phenomenology to describe the mode of degradation of the core.

4.4 Core-catcher studies

Likely they will have to cope with a reactor vessel more compact than in previous projects, while ensuring still the decay heat removal and non-criticality. The design (shape, but also location: in or ex-vessel), the materials, the modeling of the debris, are the main R&D topics to be addressed.

4.5 Strengthened systems for defense in depth

In addition to provisions mentioned for an upgraded monitoring of the core, it will be necessary:
- to exclude by design scenarios such as the ingress of a large gas bubble in the core, a catastrophic failure of the core support structures, a compaction of the core,
- to enhance the diversification of decay heat removal systems, either from the point of view of their location, the physical principles used, the architecture of the plant and of its confinement,
- to reinforce provisions against leakages and fires, reactions of sodium with fluid used for energy conversion,
- to protect the plant against upgraded external aggressions such as earthquakes, new plane crash hypothesis.

4.6 Accident modeling

Accident modeling tools will be re-considered owing to their capacity to deal with the needs yielded by innovations selected for future SFRs. The basis for severe accidents involving very complex multi-physics aspects, will be treated with SAS4A and SIMMER (with a refined pin model called DPIN) codes; the so-called CATHARE CEA's code, currently being adapted for sodium applications, will be used for transient calculations and join SAS/SIMMER for the primary phase including boiling but prior to the loss of geometry.
So-called MC3D and PLEXUS CEA's codes will be available respectively for corium-coolant interaction and dynamic mechanical loads of structures. Debris beads behavior is covered by LIDEB and MC3D for some aspects.

Sodium fires will be addressed with FEUMIX and PULSAR (spray type fires).
Transfer of species (including radiotoxics) in the reactor building and releases will be assessed with CONTAIN code.
It is considered at this stage that the qualification of these tools can rely on the extended existing data bases, especially the numerous experiences using simulants of fuel and coolant, and experiences in representative situations (sodium, nuclear heat and fuel) CABRI and SCARABEE [3] , as long as oxide fuel is concerned. This does not exclude that studies to come induce some new needs, but they are not identified at now.
The situation is very different for cores that could use dense fuels such as carbide; if promoted in the frame of future industrial commercial units, dedicated programs shall be requested timely.

5. Looking for an optimized PCS to reduce sodium risk
5.1 A gas power conversion system

The main incentive for such an innovative option is to delete the risk of sodium water reaction and its potential consequences. Notice also that such an option opens, in the case of a loop type reactor, the possibility to suppress the intermediate sodium loop, and by the way to decrease the investment cost (see Fig 3.).

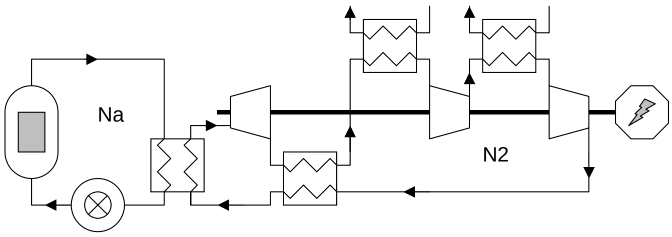

Fig 3. Typical layout of a loop type SFR, without intermediate sodium loop and coupled (via an IHX Na-gas) to a nitrogen Brayton PCS (one turbine, two compressors on the same shaft, high power heat recuperator)

Nevertheless at a given core outlet temperature, classical gases (such as nitrogen, or argon, eventually mixed with some amount of helium) will require a significant effort to compete with the Rankine water/steam cycle efficiency. These efforts can be made on the pressure level, on the improvement of the (indirect) Brayton cycle through optimized and enhanced components, use of re-heating by sodium, and/or by raising the temperature level (at the core outlet). An alternative for recovering an attractive efficiency (higher than 40% -Super Phenix value-) without temperature increase could be to use supercritical CO_2. This requires developing the necessary innovative technologies concerning components and materials.
The corresponding studies aiming in a first step at establishing the feasibility are:
- for Super critical - CO_2: the cycle stability (including in load follow-up hypothesis), the components deasibility (especially turbine, compressors) and the sodium- CO_2 interaction through dedicated tests,
- for all gases: the thermodynamical optimization and associated “hot” temperature level, the protection provisions: detection of leakages, dedicated phases separator component, valves for insulation and decompression, possibility of a “short” intermediate loop between gas and primary sodium, the safety analysis versus the risk of massive gas ingress in core, the prospect about materials (compatibility with fluids and with required temperature level) and the preliminary studies of IHXs: heat recuperator, Na-gas IHX (including sodium plugging hazard).
Good trends on this step would then allow undertaking heavier developments involving especially Na-gas IHX test at the scale of ~1MW exchanged, prior to larger ones if the option is definitely confirmed.

5.2 Optimization of materials choice according to temperatures level

Independently of the temperature level, future SFRs must be able to sustain a significant enhancement of their lifetime, up to 60 years, for those components that will not be replaceable. For coping with this requirement, feedback from Phenix reactor (after its shutdown foreseen in 2009) will be used as it includes an interesting panel of steels either austenitic and ferritic, representative of relevant families and aged for a long time in representative conditions.
For the non replaceable structures in the primary vessel, it is believed that hot and cold parts can be kept made out of the reference austenitic steel (Super-Phenix, EFR): Cr17-Ni12-Mo-Mn-(N). Nevertheless, in case of a temperature increase (possibly required for instance by efficiency concerns with gas PCS), by +50°C (i.e. 550 to 600°C), austenitic Cr25-20, Ni30-20 will be assessed for hot parts, with a special emphasis on creep performance and weld-ability. A more ambitious

increment: (+100°C), if decided, will require a long term program on nickel based alloys. In both cases "corrosion" by sodium is a concern owing to Ni dissolution enhancement by temperature that will be checked by dedicated sodium tests.
For other components such as heat exchangers, piping, austenitic steels could be challenged by ferritic-martensitic ones. Such a choice can be justified by mechanical properties (creep resistance), but also costs concerns as thermal properties (heat conductivity, thermal expansion coefficient), could allow for a lesser level of thermal induced stresses and a reduction of masses involved. The program includes the definition and optimization of a specific ferritic/martensitic grade within the range 9 to 12 chromium and to assess its attractiveness component by component.
Conversely, a specific action will be devoted to evaluate the profit expectable from a limited (20°C) drop of the hot temperature of the cycle in terms of ageing of base and weld materials.
At the end it is worth to mention that the outcomes of these researches will be implemented (provisions for procuring, material data, mechanical analysis methods, construction and inspection) in the code and standards dedicated to fast reactors (so called "RCC-MR"). A new release of this guide (undertaken for Super Phenix, enriched for EFR studies) is foreseen this year 2007.

5.3 An enhanced Steam Generator energy conversion system

A first objective is to mitigate the risk of sodium-water reaction and its consequences; this yields a first set of actions:
-reinforce reliability by technologies such as double walled exchange tubes, modularity etc…
-assess the viability of keeping a "compact" secondary sodium loop (up to set, inside a same vessel, a SG and an IHX units, thermally coupled by a very limited amount of sodium, or by an alternative coupling fluid)
-assess the possibility of replacing secondary sodium by another fluid compatible with water and sodium. With that view different metals mixtures and some salts are envisaged. They will be tested in terms of chemical stability (for salts), reactivity with sodium, physical nature of reaction products, and corrosion of materials and provisions that could allow for its control.
A second objective is to enhance the performances: with that view supercritical water cycle will be studied. It is worth to mention that such a cycle could allow for increasing performances by increasing the pressure (from 180bars for previous SFRs to 250 for instance, allow 2% efficiency earning) but a temperature increase is also to be considered (and will yield the same type of materials concerns as for gas PCS and already presented, § IV.B). Nevertheless supercritical water rises specific corrosion problems, that can be solved by use of nickel based alloys such as nickel based alloy 690 (to be checked also in sodium environment)

6. Reactor design re-examination

6.1 Reactor primary system

The previous paragraphs aimed at propose and evaluate innovations. The question is at last to see how long these innovations can participate to coherent reactor layouts, in the frame of integration studies, and to check these layouts against the high level goals with dedicated tools (economy, safety..).
As to the primary system, a lot has been done in Europe, and especially in France about the so-called integrated primary (pool type) system. This system provides a robust design of the primary confinement, against loss of primary sodium (and by the way primary sodium fires), against loss of the primary hydraulic loop, ensures a high thermal inertia and guaranties a good natural circulation in the main vessel. The cold plenum contributes to the mitigation of thermal shocks and gas transport. It is also favorable to alleviate radioprotection concerns during operation and allows designing easily a hydraulic path for cooling down the main vessel. As identified drawbacks, it is worth mentioning the difficulties to achieve a compact reactor block, to have an easy access to internal structures for monitoring and repair; it implies in vessel rotating components and earthquakes effects are complex because involving strong fluid-structure interaction.
Loop type system has the important potential to make easier the intermediate heat transport loop suppress. It can offer some easier maintenance and repair conditions for large components that are separated from the reactor tank, and can be integrated in a single component. There are no rotating parts in the reactor vessel and there is a potential for more compact components (main vessel). This design is likely more easy to justify vs. earthquakes because less sensitive to sloshing effects. Drawbacks concern the risks associated to the loss of a primary loop (fire, leak, flow reversal, gas entrainment), lower thermal inertia and risk of gas transportation. Keeping the main vessel below the creep regime is not easily achievable. Operation conditions can be made more difficult owing to active, double walled, primary sodium transport piping. The program will consider both options and will address the different topics mentioned just before through integration studies.

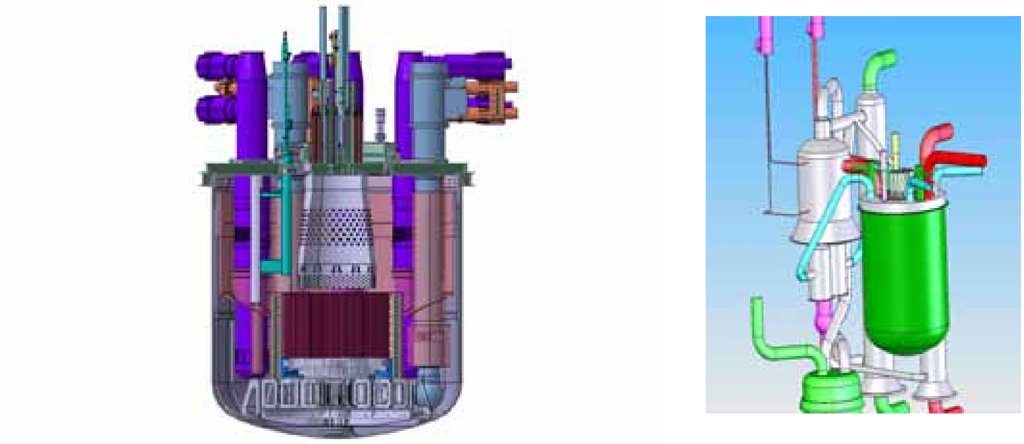

Fig 4. Optimized pool type (left), and sketch of an optimization of loop type (right) primary circuits

Beyond this comparison pool vs loop, size effects will be dealt with especially in order to assess possible threshold effects than could incite to consider limited power output plants. When precise enough, and with the input from other systems and components studies, the designs will be compared from the points of view of economics (SEMER code), safety, inspection and repair, availability.

6.2 Intermediate system optimization

For those reactor layouts using an intermediary loop, the target for this last will be to reduce the cost (including for maintenance and manufacturing processes), looking for compactness (using ferritic-martensitic materials) and reduction or simplification of the number of components and auxiliary circuits. Integrated components, short loops, improvement and simplification of provisions against sodium leaks (including inert gas filled casemates) are the tracks foreseen to be followed.

6.3 Components & systems optimization

As for auxiliary systems, the sodium purity target and technologies for traps will be reconsidered. Sodium quality control will be adapted looking for direct measurement of impurities content (like O, H, C) in addition to conventional plugging temperature. Tritium management will need a particular attention to the regeneration of traps; in case of use of a gas PCS (as no hydrogen will be injected in sodium by reduction of water on the SG tube wall), a specific strategy is to be imagined. Cover gas treatment either at the input (removal of impurities) and at the ouput (gaseous FPs), is also subjected to improvement, so is the treatment of aerosols in gas volumes above the sodium.

6.4 Fast fuel handling

This point is very important as it has a key contribution to the availability of the plant, and can be determining versus the design of the primary system:
(1) its geometry, as it must in any case allow for access to all subassemblies and as room for in-vessel storage of used fuel is necessary according to the option chosen,
(2) its efficiency: how fast used subassemblies can be removed out of the core and replaced by fresh ones. This is particularly important if in reactor vessel interim storage is not the option chosen, but it could be also a safety concern for instance if an inspection of the core support is needed or following an accident.
(3) The design of the handling system can even concern the layout of the complete plant, in the case when a modular architecture appears to be attractive: as matter of fact the ex-vessel equipments, and especially any interim storage tank and washing facility, can be shared by different modules

Fig 5. Schematic sketch of a modular four units reactor with unique washing facility and interim in sodium storage tank

R&D actions in the program are aimed at the following:
- study three options for in-vessel fuel handling that are not indifferently applicable to the options for the primary system:
    1. optimization of existing system: two rotating plugs and one interim put down-take over position, with improvements concerning efficiency and high-power used SA discharge.
    2. assess one rotating plug plus pantograph solution
    3. assess direct handling using a dedicated hood.
- study transfer and washing of the used fuel subassemblies at a power ranging in between 10 and 25 KW.

6.5 Enhanced ISIR

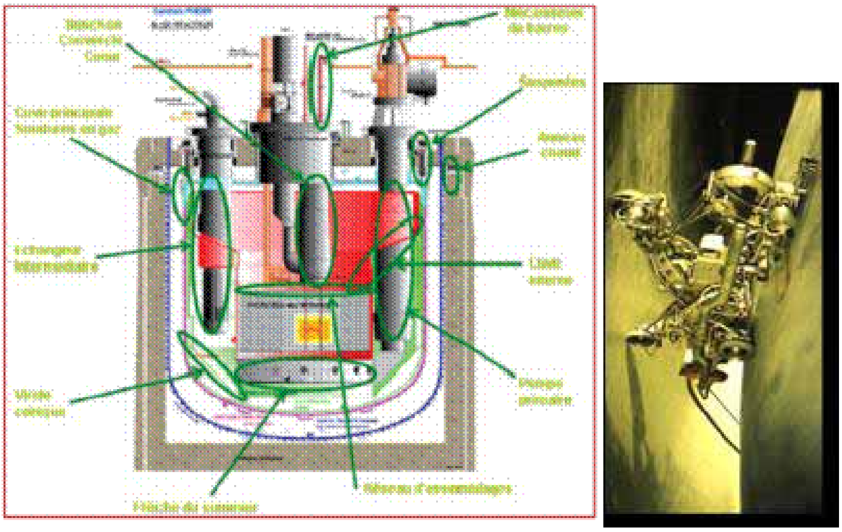

Fig 6. Parts in Phenix reactor that have already been inspected with different technologies (left) and “MIR” device developed for the volumic inspection of the welds on the primary vessel of Super-Phenix (right)

Beyond these performances, for future SFRs, possibilities of in-service inspection and repair have to be clearly enhanced again. This will be made first by considering inspection strategies at the design stage and from the point of view of the criticality of each component or sites on this component. Strategies of access will be chosen in coherence: geometrical considerations and hatches, necessity or not to be able to empty the primary sodium, or part of it.
Developments will be pursued on under-sodium ultrasonics technologies:
For monitoring systems used during reactor operation, a key point is to define a piezzo-electric material suitable for the high temperature of the hot pool/legs in the reactor.
For periodic examination, classically made at “cold” shutdown conditions, a key point is to enhance the quality of transmission of the US generated by the transducer to the fluid, and back from the target to the transducer.

At the same time the modelling of US propagation and reflection will be developed in order to help the optimization of dipped transducers technologies. Mono- and multi-elements will be developed as well, depending of the application (telemetry, far viewing, close viewing, volumic NDT against small or large defects).
Development of distant US Technologies, allowing to check a structure from its external side, using it as a wave guide (already employed for examination of the Phenix distant welds on the conical shell supporting the core) will be pursued.

7. Conclusions

The program presented above, will be organized with regard to two short term milestones: 2009 and 2012. The first period is dedicated to propose and study innovations that will be integrated in very preliminary sketches. The 2009 milestone will be the opportunity to select promising orientations. Between 2009 and 2012, selected technologies will be studied in depth, and an integration work towards one reactor layout and one backup will be performed. Each milestone with be also the opportunity to address the question of the prototype that will have the aim to feature the technologies of the power plant of the future as far as possible. In 2012, the main specifications of this reactor are to be fixed and will take into account its mission regarding demonstrations on the fuel cycle. The facilities for its own cycle will also have to be defined, with regard to the first core, and to possibilities of cycle experiments at the scale of some sub-assemblies.

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Session 18.1.2:

Status of future concepts

CURRENT STATUS OF JAPANESE SODIUM COOLED LOOP TYPE FAST REACTOR (JSFR) DEVELOPMENT

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ABSTRACT

A new project, Fast Reactor Cycle Technology Development Project (FaCT Project) was launched in last autumn in Japan. In the project, conceptual design study on Japanese Sodium cooled loop type Fast Reactor (JSFR) which was selected as the most promising concept for next commercialized reactor in previous study and research and development of innovative technologies adopted in the concept are implemented toward an important milestone at 2015. In order to satisfy the high design requirements, several innovative technologies were identified and included in the current design. They are categorized into three areas: for economic competitiveness, enhancement of reliability, and enhancement of safety. This paper describes the current status of the project, especially on the targets and the design requirements for JSFR as well as some related innovative technologies development, integrated IHX/Pump, thermo-hydraulic optimization in compacted reactor vessel and experimental study on FBR core-disruption accidents aiming at establishment of advanced safety logic, "elimination of severe re-criticality events".

1. Introduction

The first stage of development of commercialized fast reactor cycle systems in Japan was finalized in 2006. Following the results in which the sodium cooled loop type fast reactor (JSFR) with oxide fuel selected as the most promising reactor concept[1], a new project, Fast Reactor Cycle Technology Development Project (FaCT Project) was launched focusing on development of the selected concept.

In order to satisfy the high design requirements, several innovative technologies were identified (See Fig.1) and included in the current design. They are categorized into three areas: for economic competitiveness, enhancement of reliability, and enhancement of safety. Shortening of piping length by adoption of high chromium steel, 2-loop system, integration of IHX with primary pump (integrated IHX/Pump), compacted reactor vessel by hot-vessel concept are in the first category. SG with double-wall-tube is categorized into the second area. Passive shutdown system, re-criticality free core by special sub-assembly, and anti-seismic technology are in the last.

This paper describes the current status of JSFR in FaCT Project,

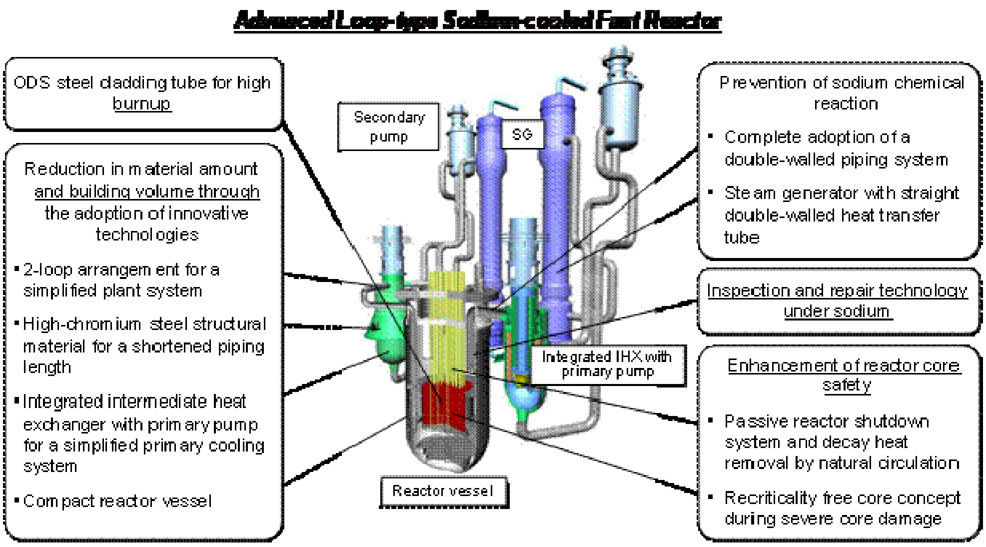

Fig 1. The features of the JSFR system.

especially on the design requirements, current design as well as some related innovative technologies development, integrated IHX/Pump, thermo-hydraulic optimization in compacted reactor vessel and experimental study on FBR core-disruption accidents aiming at establishment of advanced safety logic, "elimination of severe re-criticality events".

2. Development targets and design requirements of JSFR

In the FaCT project, R&D activities will be carried out under the development targets as summarized in Table 1. And the design requirements of JSFR shown in Table 2 are established in order to satisfy the development targets.

Category	Targets
Safety and Reliability	SR-1: Ensuring safety equal to or better than contemporary LWR(LWR) SR-2: Ensuring reliability equal to or better than LWR
Sustainability	
Environment Protection	EP-1: Radioactive influence through normal operation no more than LWR EP-2: Emission control of environment transfer substance which can restrict in safety limits
Waste Management	WM-1: Reduction of the amount of radioactive waste equal to LWR WM-2: Improvement of waste manageability equal to or better that LWR WM-3: Reduction of radio-toxicity equal to or better than LWR
Efficient Utilization of Nuclear Fuel Resources	UR-1: Breeding performance to enable transition to fast reactor, and its flexibility
Economic Competitiveness	EC-1: Electricity generation cost equal to or cheaper than the competing energy sources in the future EC-2: Investment risks no more than LWR EC-3: External costs no more than LWR
Nuclear Non-Proliferation	Np-1: Adoption of institutional measures and application of technical features which can enhance non-proliferation Np-2: System design of physical protection and its development to prevent theft of nuclear materials and sabotage

Tab 1: The development targets for FaCT Project.

Item	Requirement
Breeding Capability	Breeding ratio: ca. 1.2, System doubling time: ca. 30 years
TRU Burning	TRU burning under fast reactor Multi-recycle and long-term storage of LWR spent fuel (Transmutation of LLFP such as I-129, Tc-99 is desirable)
Radioactive Release	Equivalent or less than present LWR application
PR&PP	Excludes pure-Pu state throughout system flow
Safety	Operability, Maintenability, Repairability and Passive safety Re-criticality free, core damage frequency less than 10^{-6}/ry
Electricity Generation Cost	Cost-competitiveness with other means of electricity production and a variety of market conditions, including highly competitive deregulated or reformed markets
Operation Cycle	ca. 18 months, and more
Construction Duration	Large-scale: 42 months, Medium-scale modular type: 36 months

Tab 2: Major Design Requirements of JSFR System

3. Current status of some main innovative technologies development

3.1 Integrated IHX/primary pump[3]

In order to reduce manufacturing, building, and operation cost, the primary pump and IHX are integrated and put into one vessel as shown in Fig.2 in the JSFR design. A critical issue in this component design is the fretting of the IHX heat transfer tube with a baffle plate. At present, some experiments and numerical analyses are carried out to develop technologies for reduction of vibrations from the pump and a vibration transfer control system.

(1) 1/4 scaled model vibration transfer experiment with vibration oscillator and analysis: An inertial vibration oscillator was placed in a simulated pump casing of a test apparatus. Frequency response characteristics of the 1/4 scale model with and without water was revealed from experimental results. The 1/4 scale vibration test was numerically analyzed by FEM-code (FINAS) with a three-dimensional shell model. This code can directly handle the vibration behaviour of complex multi-cylinders with considering fluid-structure interaction effects. Fig.3 shows vibration responses at innermost shell structure and tube of IHX. The analytical results were in good agreement with the experiment, with 10% accuracy in eigenvalue. The results demonstrated that the numerical analysis by FINAS can be applied to the IHX design optimization. In the IHX analysis, it is evaluated that there is no resonant vibration mode at pump speed of 100% operation capacity, 44% (initial low power operation), and 10% (stand by) with a safety margin of 20%.

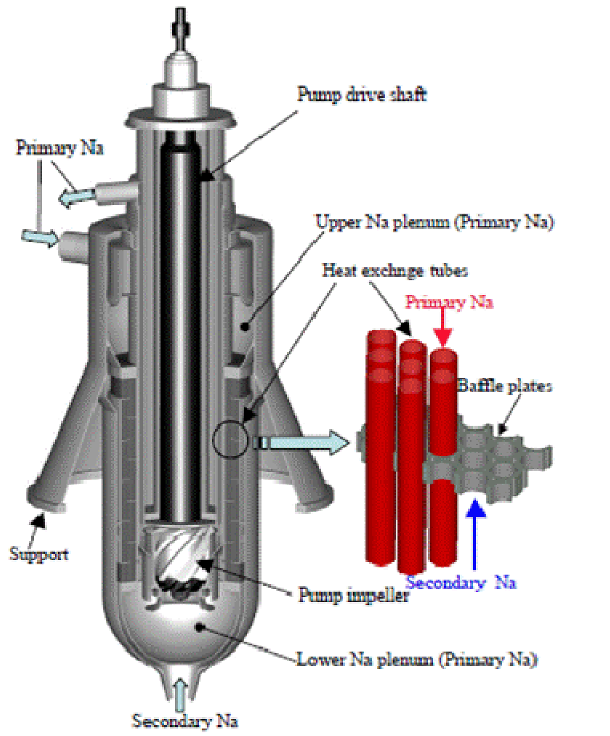

Fig.2 Integrated IHX/Primary pump

(2) 1/4 scaled model vibration transfer experiment with pump and analysis: Most unfavourable feature of this design is that the oscillation eigenvalue of pump casing is lower than the value of revolution per second (RPS) of pump, because the pump casing is slim and long to be installed in IHX. This design makes a vibration resonance between pump rotating vibration and casing eigenvalue in pump speedup operation. The results showed that pump speed and casing have a resonance and the shaft vibration has the peak value at around 1000rpm. However, the pump structure has enough attenuation and the vibration is lower than the limit.

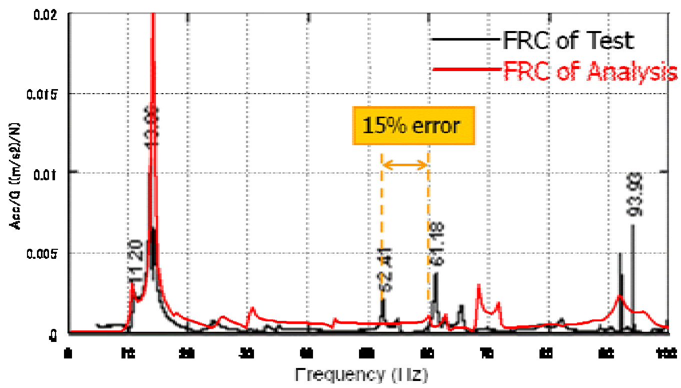

Fig.3 Frequency response curve at dividing wall of IHX/pump bottom

3.2 Thermal-hydraulic optimization in compacted reactor vessel[4]

Thermal stratification phenomena during the scram transient were investigated by using an 1/10th scaled model of the reactor vessel upper plenum, where water was used as working fluid. Balance between buoyancy force and inertia force, i.e., Ri number in the experiment was set equal to that in the designed reactor. The temperature difference between the initial hot temperature in the plenum and cold flow from the core after the scram was set at 25˚C. Then the flow velocity at the core outlet became only 1/10th of that in the designed reactor. Thus, Re number distortion is order of 100. The experimental study was carried out to see the influences of the column type UIS (Upper Inner Structure) with a slit where a subassembly was transported during a fuel exchange operation and also to find mechanism of characteristic phenomena. Configurations of components in the upper plenum, i.e., height of a cylindrical plug in front of the UIS slit and an outer cover of the UIS were also examined as a mitigation measure of thermal load during the thermal stratification.

(1) Experimental Setup: The 1/10th scaled model for the upper plenum of reactor vessel is shown in Fig.4. The left side figure shows the reference geometry. In the reactor design of JFSR, the column type UIS, which has the radial slit and no outer cover, is located at the centre of the upper plenum. Double dipped plates are set to reduce the flow velocity and avoid the gas entrainment at the free surface. Other

two cases in the right figures are the geometry parameters to investigate influences on the characteristics of thermal stratification phenomena and mitigation methods for the thermal stress.
Temperatures in the upper plenum were measured by some series of copper-constantan thermocouples at 10Hz sampling frequency. The outlet temperatures of several core and blanket fuel subassemblies were also measured at the centre of each outlet hole. The measurement error was less than 0.1°C.
(2)Results and Discussion: The temperature data in the reference case are analyzed to see the stratification phenomena. Vertical temperature distributions at inner position in the plenum near the UIS slit and the opposite DHX are shown in Fig. 5. The temperature is normalized by the temperature drop (ΔT) at the core outlet after the scram and the lower limit of core outlet temperature (Tc). Typical distribution in the thermal stratification can be seen in this figure. The maximum temperature gradient at the stratification interface near the UIS slit was larger than near the DHX. This thin interface layer is resulted from the impingement of the jet through the UIS slit at the stratification interface and local entrainment of the fluid at the bottom of the interface layer.
Two configurations were examined to find a mitigation measure. One is the FHM plug, which is inserted to the lower position in the upper plenum and the other is a perforated outer shell of the UIS. The FHM plug is designed to change direction of the jet through the UIS slit. The UIS outer shell guides the cold fluid exiting from the core to flow upward. It will help better mixing in the upper plenum. The temperatures in these cases were compared to see the influences of the modified structures on the thermal stratification interface. The vertical temperature distributions at the slit and DHX sides are shown in Fig. 6 at t =1200s from the scram. In the slit side, both the FHM plug case and the UIS outer shell case showed that the temperature gradients at the thermal stratification interfaces were smaller than that in the reference case. The FHM plug case has advantage of a minor impact on the in-vessel components design.

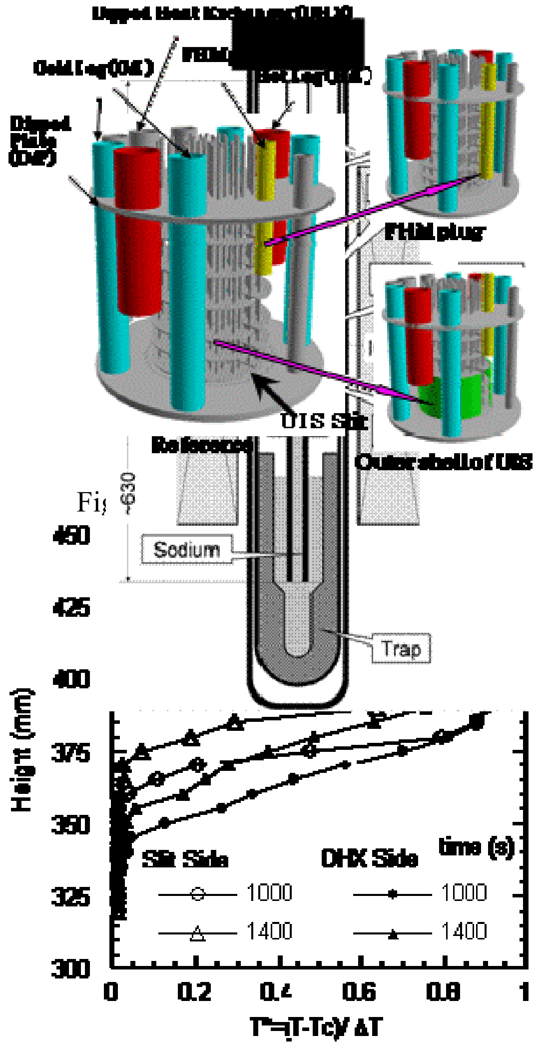

Fig.5 Vertical temperature distributions near the UIS slit and DHX in reference case

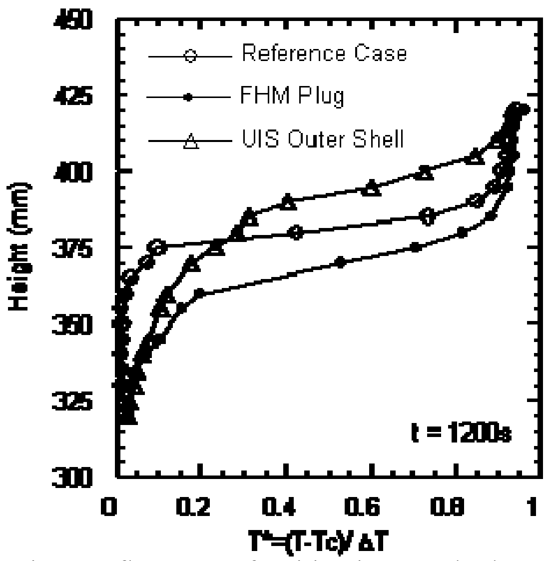

Fig.6 Influences of mitigation methods on vertical temperature distributions near the UIS Slit

3.3 Re-criticality free core[5]

Several in-pile and out-of-pile tests were conducted under a co-operation between JAEA and National Nuclear Centre of Republic of Kazakhstan (EAGLE program). One of the main objectives of these tests was demonstration of effectiveness of special FBR design concepts to eliminate the re-criticality issue in the course of core disruption accidents. Figure 7 shows schematic of a typical in-pile test apparatus of the EAGLE program. The geometry of this test apparatus is corresponding to a typical special design concept equipped with a “discharge duct” within each fuel sub-assembly. The discharge duct of 2mm-thick stainless steel filled with liquid sodium was placed at the central part, and was surrounded by 75 UO2-fuel pins with 400mm fissile height giving total fuel amount of ~8 kg. The test ID1 (Integral Demonstration test 1) was conducted with this test apparatus in IGR (Impulse Graphite Reactor). It was intended to produce a molten fuel-steel-mixture pool with the trapezoidal power diagram simulating the hottest part of the degraded core in a ULOF accident. Thermocouples placed within the fuel-pin bundle region suggested molten-pool formation from ~27.4 second on. The duct failure took place at ~28.7 second, about one second after start of the strong duct heating by the pool, and it was followed immediately by a rapid sodium void development toward the bottom as shown in Fig. 8. Through this developed void, an effective mixture discharge lasting for about one second took place. This result showed a significant potential of core-material relocation even under a relatively low pressure difference (up to 0.12MPa). Although post-test examinations to quantify final material distribution are still to be performed for this test, following preliminary conclusions have been drawn out through data analysis of all the in-pile and out-of-pile tests.

(a) Sodium-filled duct wall fails at an early phase of fuel-steel mixture pool formation corresponding to the hottest core part.

(b) Mixture discharge takes place in a short time range provided that fuel enthalpy is high enough and a meaningful pressure difference is maintained.

(c) Relocated mixture can be quenched forming debris as far as sufficient amount of coolant is available.

4. Conclusion

The targets and the design requirements for development of advanced sodium cooled loop type fast breeder reactor (JSFR) in FaCT -project are briefly discussed in this paper. And the present status of three main R&D issues is reported. Preliminary result of each R&D is summarized as follows,

(1) As for integrated IHX/pump, the proper design is to be realized based on both experimental and numerical studies and will be applicable to JSFR.

(2) The experimental results showed that the stratification interface had steep temperature distribution near the UIS slit due to the impingement of the jet through the slit. This steep temperature gradient at the stratification interface can be mitigated greatly in a case where the FHM plug was located at the lower position near the core top in the upper plenum.

(3) Concerning re-criticality free core, so far main outcomes are encouraging the present computer-code simulations for the ULOF accident predicting that the reference FaCT design concept is free from the re-criticality issue.

Fig.7 Typical EAGLE in-pile test apparatus

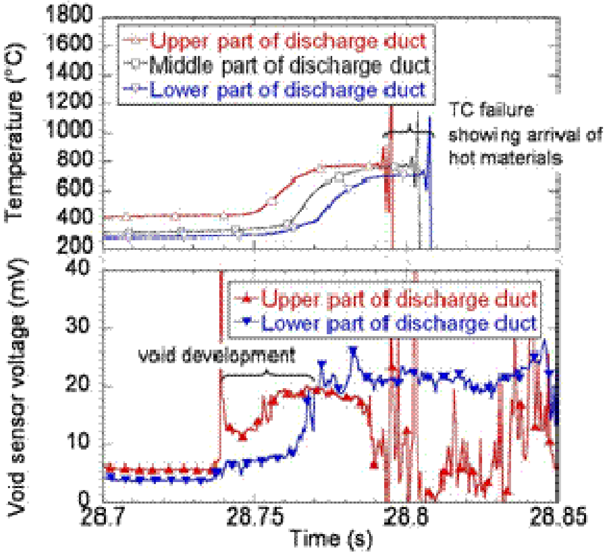

Fig.8 Response of TCs and void sensors in the discharge duct in the ID1 Test

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THE (EUROPEAN) HTR TECHNOLOGY NETWORK (HTR-TN) AND THE DEVELOPMENT OF HTR TECHNOLOGY IN EUROPE

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ABSTRACT

Mastering global warming risks and securing the European energy supply cannot be obtained only by developing CO_2 free electricity generation, as electricity represents only a limited part of energy consumption, while most of the remainder is provided by fossil fuel burning. High Temperature Reactors (HTR) can contribute to reduce CO_2 emissions by supplying heat needed by many industrial processes. Due to promising market prospects and attractive safety features, several industrial HTR prototype projects emerged in the world during the last decade. Europe, presently leader in nuclear energy, should maintain its rank in this race for a new frontier in nuclear energy. HTR-TN, created in 2000 for building a coherent partnership for HTR development in Europe, proposes a roadmap for the emergence of a new generation of reactors addressing heat needs of European industry. A first step should be a worldwide first-of-the-kind demonstration of the coupling of a HTR with an industrial process heat application. HTR-TN initiated the development of advanced generic HTR technology in the 5th and 6th Framework Programmes, with already important results. Beyond continuation of generic R&D, the large components of the reactor, the coupling system and the application part of the demonstrator, which are beyond state-of-the-art, should be developed and qualified during the 7th Framework Programme, requiring the development of an infrastructure of large specific test facilities.

1. Introduction

Europe has been a leader in High Temperature Reactors (HTR) development from the 60s' to the 80s' (DRAGON, AVR, THTR reactors have been constructed and operated). After a break due to the nuclear phase out in the leading country, Germany, the HTR development restarted with a small project, INNOHTR, funded by EURATOM in the 4th Framework Programme (FP4), followed by a cluster of 10 coordinated projects in the 5th Framework Programme (FP5) and by a large Integrated Project, RAPHAEL, in the 6th Framework Programme (FP6) with the additional contribution of a Specific Targeted Research Project dedicated to the study of the potential of HTR to burn actinides, PUMA. As the development of a new type of nuclear system requires a large and long term effort far beyond the time period of a single Framework Programme (up to two decades), the partners of the European HTR development programme founded in 2000 the (European) HTR Technology Network (HTR-TN) in order to elaborate a long term R&D strategy and to organise a stable partnership for implementing it. Significant results have already been obtained and more are expected in the coming years. But for the time being, the HTR European programme was only dedicated to the consolidation of generic high temperature technologies and to the exploration of advanced solutions for improving the performances of future HTR (the VHTR objective). Now the time is coming for a new step forward towards the development of industrial high temperature systems in Europe, as it is already the case in other parts of the world.

Therefore in the next Framework Programme (FP7), on top of the continuation of programmes dedicated to the development of generic technologies, a support from the European programme to the development of a HTR prototype of industrial size and of its components should be considered, and the large test facilities required for the qualification of these components should be built.

2. The specific role of HTR/VHTR in the future fleets of nuclear reactors

In the context of political uncertainties on the access to fossil fuel resources, of long term worldwide depletion of these resources and of global warming risk, nuclear energy can play an important role for securing energy supply of Europe during the 21st Century. But at present, nuclear fission is dedicated almost exclusively to electricity generation, which, however, accounts for only 16% of the energy consumed in the world, 79% of the remaining energy consumption coming from fossil fuel burning [1]. Therefore, in order to contribute significantly to reduce the dependence on fossil fuel supply and to master CO_2 emissions, beyond electricity generation, nuclear energy should also address a significant part of the rest of the energy market. As nuclear plants are producing large quantities of heat that cannot be transported on long distances if they are not converted into electricity, the non-electricity target for nuclear energy should be the industrial processes that consume large quantities of heat in a limited area. Nevertheless as some of the industrial processes that can be considered are meant at improving the extraction of fossil fuel or producing synthetic fuel (extracting oil from tar sands, lightening of heavy oil, coal to liquid transformation or hydrogen production), nuclear reactors could also indirectly reach scattered energy uses, in particular for transport.

Most of the industrial process heat applications require much higher temperatures than the operating temperatures of present Light Water Reactors (LWR) or of the future liquid metal fast reactors, which are therefore doomed to be mainly stuck to electricity generation. Moreover the quantity of energy required is never more than a few hundred megawatts, while most of the present or future systems become competitive only for a thermal production of several thousand megawatts. Because it produces heat at high or very high temperature using a smaller reactor with a very robust safety concept, the modular HTR/VHTR systems have the potential to address a wide range of industrial process heat applications that cannot be addressed by LWR or liquid metal fast reactors, which does not exclude medium-sized electricity generation, or process heat and electricity cogeneration. Therefore it is clear that HTR/VHTR should be included in future nuclear reactor fleets not in competition with other types of nuclear systems, but in complement to them mainly for addressing the specific mission of providing high temperature heat for industrial processes.

3. Possible approach for entering HTR on the industrial process heat market

The result of a survey of present industrial processes, which require at least 100 MW of heat within a single industrial site, is shown in figure 1 [2]. It appears that:

- Potentially the HTR/VHTR heat market is far from being a niche market: such reactors could address heat needs of many industrial processes. Therefore the future of HTR/VHTR is not only depending on the possible long term development of a hypothetical “hydrogen civilisation”, but on the incentives (increasing costs, CO_2 tax, security of supply…) that present industries will have in the short/medium term, to switch from heat supply by fossil fuel burning to alternative supplies.
- There is a first group of processes requiring temperatures below 600°C, mainly for steam production, and a second group above 900°C, with practically no need between 600 and 900°C. For the first group, the heat could be provided by a HTR using existing industrial materials and proven TRISO HTR fuel. On the contrary, the materials and the fuel required for the VHTR, which would produce the heat needed by the second type of applications, are still to be developed. Therefore applications at very high temperature (> 900°C) could not be considered but in the longer term.

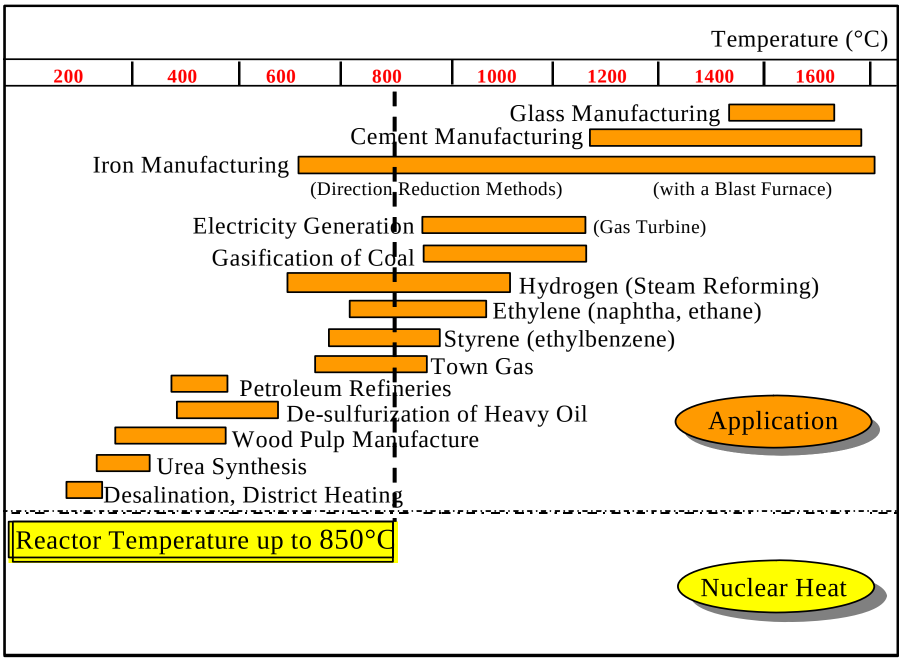


Figure 1: Present heat intensive industrial processes

Therefore, in order to minimise risks, HTR/VHTR penetration on the industrial process heat market should be undertaken step by step.

First a full industrial scale demonstration of the feasibility of the coupling between an HTR and a process heat application should be obtained as soon as possible at a reasonable temperature level. As nuclear energy has only been used at industrial scale for electricity generation, such a first-of-the-kind demonstration, even at a moderate operating temperature, is already quite a major challenge. It is necessary in order to verify that a nuclear reactor can actually be connected to an industrial process and face its requirements and hazards, which will certainly be quite different from those of utilities for electricity generation. It is also necessary in order to demonstrate to heat intensive industries, used to fully integrate fossil fuel burning heat supply in their processes (in particular industries processing fossil fuels), that resorting to nuclear energy could not only be feasible, but also beneficial for them in terms of economic competitiveness, CO_2 emissions and sparing of natural resources. Taking into account the time necessary for R&D and qualification work, procurement of components, licensing and construction, the prototype demonstration could not be operated before the end of next decade.

Adding to the challenging objectives of the first demonstrator the very high temperature target would drastically increase the risks and the length of development. But, while the first demonstrator will provide – at a reasonable temperature level – the first experience feedback from coupling a nuclear heat source with an industrial process, which is presently missing, the R&D for higher performance materials and fuel will be continued and will expectedly produce results allowing defining a credible design for VHTR. Nevertheless entering into this new phase, market needs for such a reactor should be reassessed, because they could have changed from those shown in figure 1: industrial processes usually have a lifetime of one or two decades and presently the trend is to search for lower energy consumption and lower temperature processes.

4. The legacy from previous European projects

Beyond the legacy from former European HTR achievements, the large effort made in FP5 and FP6, as well as the complementary national programmes, led to significant results concerning generic aspects of the HTR technologies. These results have already been set out in different papers (see for instance [3] and [4]) and therefore only the main achievements will be recalled here:

- A steel needed for operating the vessel at higher temperature than with PWR vessel steel (limited to 350°C), modified 9Cr1Mo, has been selected and the main elements of feasibility for using it for a

HTR vessel have been obtained (weldability of thick plates, no significant impact of irradiation on mechanical properties, at least 50°C increase in negligible creep limit compared to PWR steel…).

- The use of nickel base alloys, which are the best candidates within existing industrial materials in terms of high temperature mechanical and corrosion performances, for application to the IHX, will not allow operating the demonstrator above 800-900°C (depending on IHX design).
- Large varieties of graphite grade samples have been characterised, tested for corrosion resistance, and are presently irradiated at 750°C and 950°C beyond the turn around point. The data already obtained will allow selecting the most appropriate grades for HTR graphite core application.
- Conceptual designs have been obtained for the main primary components (in particular different plate IHX concepts have been examined), but only few testing of these components could be performed (for instance for the IHX in a helium loop (figure 3), the hot gas duct thermal barrier (figure 4), the circulator magnetic bearings (figure 5)) due to the need of large dedicated test loops which are not existing yet.

Figure 3: HE-FUS3 helium loop, IHX mock up testing, ENEA Brasimone (Italy)

Figure 4: HETIMO test bench, thermal barrier testing, CEA Cadarache (France)

Figure 5: FLP 500, facility for testing the dynamics of a shaft supported by magnetic bearings, IPM Zittau (Germany)

Figure 6: The NACOK air ingress integral loop, FZJ

- The bases for fabrication of HTR fuel have been recovered and a laboratory scale facility for manufacturing HTR TRISO fuel particles and fuel elements (compacts) has been commissioned in France. Methods for quality control for fuel fabrication are developed.
- The performance of state-of-the-art HTR fuel (recovered from the best German former fabrications) at very high operating temperature (up to 1250°C for the fuel particle) and burn-up (up to 15-20% FIMA) are being explored in two irradiations, HFR-EU1 and HFR-EU1bis.
- The main calculation tools required for HTR design and licensing exist or are being presently developed by different European organisations. A joint qualification effort has been undertaken in FP5 and FP6 through benchmarks and acquisition of new experimental data (fuel irradiation to very high burn-up, PIE and heat-up tests for fuel performance modelling and irradiation of fuel coating material samples for elaborating laws of evolution of coating material properties under irradiation, isotopic analysis of very high burn-up fuel for fuel depletion calculation, recovery of operating data of the EVO 50 MW helium Brayton cycle loop for system transient calculation, NACOK loop test (figure 6) for calculation of air ingress (coupling of natural convection and graphite oxidation)).
- Long term leach tests of irradiated HTR fuel in geological disposal conditions have been launched. Preliminary results, to be confirmed in the continuation of these tests, already show that TRISO particles keep their unique robustness in final disposal conditions and that their lifetime in such

conditions should be at least 10 000 years. On the other hand a preliminary survey allowed identifying promising paths for separating fuel from graphite and for managing separately both waste streams in such a way that the volume of ultimate wastes to be disposed off is minimised.

5. A European roadmap for the development of a HTR demonstrator for industrial heat application

As shown in Part 3, the development of HTR/VHTR systems should open a new frontier to nuclear energy, allowing it not only addressing electricity needs, but also other types of energy needs, which represent the largest part of the energy consumption. The European Union, faced to the challenges of securing its energy supply and of mastering its CO_2 emissions, should not miss such opportunity.

The development of HTR industrial prototypes already started in several leading nuclear countries, with projects of industrial prototypes: NGNP in the USA, the PBMR in South Africa, HTR-PM in China, GT-HTR 300 in Japan and NHDD in Korea. Europe, which has been until now a worldwide leader in nuclear energy, but which limited its HTR developments to generic R&D, should not be absent from this race for mastering nuclear high temperature technologies and should enter as soon as possible in the development of a HTR prototype coupled to an industrial process heat application, endeavouring to embed this project in an international cooperation framework. But the progress in design and R&D should not wait for the development of such an international partnership framework, which will take time and should be an important task for a future FP7 HTR project.

Developing a partnership with industrial process heat end users is also an essential task: it is not because the HTR produces high temperature heat that the adequacy with industrial end users' needs is assured. Industrial requirements for heat production must be defined in an interactive way between the nuclear reactor designer and the industrial heat end user. The reactor designer will adapt his design to industrial requirements, and the industrial end user will also have to tune his process for integrating more easily the nuclear heat source into the optimised scheme of its plant. An important part of the programme should therefore be dedicated to this evolution of the industrial process.

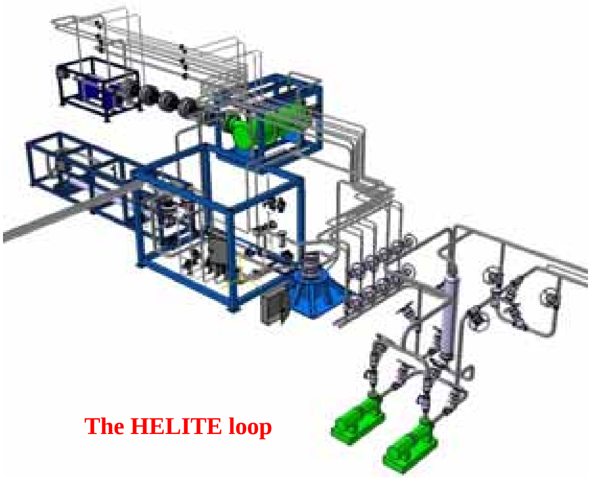
Happily such ambitious developments will not start from scratch: they will rely on the legacy of former European HTR projects and on the generic R&D programme started of FP5 and FP6. Nevertheless, there are still some basic R&D issues that have not been fully examined because of the length of the necessary work, or even not been addressed at all yet, because they did not get a top priority in the early phases development, even if they will be required for the industrial development of a new type of HTR/VHTR. FP7 should complete most of the remaining generic R&D tasks, even if some residual ones should be continued in the 8th Framework Programme (FP8):

- Even if important progress has been made in FP5 and FP6 for selecting and validating the materials needed by HTR/VHTR projects, the work have to be continued for completing qualification files of these materials, particularly when high fluence irradiation or low stress creep require long tests.
- Though important code qualification elements have been obtained during FP5 and FP6, the experimental databases required for certification of computer codes are far from being comprehensive. For example, in reactor physics, additional critical experiments are necessary for reducing uncertainties on neutron flux distribution; elements of validation of the thermal feedback on the power distribution obtained through coupled neutronic / thermo-fluid dynamics calculations could be found in some HTTR measurements; the "hot ASTRA" test just selected by ISTC will allow qualifying the calculation of the core temperature coefficient from room to the operating temperature, which is a key element of HTR safety demonstration. Additional fuel irradiation and safety tests will be required for improving fuel modelling and qualifying fuel performance codes. Due to the distinctive behaviour of HTR materials (e.g. graphite) or to particular features of HTR design (e.g. seismic behaviour of a stack of hexahedral blocs), there are HTR specific issues in mechanical modelling that must be addressed. The thermo-fluid dynamics modelling of some critical zones (e.g. mixing of hot and cold helium in the lower reactor vessel plenum) will have to be qualified. Depending on the conclusions of RAPHAEL, additional qualification work might have to be performed on the transient system analysis codes, including perhaps some tests performed on gas loops in configurations representative of the reactor design.

- The irradiated fuel leach tests, started in FP5, presently continued in FP6 are still to be extended in FP7 in order to reduce the uncertainty in predictions. Moreover the management of irradiated graphite, only touched in a preliminary way in FP5 and absent in FP6, will have to be studied in order to develop solutions in this area which will be critical for HTR acceptability.
- For having a chance to implement the technologies developed in the European programme in an actual industrial reactor prototype, some technological developments neglected until now will have to be addressed. This is the case for the development of an instrumentation adapted to the operating conditions of HTR/VHTR: while, until now, the in-reactor instrumentation of this type of reactor has always been very limited, it is desirable to acquire more comprehensive operation data in a future prototype in order to check the performance of materials and components and to be authorised later to operate industrial reactors at a high performance level. HTR head end fuel reprocessing and recycling technologies (e.g. technologies for breaking the coating of the irradiated particles and recovering the kernels, as well as the development of manufacturing processes for actinide fuel and the testing of the behaviour of this fuel under irradiation) have to be developed for compatibility with closed cycle strategies that might be needed for satisfying sustainability requirements. Last but not least, after a first "state-of-the-art" study in RAPHAEL, the modelling of fission product and dust transport, for which very large uncertainties exist, will likely need improvements in order to determine more accurately the HTR source term.

Moreover in order to investigate the potential of HTR/VHTR for higher performances in the long term, the exploration of innovative solutions for fuel and materials, started in RAPHAEL will have to be continued, keeping carefully the balance with shorter term needs for the demonstrator.

In the meantime industry started to select the main design options (pebble / block type fuel, IHX design concept, number of primary loops, general architecture of the system, temperature, pressure and power level, materials for the main components, etc.) in parallel to FP5 and FP6 projects. But the selection of some design options (most particularly for the IHX concept) and the validation of these choices require mock-up tests. As only few ones could be performed until now (see Part 4), a large experimental programme should be launched in FP7. It is recommended to start with separate effect tests in different facilities, each one providing one type of representative conditions: thermal loads / flow rate / coolant / chemistry. Such relatively small facilities will be more easily available than large integral test loops, and moreover they will allow identifying separately the sensitivity to each parameter, facilitating design optimisation, which would not be the case with integral tests. Then the selection of design options should be validated in a large integral test loop providing most of representative operating parameters, like the ones planned in CEA and Forschungszentrum Karlsruhe (figure 7). For the final qualification of the largest components, a bigger loop, in the range of 10 to 20 MW, will even be necessary. As the development of these large facilities will take several years and will require important funding, it is necessary to plan it right from the beginning of FP7.



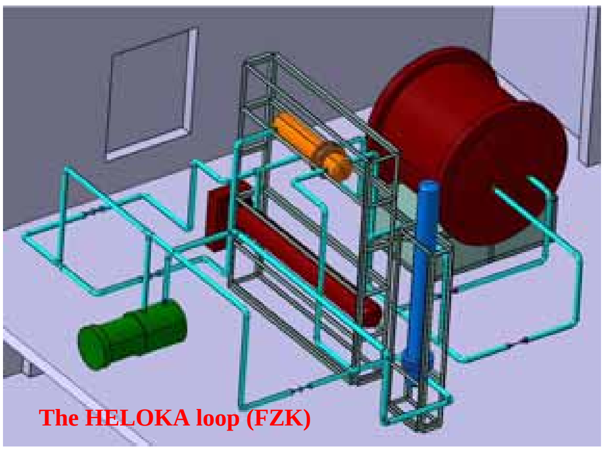


Figure 7: large helium loops planned in Europe

Design option selection will strongly depend on the possibility of licensing a reactor integrating such options. The definition of a safety reference frame taking benefit of the specific safety features of modular HTR is a key task that has to be undertaken during FP7 in continuity with the safety approach studies of FP5 and FP6, to pave the way for designing the demonstrator.

As already mentioned, there will be a strong interaction between the reactor design and the industrial application to which the reactor will provide heat. This interaction must therefore be handled right from the beginning of the project. During FP7, the end users will elaborate their requirements, which will be an important input for the selection of the reactor design options and the reactor designer will formulate the constraints imposed by the use of a nuclear heat source, which will be taken into account by end users for adaptation of their processes. Moreover it should be noted that, depending on the temperature level of the heat to be provided to the industrial process and on the safety requirements concerning the limitation of interactions between the nuclear reactor and the chemical plant (in particular, but not only the distance between the 2 facilities), one should pay a careful attention to the design of the heat transport system between them, which might be beyond the state-of-the-art.

The scheduling of the different phases of the programme of development of the demonstrator is represented in figure 8.

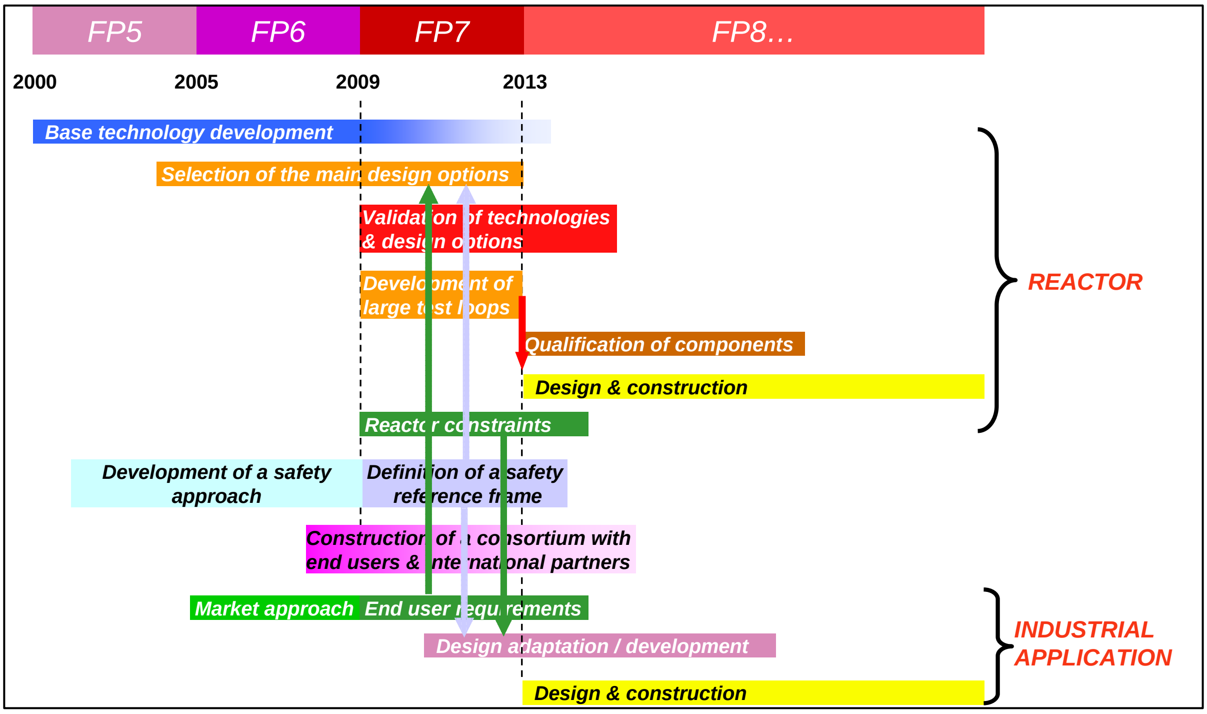


Figure 8: Schedule for the development of the demonstrator

6. Conclusion

It is clear that such an ambitious project, including design of an industrial prototype, will be possible only with a drastic increase of its public funding, in particular the one coming from the EURATOM Framework Programme, in comparison with the present situation. But even with a large European effort to support this project, HTR-TN would recommend internationalising it in order to develop synergies and to alleviate the European burden, either by merging with an existing international leading project (NGNP, PBMR, etc) or by developing a European project and attracting international partnership. Constructing such an international partnership should be a major task of the project, but it will be possible only if Europe appears to have on its own a strong programme with clear objectives and an obvious added value.

The ambition is high, but the challenge is of strategic importance. Energy will play a central role for the future of Europe: the cost and the long term security of energy supply will have a key impact in the prosperity of the EU; the increasing energy consumption is the main source of the global warming risk. If nuclear energy, thanks to competitive HTR/VHTR, enters the largest part of the energy market, which is not the electricity market, it can therefore significantly contribute both to the European prosperity and to the mastery of European CO_2 emissions.

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ABSTRACT

This paper presents the current status of the development of ELSY (the acronym for the European Lead-cooled System).
The ELSY reference design is a 600 MWe pool-type reactor cooled by pure lead. This concept is under development since September 2006, and is sponsored by the Sixth Framework Programme of EURATOM. The ELSY project, coordinated by Ansaldo Nucleare, is being performed by a consortium consisting of twenty organizations including seventeen from Europe, two from Korea and one from the USA. The partners are from industry, research organisations and universities.
ELSY aims to demonstrate the possibility of designing a fast critical reactor using simple engineered technical features, whilst fully complying with the Generation IV goals of sustainability, economics, safety, proliferation resistant and physical protection.
Compactness of the reactor building is possible due to the elimination of the Intermediate Cooling System, and the adoption of innovative DHR systems. Among the critical issues, the effect of the large mass of lead has been considered; this assessment allows being very confident in the feasibility of the reactor vessel and its support.

1. Introduction

The Generation IV (GEN IV) Technology Roadmap [1], prepared by GIF member countries, identified the six most promising advanced reactor systems and related fuel cycle and the R&D

necessary to develop these concepts for potential deployment. Among the promising reactor technologies being considered by the GIF, the LFR has been identified as a technology with great potential to meet the needs for both remote sites and central power stations.
In the GEN IV technology evaluations, the LFR system was top-ranked in sustainability because it uses a closed fuel cycle, and in proliferation resistance and physical protection because it employs a long-life core. It was rated good in safety and economics. The safety was considered to be enhanced by the choice of a relatively inert coolant. The LFR was primarily envisioned for missions in electricity and hydrogen production and actinide management. Given its R&D needs for fuel, materials, and corrosion control, the LFR system was estimated to be deployable by 2025. The LFR system features a fast-neutron spectrum and a closed fuel cycle for efficient conversion of fertile uranium. The LFR can also be used as a burner of all actinides from spent fuel and as a burner / breeder with thorium matrices.
The GIF LFR Provisional System Steering Committee has prepared a draft of the System Research Plan (SRP) for the Lead-Cooled Fast Reactor [2] with molten lead as the reference coolant and lead-bismuth as backup option. Figure 1 below illustrates the basic approach being recommended in the LFR SRP. It portrays the dual track viability research program with convergence to a single, combined demonstration facility (demo) leading to eventual deployment of both types of systems.

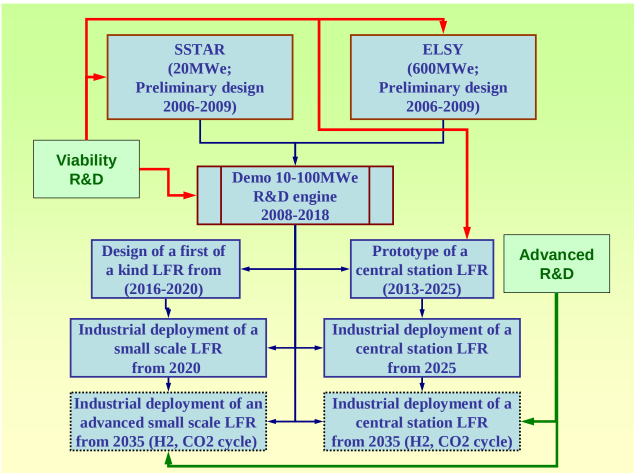

Fig.1. LFR SRP Conceptual Framework

This approach consists of the design of a small transportable system of 10–100 MWe size that features a very long refuelling interval, and of a larger system, rated at about 600 MWe, intended for central station power generation. Following the successful operation of the demo around the year 2018, a prototype development effort is expected for the central station LFR leading to industrial deployment at the horizon of 2025-2030. In the case of the small transportable (SSTAR) option the development of a first of a kind unit in the period 2016-2020 is foreseen. The design of the industrial prototype of the central station LFR and that of the first of a kind SSTAR should be carried out in parallel to the construction of the Demo and planned in such a way as to start construction as soon as beginning of the Demo operation at full power has given the main confidence of the viability of this new technology.

2. ELSY consortium

A major step in favour of the LFR occurred when EURATOM decided to fund ELSY (the acronym for the European Lead cooled System) - a Specific Targeted Research Project of the 6th European Framework Program (FP6) – proposed to investigate the economical feasibility of a lead-cooled, critical reactor of 600 MWe power [3-4] for nuclear waste transmutation. Since September 2006, a consortium of twenty organizations (from industry, research centres and universities) including seventeen from Europe, two from Republic of Korea and one from United States (Tab 1) has been pursuing the development of ELSY.
The ELSY project, scheduled to last three years, aims at demonstrating the possibility to design a competitive and safe Lead-cooled fast power reactor using simple engineered features. This prospect is

appealing also to private investors who have offered to participate in the initiative. This would create the conditions for advancing the ELSY activity even beyond the current sponsorship under Euratom's FP6. The use of compact, in-vessel steam generators and a simple primary circuit (Fig 2) with all internals possibly being removable are among the reactor features needed for competitive electric energy generation and long-term protection of investment.

Participant organisation		Country
Ansaldo Nucleare S.p.A	ANSALDO	Italy
AGH, Akademia Górniczo-Hutnicza	AGH	Poland
Centro Elettrotecnico Sperimentale Italiano	CESI	Italy
Inter Universities Consortium for Nuclear Technological Research	CIRTEN	Italy
Centre National de la Recherche Scientifique	CNRS	France
Empresarios Agrupados Internacional S.A.	EA	Spain
Electricité de France	EDF	France
Ente Per Le Nuove Tecnologie, L'energia e L'ambiente	ENEA	Italy
Forschungszentrum Karlsruhe GmbH	FZK	Germany
Institute for Nuclear Research	INR	Romania
European Commission, Joint Research Centre	JRC	Europe
Royal Institute of Technology-Stockholm	KTH	Sweden
Nuclear Research and Consultancy Group	NRG	Netherlands
Ustav jaderneho vyzkumu Rez, a.s. (Nuclear Research Institute Rez, plc.)	UJV	Czech Republic
Paul Scherrer Institut	PSI	Switzerland
Studiecentrum voor Kernenergie•Centre d'Etude de l'énergie Nucléaire	SCK•CEN	Belgium
Seoul National University	SNU	Korea
Del Fungo Giera Energia S.p.A.	DEL	Italy
Massachusetts Institute of Technology	MIT	USA
Korea Electrical Engineering and Science Research Institute	KESRI	Korea

Tab 1: Organizations involved in the ELSY project

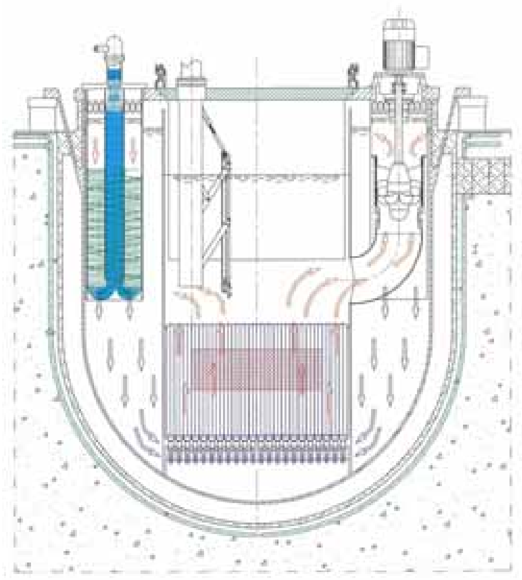

Fig 2 Preliminary scheme of the ELSY Reactor

The preliminary parameters of ELSY are specified in Tab 2.
To meet the technological needs of the ELSY project, it is important to capitalize on the strong synergy with other two European initiatives, “The Integrated Infrastructure Initiative VELLA,” [5] which is devoted to the dissemination of knowledge in the field of lead and lead-alloys technology, and the ”Integrated Project EUROTRANS [6]”.

Plant Characteristic	**Tentative Plant Parameters**
Power	600 MWe
Thermal efficiency	40 %
Primary coolant	Pure lead
Primary system	Pool type, compact
Primary coolant circulation, at power	Forced
Primary coolant pressure loss, at power	~ 1,5 bar
Primary coolant circulation for DHR	Natural circulation + Pony motors
Core inlet temperature	~ 400°C
Core outlet temperature	~ 480°C
Fuel	MOX with consideration also of nitrides and dispersed minor actinides
Fuel cladding material	T91 (aluminized)
Fuel cladding temperature (max)	~ 550°C
Main vessel	Austenitic stainless steel, hung, short-height ~ 10 m; diameter ~ 12 m
Safety vessel	Anchored to the reactor pit
Steam generators	N° 8, integrated in the main vessel
Secondary cycle	Water-supercritical steam at 240 bar, 450°C
Primary pumps	N° 4 or 8 mechanical, in the hot collector
Internals	Removable
Inner vessel	Cylindrical
Hot collector	Small-volume, above the core
Cold collector	Annular, outside the inner vessel, free level higher than free level of hot collector
DHR coolers	N° 4, DRC loops + a Reactor Vessel Air Cooling System .
Seismic design	2D isolators supporting the reactor building

Tab 2 Tentative parameters of the ELSY plant

2. Plant power and reactor vessel sizing

The ELSY power plant is tentatively sized at 600 MWe because only plants of the order of several hundreds MWe are expected to be economically affordable on the existing, well-interconnected grids of Europe. Because the mass of lead of a LFR is worldwide a-priori considered a critical issue for the reactor vessel which can limit the plant power, a preliminary mechanical verification, including seismic loads, has been performed from the beginning of the design activity based on preliminary parameters. The reactor vessel has been checked to ASME III code applying response spectra of a similar nuclear plant, EUR requirements and a reactor building supported by 2D seismic insulators.
The results show that the code requirements are satisfied for all service levels and allow being confident in the feasibility of the vessel and its support. The ongoing activity is now aimed to confirm that the assumed relatively small vessel dimensions, are realistic thanks to innovative solutions of the primary system layout. A LFR of a power larger than a medium power is potentially feasible according to these preliminary evaluations.

3. Coolant and thermal cycle

A large experience exists on LBE in Russia [7] and elsewhere [8-9]. Since lead is much more abundant (and less expensive) than bismuth, in case of deployment of a large number of reactors, pure lead as

coolant offers enhanced sustainability. Furthermore, the use of lead strongly reduces the production of the highly radioactive decay-heat generating polonium in the coolant with respect to LBE [10]. These are the main reasons for selecting lead as primary coolant for ELSY.
Operation at a higher lower limit of the thermal cycle, required by the use of pure lead, would be necessary also in the case of LBE to improve plant efficiency and to avoid the excessive embrittlement of structural material subjected to fast neutron flux.
The risk of lead freezing is reduced by the choice of a pool-type configuration.
The choice of a large reactor power suggests the use of forced circulation to shorten the reactor vessel, thereby avoiding excessive coolant mass and alleviating mechanical loads on the reactor vessel.
Thanks to the favorable neutronic characteristics of lead, the fuel pins of a lead-cooled reactor, similarly to LWRs, can be spaced more apart than in the case of sodium, resulting in a lower pressure drop across the core. As a consequence, in spite of the higher density of lead, the pump head can be kept low (on the order of one to two bars) with a reduced requirement for pumping power.
A possible primary-side thermal cycle of 400°C/480°C in lead, without an Intermediate Cooling System, offers reduced risk of steel creep and milder thermal transients, while providing the thermal efficiency above 40% with a supercritical Rankine steam cycle at 240 bar, 450°C .
The reactor vessel is designed to operate at the cold temperature of 400°C, which would be a safe condition even if oxygen control in the melt is temporarily lost. All reactor internals will have to operate at higher temperatures, at which it is necessary to rely on oxygen control, whereas fuel cladding could be surface-treated (aluminization seems to be a promising route) for a greater safety margin. An improved primary-side thermal cycle at higher core outlet temperature could be adopted in the longer term, as new materials become available

4. Decay heat removal

According to the predicted low primary system pressure loss and the favorable transport properties of lead, decay heat can be removed with lead in natural circulation in the primary system.
A simple system for decay heat removal is the Reactor Vessel Air Cooling System (RVACS), which consists basically of an annular tube bundle of U-tubes arranged in the reactor pit with atmospheric air flowing pipe-side in natural circulation. RVACS is a passive system, but its use without other systems can only be considered for small-size reactors since the vessel outer surface is relatively large in comparison with the reactor power. In the case of ELSY, the RVACS performance is sufficient only in the long term (about one month after shut down) and a Direct Reactor Cooling (DRC) system is needed equipped with coolers immersed in the primary system. Stringent safety and reliability requirements of the DRC system will be achieved by redundancy and diversification.
The DRC system is made of four loops; two loops operating with water (the W-DHR loops) and the remaining loops with water and/or air (the WA-DHR loops), (Fig 3).

Each W-DHR loop is made of a cooling water Storage Tank, a water-lead Dip Cooler, interconnecting piping, and steam vent piping to discharge steam to the atmosphere. The two W-DHR loops with the contribution of the RVACS are sufficient to remove the decay heat in order to respect the temperature limit of 650°C specified for the 4th Category, service level D, over a week time from reactor shut down.

Each WA-DHR loop is made of an inlet air duct, an air-lead Dip Cooler and an outlet air duct. The inlet air duct is equipped with an electric fan supplied by batteries. Isolation valves are installed in the inlet air and outlet ducts. A connection of the WA-DHR Dip Cooler to the cooling water storage tank of a W-DHR loop is also provided to for improved cooling with a mixture of air and water. The two WA-DHR loops with the contribution of the RVACS and the use of the water of the W-DHR loops in the short term from the reactor shut down, are sufficient to evacuate the decay heat in order to respect the temperature limit of 650°C established for the 4th Category, service level D. In the long term operation with air natural circulation is sufficient to respect the temperature limit.

The respect of the temperature limits of the 2nd and 3rd Category is ensured in operation with three out of the four DRC loops and the RVACS.

The Dip Cooler tube bundle is made of bayonet tubes. The bayonet consists of three concentric tubes, the outer two of which have the bottom end sealed. Water evaporation or air heating takes place in the annulus between inner tube and the intermediate tube. The annulus between the outer tube and intermediate tube is filled with He gas at a pressure higher than the lead pressure at the bottom end of the bundle. All annuli are interconnected to form a common He gas plenum, the pressure of which is continuously monitored. A leak from either walls of any of the outer tubes, is promptly detected because of depressurization of the common gas plenum.

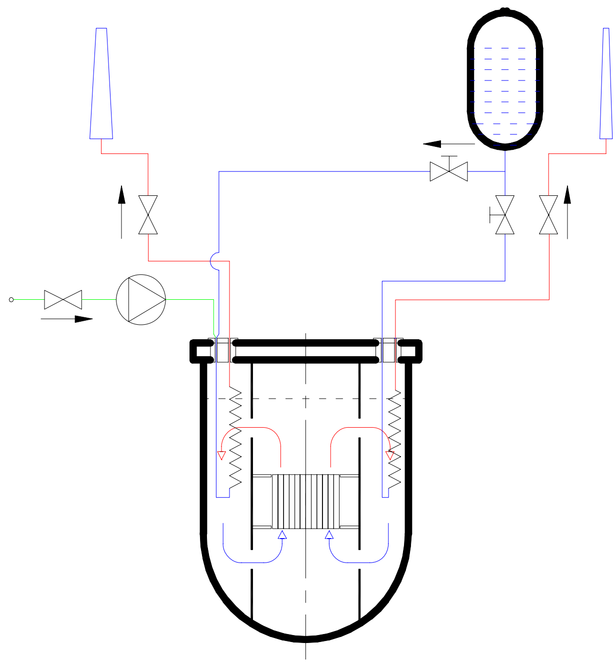

Fig 3. The DRC W-DHR (right-side) and WA-DHR loops, process scheme showing stored cooling water interconnection.

The bayonets of the ELSY DRC Dip Coolers are different with respect to classical bayonets, which consist each of only a pair of concentric tubes. The two outer tubes do not constitute a double walled tube, but are mechanically and thermally decoupled. This configuration allows to localize the most part of the thermal gradient, between lead and boiling water across the gas layer, avoiding both risk of lead freezing and excessive thermal stresses across the tube walls during DHR steady state operation and transients.

5. Primary system and reactor building

Figure 2 shows the cylindrical inner vessel concept, a scheme evaluated as a starting point for the primary system design of ELSY. Hot lead is pumped into the pool above the PP and driven through the SGU tube bundle into the cold pool. The free level of the hot pool inside the SGU is higher than the free level of the cold pool outside that is higher, in turn, than the free level of the hot pool above the core enclosed by the inner vessel.
A free level difference of cold and hot collectors at normal operating condition of only 1-2 m is sufficient to feed the core, eliminating the complicated, pressurized core feed system (known to sodium fast reactor community as Liposo and Sommier, in French) typical of the pool-type, sodium-cooled reactors.
Simplification of the internals will offer the possibility of removable in-vessel components, a provision for investment protection. In spite of the identified advantages of this scheme, design improvements are being developed at least to make the primary system more tolerant to Steam Generator Tube Rupture (SGTR) accidents. Compactness of the reactor building is the result of reduced footprint and height. The reduced footprint is allowed by the elimination of the Intermediate Cooling System, the reduced elevation is the result of the forced circulation, of the new DHR DRC system and of the design approach of reduced-height components.

6. ELSY can meet the generation IV goals

The main features identified in order to achieve the GEN IV goals are based either on the properties of lead as a coolant or are specific designs to be engineered for ELSY.

Sustainability
Because lead is a coolant with low neutron absorption and scattering, it is possible to maintain a fast neutron flux even with a large amount of coolant in the core. This allows an efficient use of neutrons, a breeding ratio of about 1 without fertile assemblies, long core life and a high fuel burn-up.
The fast neutron flux significantly reduces net MA generation, Pu recycling in a closed cycle being the condition recognized by GEN IV for waste minimization.
The potential capability of the LFR system to safely burn considerable amounts of recycled minor actinides within the fuel will add to the attractiveness of the LFR. To this end, different core configurations are being studied and compared (see a specific paper at this conference).

Economics.
A simple plant will be the basis for reduced capital and operating cost. A pool-type, low-pressure primary system offers great potential for plant simplification. The use of in-vessel Steam Generator Units (SGU's), and hence the eliminating the intermediate circuit, is expected to provide competitive generation of electricity in the LFR. The configuration of the reactor internals will be as simple as possible. The very low vapour pressure of molten lead should allow relaxation of the otherwise stringent requirements of gas-tightness of the reactor roof and possibly allow the adoption of simple fuel handling systems.

Reduction in the risk to capital results from the potential of removable/replaceable in-vessel components.

Safety and Reliability
Molten lead has the advantage of allowing operation of the primary system at atmospheric pressure. A low dose to the operators can also be predicted, owing to its low vapour pressure, high capability of trapping fission products and high shielding of gamma radiation. In the case of accidental air ingress, in particular during refuelling, any produced lead oxide can be reduced to lead by injection of hydrogen and the reactor operation is safely resumed.

The moderate ΔT between the core inlet-outlet temperature reduces the thermal stress during transients, and the relatively low core outlet temperature minimizes creep in steels.
It is possible to design fuel assemblies with fuel pins spaced as in the case of fuel assembly of the water reactor. This results in a moderate pressure loss through the core of about one bar, in spite of the high density of lead, with associated improved heat removal by natural circulation and the possibility of an innovative reactor layout such as the installation of the primary pumps in the hot collector to improve several aspects affecting safety. In case of leakage of the reactor vessel, the lower free level of the coolant will be sufficient to ensure the coolant circulation through the core and the safe decay heat removal. Any leaked lead would solidify without significant chemical reactions affecting the operation or performance of surrounding equipment.

With high-density lead as a coolant, fuel dispersion dominates over fuel compaction, making the occurrence of complex sequences leading to re-criticality less likely. In fact lead, with its higher density than oxide fuel and its natural convection flow, makes it difficult to lead to fuel aggregation with subsequent formation of a secondary critical mass in the event of postulated fuel failure.

Proliferation Resistance and Physical Protection
The use of MOX fuel containing MA increases proliferation resistance. The use of a coolant chemically compatible with air and water and operating at ambient pressure enhances Physical Protection. There is reduced need for robust protection against the risk of catastrophic events, initiated by acts of sabotage because there is a little risk of fire propagation and because of the passive safety functions. There are no credible scenarios of significant containment pressurization.

7. Conclusions

A major step in favor of the LFR did occur when EURATOM decided to fund the ELSY project, in response to the call “Nuclear Waste Transmutation in Critical Reactors,” to investigate the economic feasibility of using critical reactors for nuclear waste transmutation.
ELSY can find relevant synergy and technical feedbacks from the ongoing FP6 activities on ADS, and the Integrated Infrastructure Initiative (VELLA).
ELSY is expected to be a simple, innovative reactor, with compact primary system and reactor building, appealing to utilities for sustainable electric energy generation with reduced capital cost and construction time. Based on the promising initial results, it is expected that ELSY can confirm the ambitious objectives of the designers and open a phase of strong international support for LFR development and deployment.
Considering that significant commonality of R&D can be found between the small, transportable system and the medium-or large-sized system of the two GEN IV approaches, the GIF SRP proposes coordinated research with a single demonstration facility that can serve the R&D needs of both approaches. Full power operation of the Demo around the year 2018 - using to the greatest extent simple solutions, standard materials and operating at relatively low temperature, to reduce as much as possible the technological risks - could also justify the construction, at that date, of the first of a kind or industrial prototypes of SSTAR and ELSY and the industrial deployment at the horizon of 2025-2030 as foreseen in the GEN IV Roadmap.

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Session 18.1.3:

Countries' perspectives on nuclear energy policy

CONCEPT OF THE GLOBAL NUCLEAR ENERGY SYSTEM

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ABSTRACT

The paper considers the world energy demand till the middle of the century; demonstrates the possibilities of nuclear energy to meet this demand, and outlines innovative technology requirements determined by the development scope. Russia's potential contribution in meeting the challenges faced by the XXI century's nuclear power is also discussed.

1. Introduction

In accordance with the IAEA data, in 2006 thirty-two countries (having about two-thirds of the world population) operated 442 nuclear power reactors with total installed capacity of 369.7 GWe (net). Construction of 29 nuclear power units of about 24 GWe total installed capacity in twelve countries characterizes the short-term global nuclear energy prospects. Around a dozen of other countries officially announced their intentions to create nuclear sectors in their national power industries.

Contemporary plans of nuclear energy development by the mid-century, which the world countries are considering independently, have an internationally acknowledged target of about 700 GWe.

The idea of consolidating the international efforts aimed at providing an open access to nuclear energy for all countries, while preserving and maintaining the non-proliferation regime, is a global nuclear energy system concept development incentive.

2. Energy demand

In order to develop any suppositions concerning the global nuclear energy outlook (scale, structure, key requirements) in the long term, it would be necessary at first to assess the potential global nuclear energy demand – at least, till the middle of the XXI century.

The growth of the global energy consumption in the XXI century is determined by the two principal causes: the growth of population (according to various expert assessments – 1.3-1.9-fold by the mid-century), and the rapprochement of the consumption levels in developed and developing countries. Different sources estimate the energy consumption to increase with a factor of 1.6-2.5 by the mid-century.

The specific energy consumption time variance factor for the developed and the developing countries seems to be decisive, so it was specifically analysed by the experts of the leading Russian nuclear centre – Kurchatov Institute [1]. This analysis gave us important results (Fig. 1).

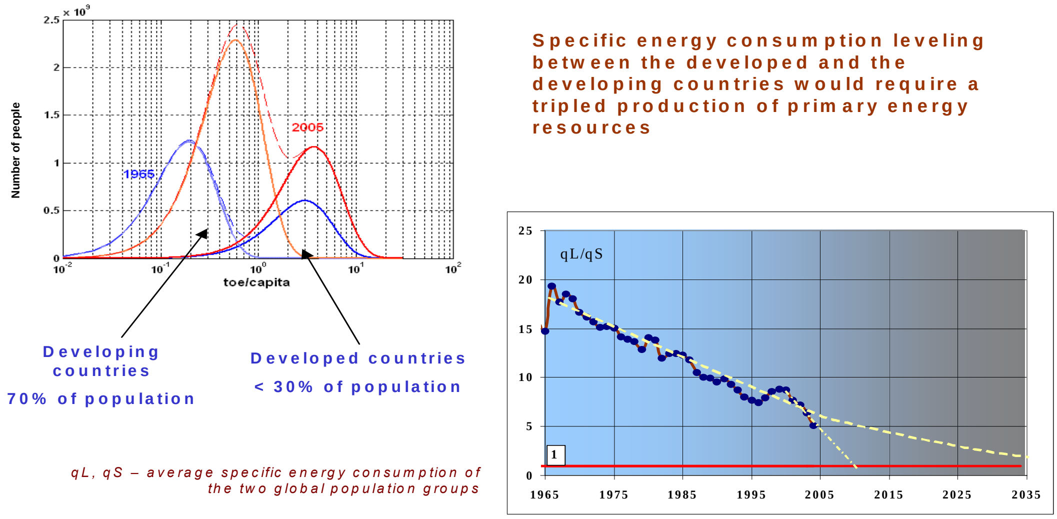

Fig.1. Rapprochement by Specific Energy Consumption

The assessment of the rate of rapprochement between specific energy consumptions in the two groups of countries is a key parameter determining the energy market situation. Assuming the rapprochement trend is maintained in the perspective, it is easy to calculate the required amount of primary energy. It can be shown that a continuation of the current world trends would result in an energy resource deficit already in the nearest future.

Assuming the primary energy sources are growing with a rate close to IAE forecasts, the fuel mix picture by the mid-century would contain an “unsatisfied demand” area (i.e., resources, which should be used to meet the projected energy demand).

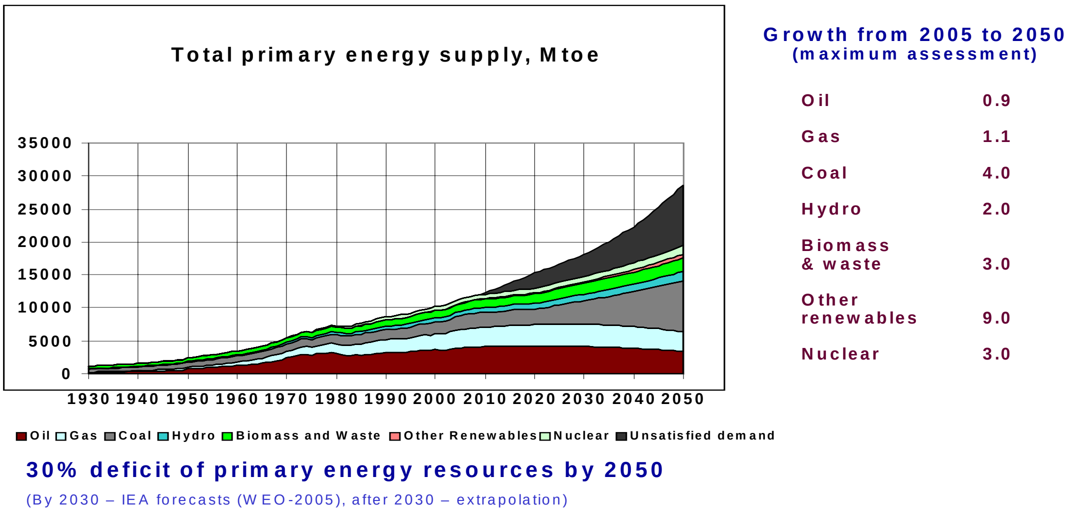

Fig. 2. Primary Energy Supply

If any new energy technology is to assure sustainable global energy development, it should release low emissions, be deployed on a large scale and have a long-term resource base available.

3. Nuclear energy development scenarios

Supposing the “unsatisfied demand” is met by nuclear energy, installed NPP capacities should make several thousand gigawatts by the mid-century.

Thus, **the projected XXI century’s world energy demand does not impose any upper limit on nuclear energy development**, the scale of which would be determined by development opportunities meeting the list of requirements, which should be analysed right now.

The bottom level of nuclear energy development by the mid-century could be represented by the above-mentioned current plans of its development in the world countries, or by the 1000 GWe level, which has been considered as a “ceiling” recently enough.

It is important that the **scale of nuclear energy development determines the key nuclear energy requirements –** among which we emphasize the fuel supply and the need of innovative technologies.

In scenarios conditionally considered as “low” (below 1000 GW by 2050), fuel supply requirements are met relatively easily, and there is practically no need of innovations or possibilities of extending the nuclear energy application sphere. In fact, such scenarios leave nuclear energy on the level of a “technological experiment” (or a result of the countries’ wish to possess nuclear technology in their national security interests), which has no significant impact on the energy supply of the mankind.

“High” nuclear energy development scenarios, which (also conditionally) are considered starting from the “medium” scenarios proposed by IPCC international expert group as far as 2000 (2000 GW by the mid-century, with nuclear share in primary energy not exceeding 20% by the century end) [2], determine principally new requirements to nuclear energy and, in the same time, open the opportunities to expand its application sphere.

Exemplary calculations performed form the nuclear energy structure: light-water reactors in once-through fuel cycle (Fig. 3), systems with moderate and high breeding in the closed fuel cycle (Fig. 4) are, naturally, quite different in terms of their basic parameters. Integral (for 100 years) demand of natural uranium ranges from speculative 30 to realistic 10 billion tons, and the maximum annual separation work – from 700 to 200 thousand tons of SWU (stabilized by the mid-century already in the moderate breeding scenario). Breeding systems would allow us to reduce the amount of spent nuclear fuel several times.

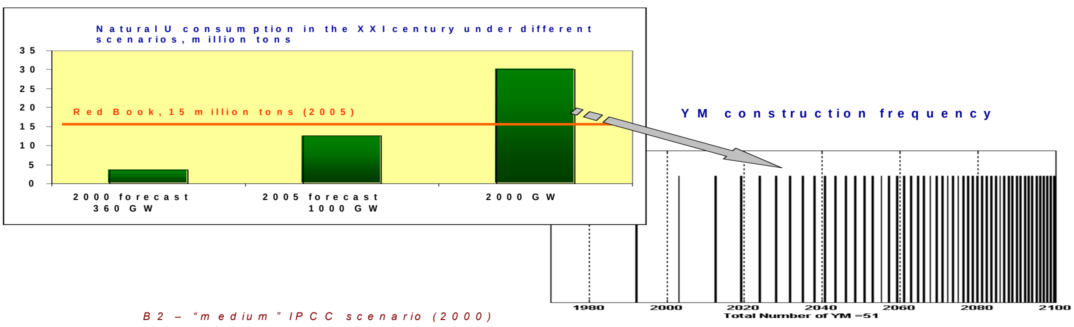

Fig. 3. Once-Through Nuclear Fuel Cycle

Without going into much detail, it should be nevertheless noted that the calculations have practically considered the whole “range of interest” of the global nuclear energy system development scenarios: from the very moderate approach with about 1000 GW of projected NPP capacity by the mid-century, to the so-called “aggressive” nuclear energy increasing its market attractiveness by replacing a fraction of other energy sources in the electricity generation and in other applications – such as hydrogen, heat or potable water production, with the global nuclear energy system capacity reaching up to ~ 10 000 GWe by 2100.

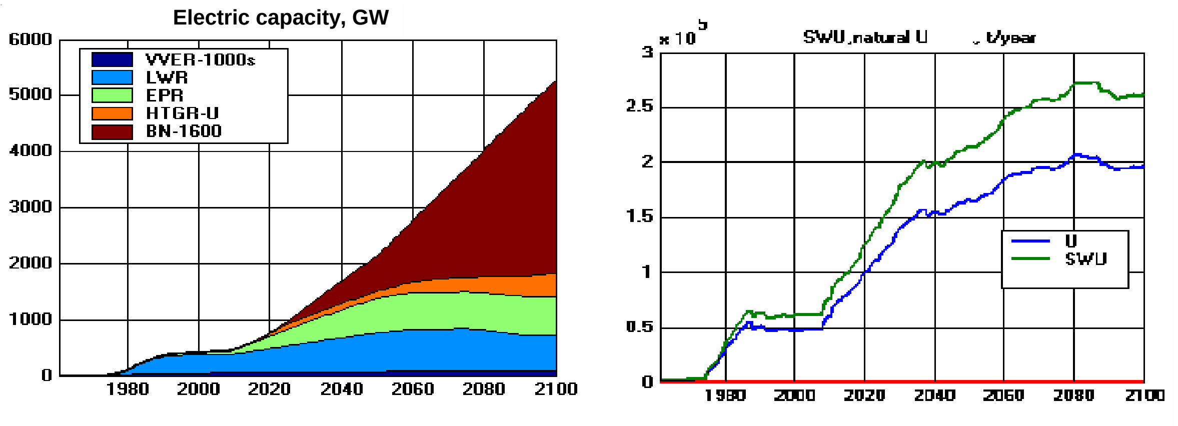

Fig. 4. Closed Fuel Cycle: “Moderate” Breeders (BR=1.25)

Thus, assuming the uranium resource constraints based on the existing data, realization of “high” nuclear energy development scenarios leaves a two-component nuclear energy system with plutonium breeding for further consideration.

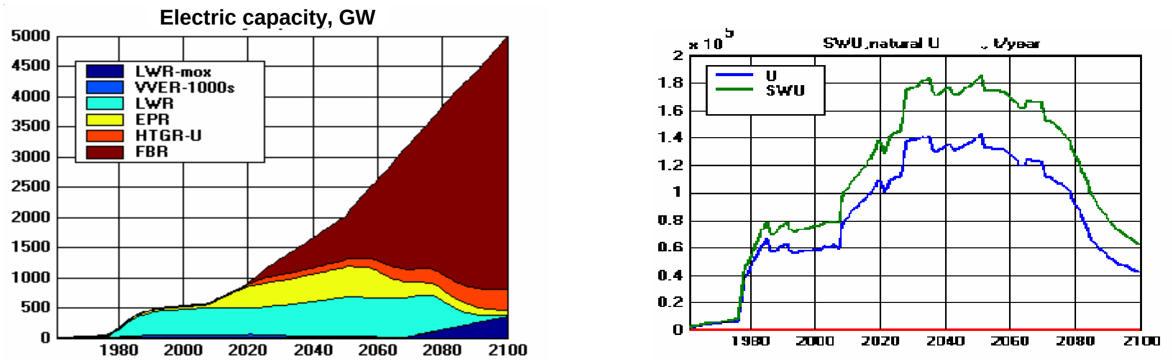

Fig. 5. Closed Fuel Cycle: “High” Breeders (BR=1.6)

Besides, these scenarios offer relatively strict conditions for the introduction rate of technological innovations. The latter include the closed fuel cycle based on new reprocessing technologies, “good” and “very good” breeders (with BR ~ 1.2÷.1.6), and “very good” LWRs with BR ~ 0.9 and with plutonium fuel.

As we see, the wish to develop nuclear energy with the given rate dictates so rapid innovations that – at least, for a considerable part of the world – its implementation would require a consolidated international effort to make nuclear energy accessible for all the countries concerned.

4. Russia’s contribution to nuclear energy challenges

Being one of the founders of the First Nuclear Era, Russia possesses vast experience of solving the key nuclear energy problems of the XXI century.

Today 10 Russian NPPs have an installed capacity of 23.2 GWe and generate about 16% of the country’s electricity. In accordance with the government’s Federal Program of the Nuclear Energy

Industry Development adopted in 2006, by 2020 the total installed capacity of Russian NPPs should reach 41 GWe, with annual energy production of about 300 TWh. The government intends to invest over 25 billion USD from the federal budget in the construction of NPPs between 2007 and 2015.

Russia's preparedness for the innovative development of nuclear energy technologies could be briefly summarized as follows:

Technologies for nuclear energy sources:
- Development of designs for the next generation of VVERs (NPP-2006 and large VVER) is nearing its completion, in view of massive NPP construction in the short term.
- Small NPP construction started; medium NPPs are being designed.
- Fast neutron reactor BN-800 is being built, in order to demonstrate and further improve the technology of the uranium-plutonium fuel cycle closing.
- New fast reactor designs are at various stages of development.
- Technologies of energy production for non-electric applications (in particular, for hydrogen production, heat supply and water desalination) are at various stages of readiness.

Closed fuel cycle technologies:
- Water chemical reprocessing technology for thermal reactor SNF, including uranium and plutonium separation and HLW vitrification, was demonstrated on an industrial level.
- Mixed uranium-plutonium oxide fuel production technologies for BN reactors were demonstrated on an experimental level.
- R&D started on alternative nuclear fuel cycle technologies (dry SNF reprocessing methods, minor actinides' transmutation, uranium-thorium cycle technology).

5. Conclusion

Concluding this brief review of scenarios and issues of the nuclear energy development in Russia and in the world, the following could be stated with confidence:
- The world is entering the system energy crisis. Nuclear energy development could stabilize the energy market situation till the mid-century.
- Russia is interested in accelerated nuclear power development, in order to preserve and use its resources efficiently. So Russia is capable to contribute significantly to the solution of challenges faced by the nuclear energy in the XXI century.

6. References

1. E.P. Velikhov et al., "Russia in the World Energy of the XXI Century", Moscow, IzdAt, 2006.
2. International Panel on Climate Change, Special Report on Emission Scenarios, Cambridge University Press, Cambridge, 2002.

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NORDIC NUCLEAR MARKET TRENDS

1 Background

The Nordic market consists of the Swedish nuclear operations at the three units in Forsmark, three in Oskarshamn and four at Ringhals, in total 10 units in operation. The results of a Swedish public vote in early 80s made the country decide against nuclear in the long run. This decisions lead to the closing of the nuclear units, Barsebäck 1 and 2.

Furthermore, The Nordic market is also represented by Finland and the two units at Olkiluoto and two units at Lovisa. In Finland AREVA is in the progress of building a new, third, N.P.P. on the Olkiluoto site.

Finally, At the Oskarshamn site the Swedish plants intermediate storage facility for spent fuel is located, CLAB or **C**entralt **L**ager **A**nvänt **B**ränsle. At this location spent fuel is shipped annually from all Swedish plants and stored for app. 30-50 years until final repository.

2 Electricity market

In the 90’s associated with the deregulation of the Nordic electricity market the price for electricity was fairly low (with a low at <0,1 SEK/kWh = 0,01 €/kWh) and sites were not investing a lot. Basically, only regular required maintenance and repairs were being carried out. With the rise of electricity price (caused by e.g. increased power demand, low water level in the hydro power dams etc) the Nordic fleet started investing in new components (less inspection time), power uprates (increased capacity), safety modernization (government requirements) etc. The higher prices and the increased demand were prime drivers to expand capacity. These drivers also caused a strict focus on outage duration driving down outage times.

3 Upgrades

The upgrades were done to most Nordic plants in two steps. First, during the 80’s a “soft” upgrade with no or small hardware changes to maximum app. 10% higher output. This was mainly achieved with higher recirculation pump flow, better fuel efficiency/usage etc. The trends for the 2000’s and onwards have been to do major upgrades thru change of hardware (e.g. reactor internals such as steam dryer and steam separator, generator/turbine upgrades), better/new fuel, safety upgrade and prolong the lifetime of the plants from 40 to 60 years. These are massive programs, the largest ones in the region (for nuclear) going on since the plants were built reaching up to 30% increased power output.

4 PULS

Oskarshamn unit 3 is currently underway with a major upgrade to 29% increased electrical output, from 1200 MWe to 1450 MWe. The project name is designated PULS, **P**ower **U**prate with **L**icensed **S**afety. The upgrade includes three parts: turbine, electrical systems and reactor systems. Furthermore the project includes lifetime extension from 40 to 60 years lifetime and adherence to new SKI (Swedish regulatory commission) safety requirements.

For the reactor systems internals a change out will be carried out of the steam separator/shroud head, steam dryer and main steam line valves. The recirculation pumps will be rebuilt for improved capacity.

5 Reactor internals replacement program

In the Nordic region several large reactor internal change outs have been carried out during the past 8-10 years including:

- Steam dryers – Olkiluoto 1&2, 2005/2006
- Steam separators – Oskarshamn 1, 1998 & Oskarshamn 2, 2003
- Core shroud head - Oskarshamn 1, 1998 & Oskarshamn 2, 2003
- Top guide – Forsmark 1&2, 2000
- Core shroud – Forsmark 1&2, 2000
- And several more smaller/medium size projects

Planned programs for the next five years include:

- Core shroud head - Forsmark 1,2&3, 2008-2010 & Oskarshamn 3, 2008
- Steam separators – Forsmark 1,2&3, 2008-2010 & Oskarshamn 3, 2008
- Steam dryers – Forsmark 1&2, 2008-2009 & Oskarshamn 3, 2008

6 Segmentation for old reactor internals

Following the large Reactor internals replacement programs several techniques have been developed to scrap/minimize volume of the old/replaced parts. The use of band saws in combination with hydraulic pliers/cutters have show most effective. Furthermore, no creation of airborne contamination (e.g. plasma cutting can cause airborne contamination) and debris size suitable for vacuuming post disposal are advantageous. To handle the large components several turning table have been used in combination with tilting/turning devices. Segmented parts have been put in containers for final disposal.

7 Field Services

The low electricity price on the market has also generated a lot of pressure to drive down outage times and increase availability. This has been achieved thru systematic approach to outage durations and obstacle removal to achieve this. The durations in the Nordic region now shows systematic outage times less then 10-15 days “braker to braker” for a full refueling outage. The record low outages at Olkiluoto show durations just above 7 days.

One key driver for achieving this has been close cooperation between key suppliers and the plants with long term contracts spanning over multiple services and really driving e.g. the lessons learned feedback process, long term planning and operational mode of the plants.

Finally, with the rising electricity price the trend in maintenance scope is also less price focus and more drivers regarding quality of performance. Outages are short enough and the main focus in the Nordic area is to remain short rather then shorten duration.

THE EFFECT OF ELECTRICITY GENERATING PARK RENEWAL ON FOSSIL AND NUCLEAR WASTE STREAMS: THE CASE FOR THE NETHERLANDS

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ABSTRACT

The paper concentrates on options for renewing the current Dutch electricity generating park in the next decades. For this purpose, the existing electric generating park of The Netherlands is modelled according to its fuel use and waste generating characteristics. The present electricity park consists of four types of generating plants: the single nuclear plant, gas-fired plants, coal-fired plants and renewable energy (mostly biomass and wind).

In this paper the effect of a generating park transition into one with a large share of nuclear energy on the waste streams, both fossil and nuclear, is analysed. Two demand growth scenarios are used, and nuclear phase-out is taken into account for comparison. For renewables, existing literature on planning is referenced, as well as for energy demand development. This implies a substantial growth for these sources, but their contribution remains limited in percentage. Additionally, in the high-demand scenario the demand growth of 1.5%/year causes a more than doubling of the electricity demand in 2060 compared to 2000. In the analyzed scenarios it is assumed that fossil fuels will become economically unattractive due to high CO_2 penalties, or even partly inaccessible due to phase-out by law. Then nuclear will substitute coal and gas to a large extent, growing to a contribution of more than 50% in 2060.

In a dynamic analysis, i.e. as a function of time, the electricity supply distribution by source is being determined with the DANESS and DEEA codes, as well as the emission of CO_2, SO_2, NO_x and high-level radioactive waste. By 2060, the CO_2 emission of the generating park with nuclear plants reduces to about one-third of that without. The nuclear sector is shared by the evolutionary reactor design EPR and the smaller-scale alternative PBMR. The additional CO_2 mitigation by the PBMRs in cogeneration mode is quantified as well: the CO_2 emission of the Dutch electricity sector could even fall below zero when the avoided emission of industrial heating is subtracted from the CO_2 emission of the fossil-fired power plants.

When replacing fossil-generated electricity by nuclear, CO_2 and other gaseous waste is traded for radioactive waste, the CO_2 amount being in the order of a million times the amount of radioactive waste. To reduce the amount of nuclear waste further, recycling can be applied. The options of direct spent fuel storage and reprocessing are compared for the amounts of waste until 2060, both in mass and in volume. Obviously, reprocessing of spent fuel results in a significant reduction of volume that is needed to finally dispose used radioactive materials in geological repositories. Also, much of the volume will be occupied by PBMR pebble fuel elements. Separation of graphite from the fuel elements, and storing the fuel particles only, would already bring a volume reduction of over 90% for this fuel type.

1 Introduction

Most scenarios for electricity supply development for Western Europe assume a decline for nuclear generation in the coming decades, or a small increase followed by a decline, e.g. the European study ‘European Energy and Transport, Trends to 2030 – update 2005’[1]. Some scenarios with high economic growth assume an increase in nuclear generation to cover the demand growth associated with the economic growth, e.g. [2].

This study however considers a scenario where nuclear energy is deliberately employed for coupled economic-environmental reasons, for a real country departing from an existing electricity generating park.

The Netherlands currently has a generating park of 21 GWe (2004), running for three quarters on natural gas, see fig. 1. Already for some decades The Netherlands is a main gas producer itself, explaining the large gas share to electricity generation and the low nuclear share, compared to the European mean nuclear share of 35%. However, the main gas source at Slochteren in the north of the country is expected to run out in 2030, and the smaller sources below the Wadden Sea at least before 2050.

So if the Netherlands don’t want to rely heavily on natural gas imports in the future, some form of transition has to take place in the electricity generating sector. The government already set fairly ambitious targets for renewable generation, and forced conservation by legal bans or rationing of electricity is beyond the way of current thinking. Current government plans indicate obligatory CO_2 sequestration for new coal plants, making the coal option economically unattractive. So the nuclear option remains the more obvious alternative to generate base-load quantities of electricity with existing technology.

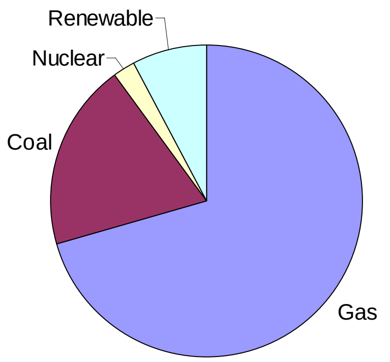

Figure 1 Installed electricity generating capacity distribution in the Netherlands. Renewable includes biomass and wind.

2 The nuclear/renewable transition scenario for The Netherlands

As electricity generation in the first place should stimulate prosperity and economy, no capital destruction by forced shutdown of power stations is envisaged. We depart from the existing electricity generating park, and the government stimulation plans for renewables are left intact.

During the past 40 years, there has been an increase in the Netherlands of the electricity consumption by a factor of 5.8, which implies an average annual growth of 4.5% [3]. It seems unrealistic to extrapolate this growth rate for the next 50 years, considering the decrease in growth of the Dutch population. Recent studies consider more moderate growth rates. The Dutch study ‘Referentieramingen’ (Reference Estimates), performed for the Government to forecast the Dutch energy consumption and the resulting environmental impact up to 2020, considered annual growth rates of 1.7% for the ‘Strong Europe’ scenario, and 2.7% for the ‘Global Economy’ scenario [4]. On the other hand, the CASCADE MINTS project, funded by the European Union under the support of the 6th RTD Framework Programme, considered annual growth rates ranging from 0.65% in 2010 to about 0.35% in 2030 for the "Baseline" case [5]. The recent Dutch study "Deltaplan Kernenergie" assumed a constant annual growth rate for the electricity production of 1.5% up to 2060 [6].

For the present study, two different growth scenarios have been considered (see also Fig.2):

1. The scenario based on the assumptions of the "Deltaplan Kernenergie", assuming a constant annual growth rate of 1.5%;
2. The scenario based on the assumptions of the CASCADE MINTS project, assuming a constant annual growth rate declining from 0.65% in 2010 to 0.35% in 2030.

In addition, the following boundary conditions have been assumed:

- Phase-out of coal-fired plants: the existing coal-fired plants are serving out their planned lifetimes and no new ones are commissioned except those that already have been planned;
- The contribution of renewable energy (wind, biomass) to the total electricity production is not determined by the market but by government planning. It will increase by 20% in 2020, and by 30% in 2040. These assumptions are in line with the forecast of the "Referentieramingen" [4];
- A gradual deployment of nuclear reactors in the next decades. Presently, only one nuclear power plant is operated in the Netherlands, the Borssele nuclear power plant. Various reactor types are being offered today or will be offered in the coming years. For the present analysis, a fleet consisting of one type of large reactor unit and one type of smaller unit, the latter suitable for heat and power cogeneration, has been assumed for the next decades. For the large unit the European Pressurized Reactor (EPR) was selected, and for the small unit the Pebble Bed Modular Reactor (PBMR).
- For the cases with deployment of nuclear reactors, the options of direct disposal of spent fuel ("Once Through" case), and reprocessing of spent fuel ("Reprocessing" case) have been considered.

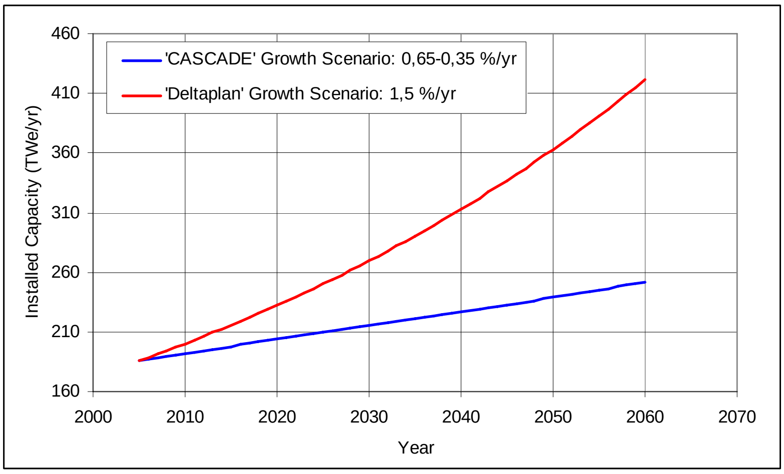

Figure 2 Forecast of the installed electricity generating capacity in the Netherlands for two different growth scenarios.

For comparison reasons, a scenario taking into account the nuclear phase out option has been considered, the nuclear phase-out scenario. For that scenario, an average growth rate for the electricity consumption of 1.5% has been assumed.

An overview of the main design parameters of the fossil-fuel fired plants and the nuclear reactors and fuel cycle is given in Table 1.

Table 1 Average design parameters of the different facilities

	Gas Fired Plant	Coal Fired Plant	Borssele NPP	EPR	PBMR
Power per plant (MWe)	400	520	484	1600	160
Plant lifetime (yr)	35	30	26	60[1]	50[1]
Plant capacity factor (-)	0,85	0,85	0,93	0,91	0,95
Efficiency factor (-)	0.45	0.39	0.35	0.37	0.41
Plant construction time (yr)	2	3	(existing plant)	5	3
Expected overnight cost (B€)	0,30	0,45	(existing plant)	1,3	0,19
O&M Cost, (Euro/MWhe)	8,6[2]	14,2[2]	4,6	3,9	4,4
UO_2 Enrichment (%)	-	-	3,1	4,2	8,1
Burnup (GWd/tHM)	-	-	33	50	90

[1] Including lifetime extension
[2] inclusive price of CO_2

3 Computer tools

Dynamic Energy Economics Analysis (DEEA) is a system dynamics tool which is able to simulate scenarios for the future deployment of fossil-fuel, nuclear and renewable energy systems. Driven by a future energy demand, new energy systems are introduced by means of a decision model that is mainly based on the profit per MWhe for each of the different electricity-generating options. DEEA is a macroscopic tool and intended to provide relatively quick results. This brings about that the code models are relatively straightforward, taking into account the overall processes and avoiding too much details. Seven types of nuclear reactors as well as gas-fired and coal-fired fossil fuel plants are characterized by gross data. Renewables are simply modelled by power and energy demand growth rate. The economics model takes into account interest and discount rates and the price of electricity, and compares this with economic factors that are specific for each energy generating system (e.g. levelized cost, fuel cost, carbon tax).

Given a future energy demand, DEEA calculates the relative contributions of nuclear, fossil fuel, and renewable energy systems to the total energy production. The development of the nuclear energy production in time then serves as the boundary condition of a detailed analysis of the nuclear fuel cycle. This analysis is performed with the DANESS computer tool.

For the assessment of the nuclear fuel cycle strategies, the DANESS code (“Dynamic Analysis of Nuclear Energy System Strategies” [7]), Version 3.2.03, was used to simulate the flows of fissile material, fresh fuel, spent fuel, high level waste as well as all intermediate stocks and fuel cycle facility throughput. DANESS is an integrated dynamic nuclear process model for the analysis of today’s and future nuclear energy systems on a fuel batch, reactor, and country, regional or worldwide level. Starting from today’s nuclear reactor park and fuel cycle situation DANESS analyzes energy-demand driven nuclear energy system scenarios over time and allows the simulation of changing nuclear reactor parks and fuel cycle options. New reactors are introduced based on the energy demand and the economic and technological ability to build new reactors. The technological development of reactors and fuel cycle facilities is modelled to simulate delays in availability of technology. Levelized fuel cycle costs are calculated for each nuclear fuel batch for each type of reactor over time and are combined with capital cost models to arrive at energy generation costs per reactor and, by aggregation, into a cost of energy for the whole nuclear energy system. A utility sector and government-policy model are implemented to simulate the decision-making process for new generating assets and new fuel cycle options. The different functionalities of DANESS may be switched on or off by the user according the intended use. The architecture of the DANESS code is depicted schematically in fig. 3.

For the calculation of the amount of nuclear waste, a fuel cycle model is used, as shown in fig. 4. Properties of all fuel cycle facilities are input, including capacity and transition time. For each reactor, a fuel type and back-end route (direct storage or/reprocessing) is set. The amounts of waste are given in tonnes heavy metal (tHM), and converted to volumes in m^3 for the results in chapter 8.

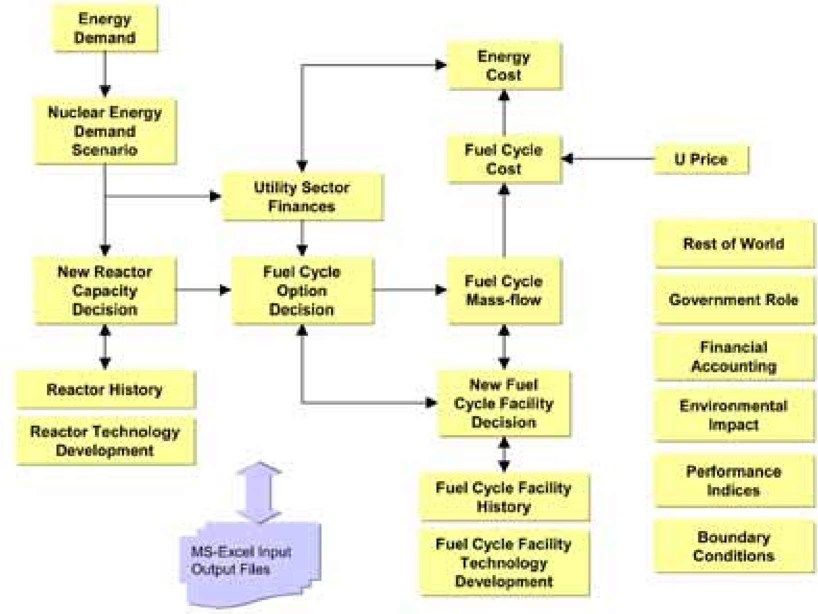

Figure 3 Schematics of the architecture of the DANESS code.

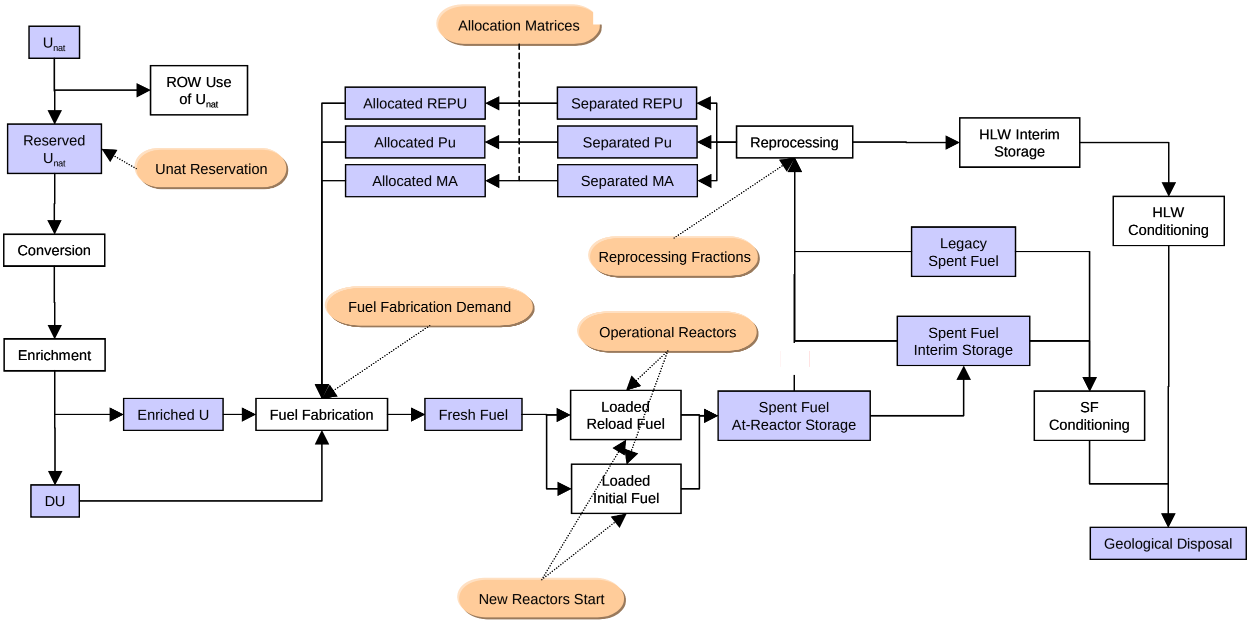

Figure 4 Fuel cycle model in DANESS code.

4 Electricity supply

The electricity supply distribution over the available sources is determined for the three selected scenarios:

- Nuclear/renewable transition with high demand rise: fig. 5,
- Nuclear/renewable transition with low demand rise: fig. 6,
- Nuclear phase-out with high demand rise: fig. 7.

The initial rise in electricity production is caused by the deployment of newly-built fossil-fuel power plants (e.g. 800MW Sloe generating plant, 800MW gas plant Eemshaven, and several others), whereas existing plants are not yet shut down. Around 2040 a 'bend' in the integrated curves can be observed: after the phase-out of coal-fired plants, the demand growth is fully covered by nuclear+renewables, so no additional growth of electricity from gas-fired plants is needed.

It can be seen that, with the prescribed growth rate of the renewables, in the high demand scenario nuclear energy will become the largest electricity source with 54% in 2060, whereas in the low demand scenario the renewables take the largest share with 52% in that year. In the nuclear phase-out scenario the electricity need that is not covered by the renewables is almost equally shared between gas and coal.

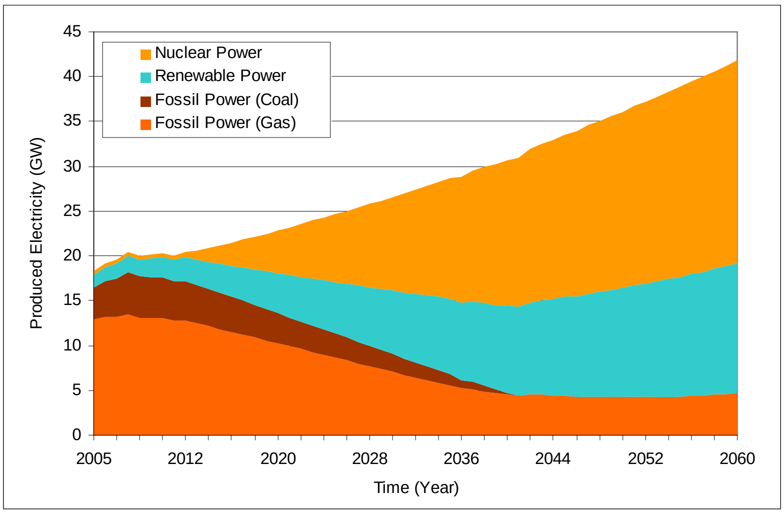

Figure 5 Produced electricity taking into account a annual growth rate of 1.5%.

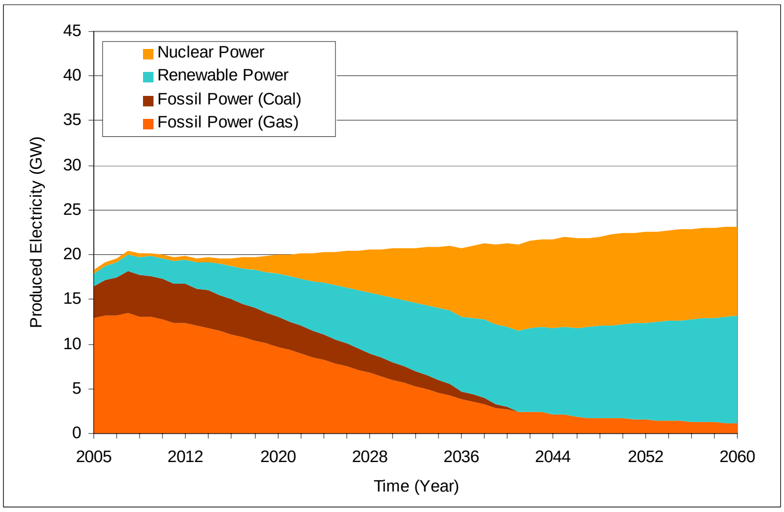

Figure 6 Produced electricity taking into account the low demand scenario.

5 Fossil waste generation

The amounts of the gaseous waste emissions from the fossil-fired stations for the next decades have been depicted in fig. 8, 9 and 10. Fig. 8 gives the carbon dioxide (CO_2) emissions, fig. 9 the nitrogen oxide (NO_x) emissions, and fig. 10 the sulphur dioxide (SO_2) emissions. The SO_2 emissions result for the largest part from the combustion of coal. Although in the Netherlands it is required to implement measures to reduce the SO_2 emissions from coal-fired plants, approximately 10% of the total generated SO_2 still is released to the atmosphere. Through better SO_2 reduction methods this percentage is likely to decrease in the future so that the estimated values shown in fig.10 represent upper limit values.

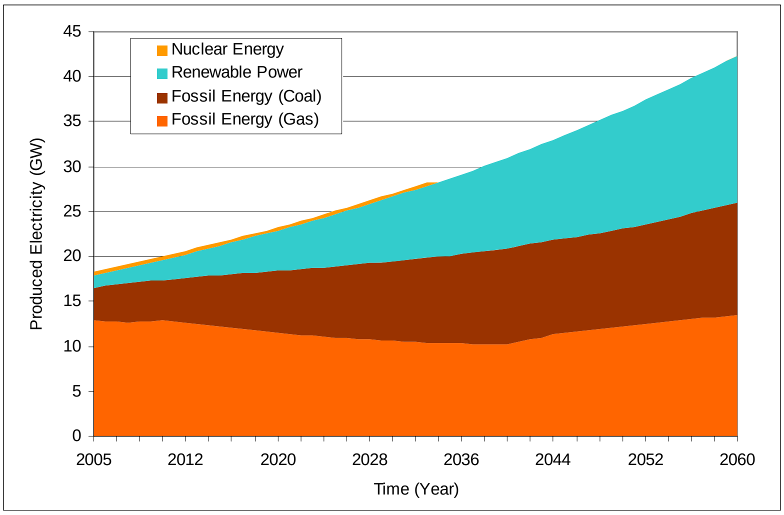

Figure 7 Produced electricity taking into account the nuclear phase-out scenario with 1.5% growth rate. The thin line for nuclear energy is the single existing plant serving out its licensed life.

The difference between the nuclear and non-nuclear scenarios is obvious: for 1.5% demand rise the accumulated CO_2 emission of the nuclear scenario in 2060 is only 12% of that of the nuclear phase-out scenario. In other words, by introducing nuclear energy on the proposed scale, 88% of the CO_2 emission of the Dutch electricity generating park can be avoided. The NO_x release rate plunges at least 90% (fig.9), whereas the SO_2 release vanishes as a result of phase-out of coal-fired plants (fig.10), which account for 95% per Gigajoule for the SO_2 release. These trends clearly indicate that the deployment of nuclear power for electricity generation is a serious option to reduce significantly the emission of hazardous exhaust gases.

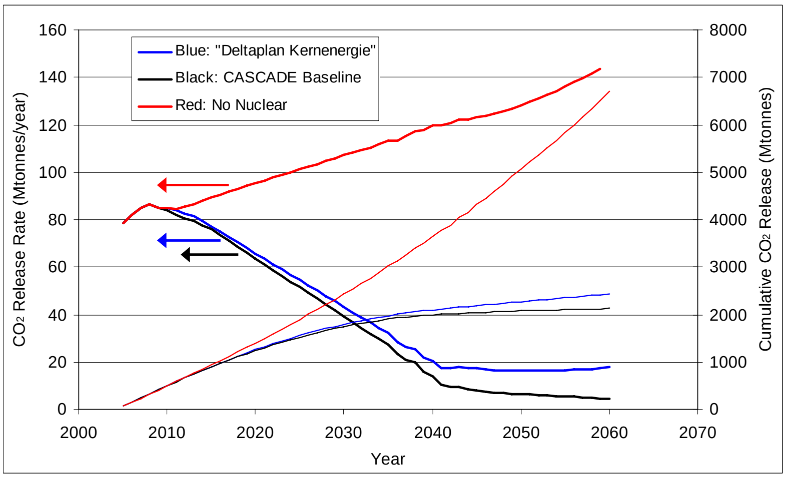

Figure 8 CO_2 release rate (thick curves) and cumulative CO_2 release from 2005 on (thin curves) for the three considered scenarios.

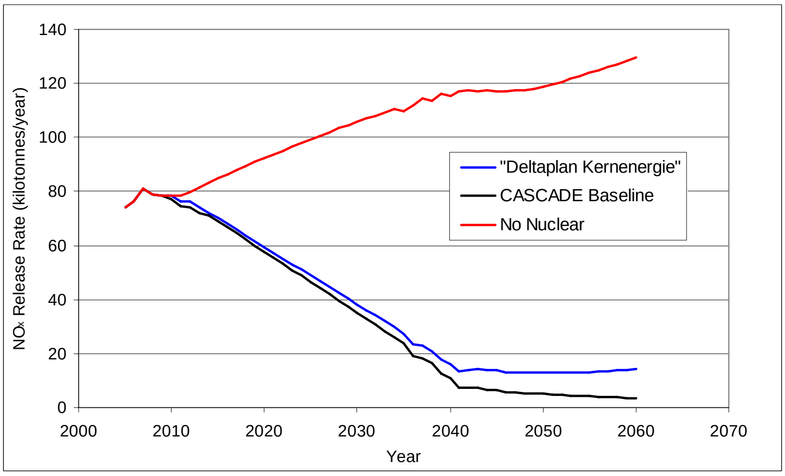

Figure 9 Calculated NO_x release rate for the three considered scenarios.

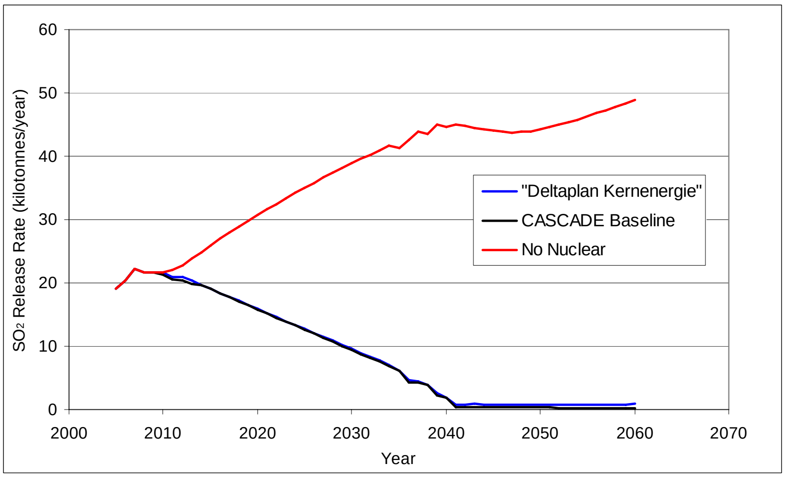

Figure 10 Calculated SO_2 release rate for the three considered scenarios.

6 Deployment of nuclear energy

The deployment of nuclear reactors (EPR, PBMR) as calculated by DANESS, is depicted in fig. 11 and fig. 12. DANESS calculates a relatively larger deployment of the small-scale PBMRs in the "CASCADE" case as compared to the "Deltaplan" case, because of the slower growth rate of the energy demand in the "CASCADE" case, which makes it less attractive to deploy the larger-capacity reactor types (1600 MW EPRs). In case of slower energy demand growth rates, the deployment of large reactors would result in a significant over-capacity of produced electricity. The more gradual deployment of the smaller PBMRs leads to a better match of the demand of electricity. From about 2040 there is less growth in the demand curve for nuclear energy, so the deployment of the smaller-capacity PBMR reactors is preferred above the large-capacity EPR reactors for the reason of demand matching. Therefore no additional EPRs are foreseen.

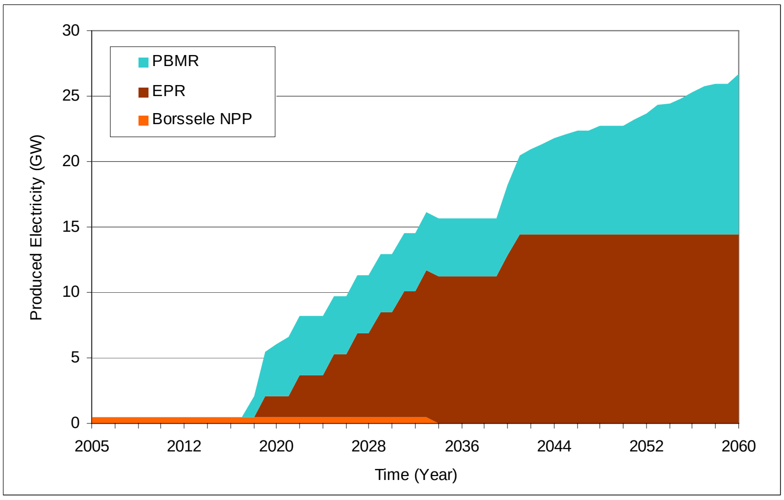

Figure 11 Calculated deployment of nuclear reactors for the case "Deltaplan Kernenergie" (1.5% annual growth rate)

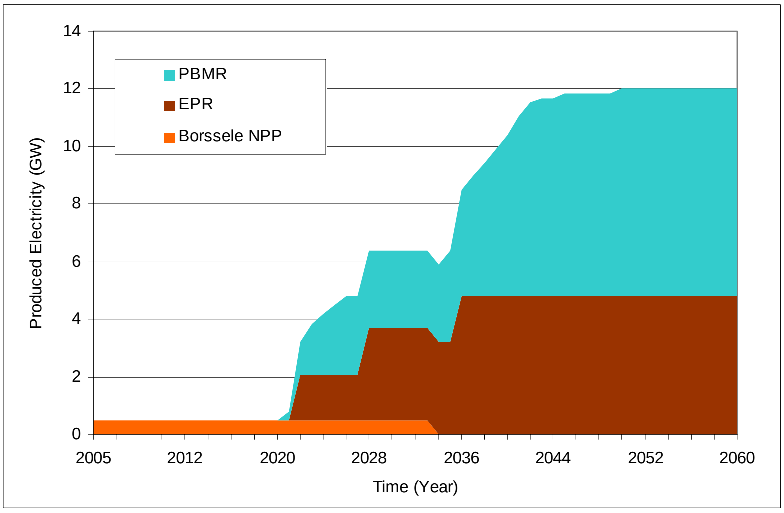

Figure 12 Calculated deployment of nuclear reactors for the CASCADE case (moderate annual growth rate)

7 Effect of nuclear cogeneration on CO_2 mitigation

Originally the development of the high-temperature gas-cooled reactor was started to be able to supply not only electricity, but also process heat and cogeneration to various sectors of industry. From previous studies (e.g. [8]), it was demonstrated that high-temperature gas-cooled reactors are capable to deliver, apart from electricity, a significant amount of heat, i.e. up to about 30% of the total thermal power.

In fig. 13 the CO_2 release is depicted for the three considered scenarios, now for the nuclear scenarios also depicting the additional avoided CO_2 emission when using the PBMR in cogeneration mode. It can be seen that the CO_2 emission of the Dutch electricity sector even falls below zero when the avoided emission of industrial heating is subtracted from the CO_2 emission of fossil-fired power plants.

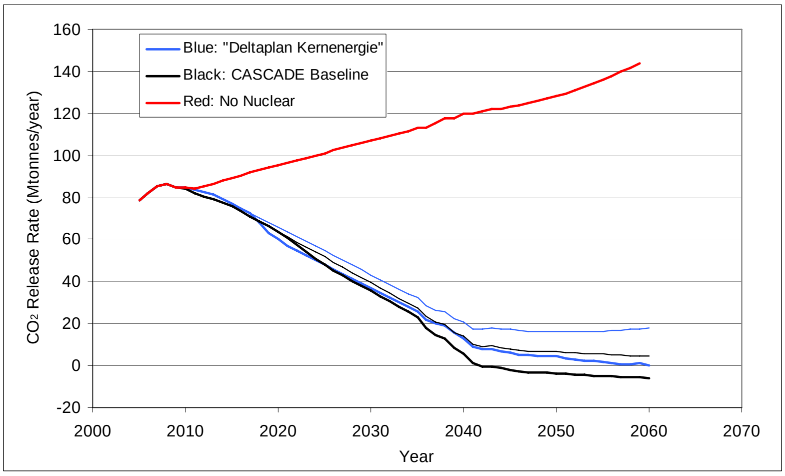

Figure 13 CO_2 release rate for the three considered scenarios. Thick curves: including additional avoided CO_2 from the deployment of heat cogeneration by PBMRs. Thin curves: no additional avoided CO_2 considered.

8 Spent nuclear fuel and nuclear waste

The amounts of nuclear wastes that have to be taken care of, have been calculated as a function of time for two options: direct disposal (‘Once Through’) and recycling (‘reprocessing’). For the two options, the amount of waste in interim storage facilities is shown in fig. 14, and the amount of waste in final disposal in fig. 15. As can be seen in the fuel cycle model scheme of figure 4, spent fuel first moves from the reactor to the ‘At Reactor’ storage, consisting of the spent fuel storage ponds at the nuclear plants. After a certain cooling down period at the reactor storage, in this study set to 5 years, the spent fuel is transferred to the spent fuel interim storage facility, where it is able to cool down further. After this, it can take two routes:

- It is sent to the spent fuel conditioning facility where it is treated for final disposal in a final disposal facility;
- It is sent to a reprocessing facility, where uranium and plutonium are recovered after which the remaining high level waste is transferred to a HLW interim storage facility for a cooling down period. In the high level waste conditioning facility the waste is prepared for final (geological) disposal.

In fig. 14, the black curve indicates the total amount of spent fuel stored at the nuclear power plants for the high demand scenario. The blue lines indicate the amount of spent fuel stored in interim storage. It can be seen that after about 2045 the waste arisings decrease as a result of the decreased growth in nuclear demand (cf. fig.11). The difference between the two blue lines is reflected by the thin red line: the amount of high level waste coming from the reprocessing plant.

In fig. 15, the amount of high level waste coming from the reprocessing plant is still very low in 2060. This is primarily caused by long transit and waiting periods for reprocessing. For this case still much high level waste is in the pipeline and will arrive at the final repository after 2060. This illustrates the contradiction of societal demand for an operational final storage facility at the start of a nuclear expansion programme on the one hand, and on the other hand the actual arriving of high level waste from the reprocessing plant only several decades later.

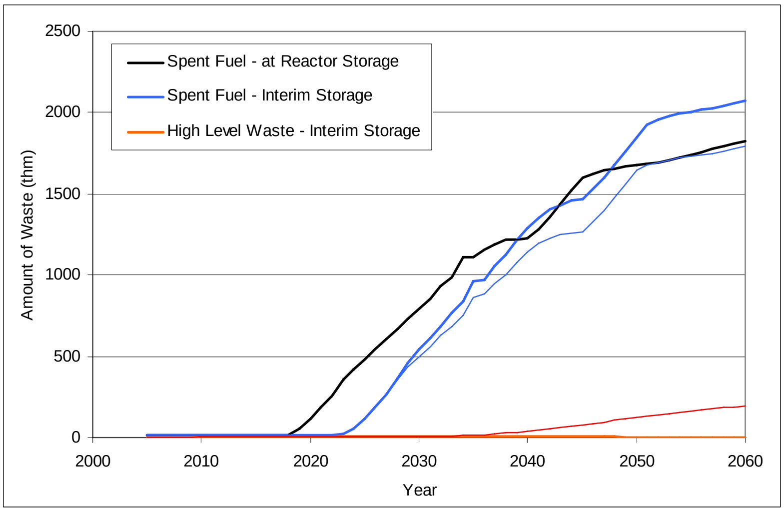

Figure 14 Amount of wastes in storage arising from the nuclear reactors for the scenario "Deltaplan".

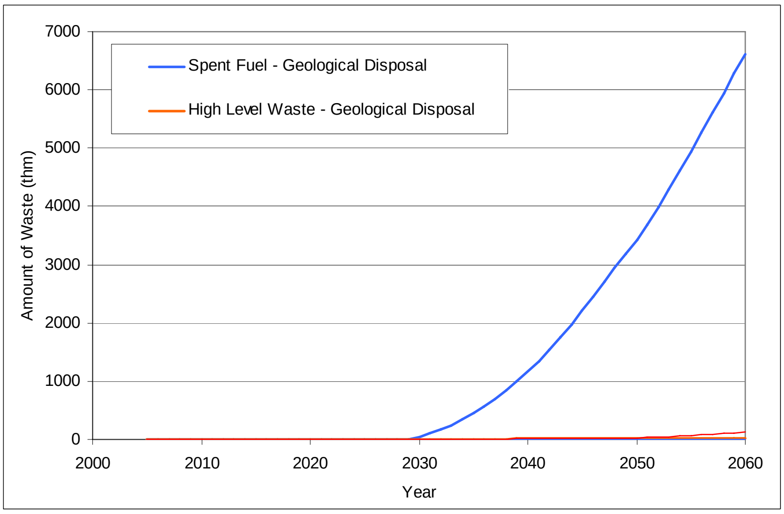

Figure 15 Amount of wastes in geological disposal for the scenario "Deltaplan Kernenergie" – thick curves: once-through case; thin lines: reprocessing case

For the low demand "CASCADE MINTS" scenario, the predicted amounts of stored waste in the year 2060 are about 30% to 55% less, depending on the type of waste, in comparison with the high-demand "Deltaplan" scenario.

In fig. 16, the actual container volumes are depicted that are needed to for interim storage of spent fuel, high-level wastes and the PBMR pebbles. For vitrified high level waste, the COGEMA HLW container [9] is considered, and for spent fuel the ONDRAF-NIRAS design [10]. For the PBMR pebbles, the German design storage canister [11] is adopted. We see the volume of the high level waste containers completely vanishing against the large volume of containers holding non-reprocessed spent fuel. For the most part, the larger SF container volume comes from containers holding PBMR pebbles, that mainly consist of graphite and only 7% of nuclear fuel. For the CASCADE scenario, the spent fuel volume rises only to 8500 m^3 in 2060, that is 71% of the volume in that year for the Deltaplan scenario. This can be seen in the bar chart of fig. 17 as well, where the spent PBMR pebble fuel is indicated separately.

From fig. 17 it is also clear that the used PBMR pebbles require by far the most storage and disposal capacity, because for the PBMRs also the moderator material (matrix graphite) of the fuel pebble is also considered as waste. The volumes that are needed to store and dispose HLW are only a minor fraction of the total required volumes.

The growth of the volume of waste containers in geological disposal over time for the whole Dutch nuclear park is similar to fig. 15, with a volume of 42000 m^3 in 2060 for the case of no reprocessing for the high electricity demand case, and 28000 m^3 for the low demand case. Fig. 18 is comparing the volume of waste containers in the final storage facility for the two demand cases in the year 2060, distinguishing between EPR and PBMR waste. For PBMR, also the amount of waste emerging when recycling the graphite, storing only the coated particle fuel (still no reprocessing). This measure already would reduce the PBMR spent pebble volume by 92%.

Table 2 lists the calculated effective volumes of the waste canisters that are needed to contain the spent fuels, high-level wastes, and PBMR pebbles.

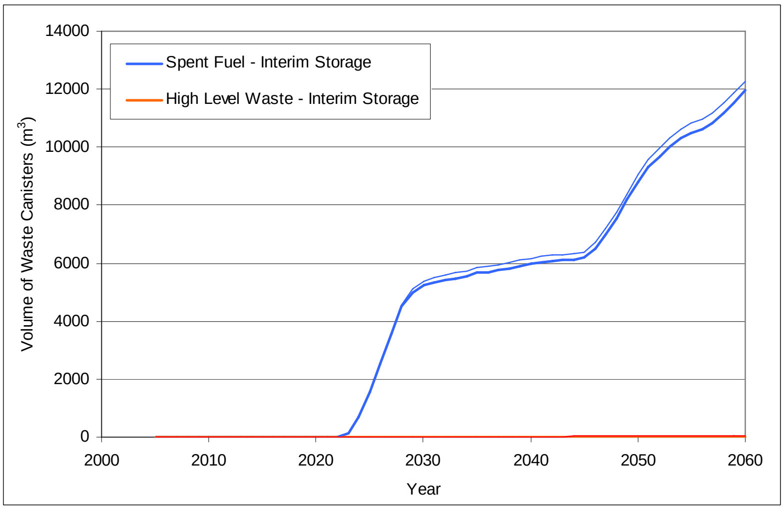

Figure 16 Volume of wastes containers in interim storage arising from the nuclear reactors for the scenario “Deltaplan Kernenergie” – thick curve: once-through case; thin line: reprocessing case.

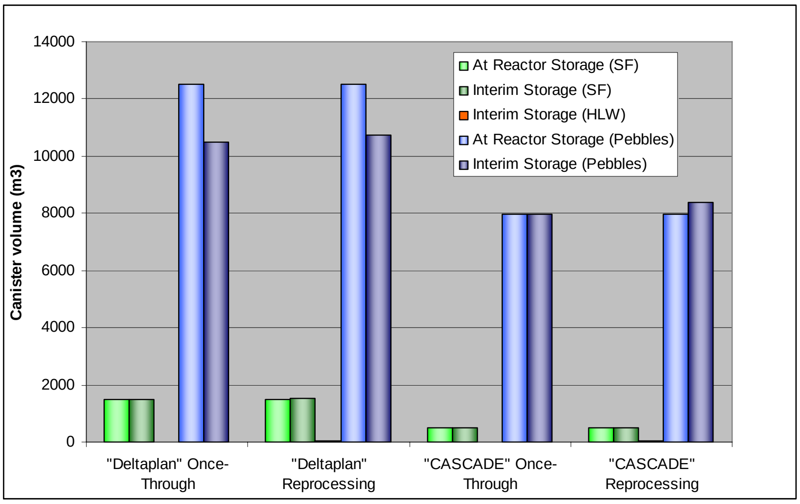

Figure 17 Comparison of the expected waste volume in storage at the nuclear plants and in the interim storage facility in the year 2060.

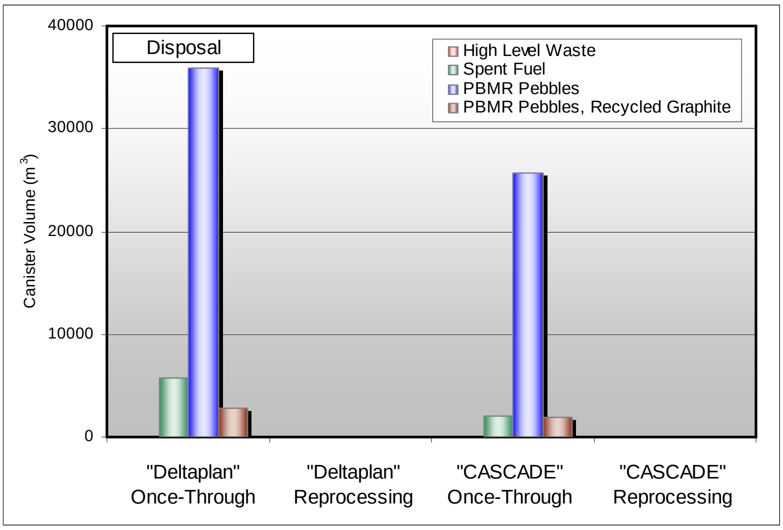

Figure 18 Comparison of the volume of waste containers in the final storage facility for the two demand cases in the year 2060, distinguishing between EPR and PBMR waste. The case for recycling the PBMR graphite, storing only the coated fuel particles, is shown as well.

Table 2 Comparison of canister volumes needed for the storage and disposal of nuclear wastes for the different scenarios in the year 2060.

Effective Volume	Type of storage	Deltaplan Once-Through	Deltaplan Reprocessing	CASCADE Once-Through	CASCADE Reprocessing
Spent LWR fuel	Interim	1468	1517	489	514
	Final	5777	0	2038	0
High Level Waste (reprocessed fuel)	Interim	0	61	0	21
	Final	7	27	7	20
Spent PBMR pebble fuel	Interim	10475	10729	7976	8375
	Final	35899	0	25675	0
	Final, recycled graphite	2764	0	1977	0

9 Comparison with existing interim storage capacity in The Netherlands

The facility for interim storage of spent fuel and high level nuclear waste is called HABOG ('Hoogradioactief Afval Behandelings- en Opslag Gebouw', Highly-radioactive Waste Treatment and Storage Building). It is located near the city of Vlissingen and the Borssele nuclear power station in the south of the country [12]. The HABOG-building is a modular building. This means the building can be extended if necessary. At this time there are three vaults for the storage of heat generating waste and three bunkers for the storage of non-heat generating waste. The license permits only a full load of two of the three vaults or bunkers. It should always be possible to unload one vault or bunker for inspection.

The capacity of each vault is 135 canisters with vitrified waste and 35 canisters with spent fuel. This means a total capacity at this moment of 270 canisters with vitrified waste and 70 canisters with spent fuel. The capacity of 2 bunkers is approximately 600 drums with different types of conditioned waste. The total volume of all the waste will be 750 m^3.

In the high demand case, this capacity will already be used by 2024, and in the low demand case in 2028. The amount of equivalent HABOG capacities for the three cases (no reprocessing, no reprocessing but with graphite recycling for the PBMR waste, and reprocessing) per demand scenario in 2060 can be seen in fig. 19. Most storage capacity is needed in the case of high demand and direct storage of all spent fuel: 72 times the current HABOG capacity. This can be reduced to 28 by recycling the graphite of the PBMR spent fuel, and to 17 by recycling the fuel itself (reprocessing). The figures for the low demand scenario are accordingly lower.

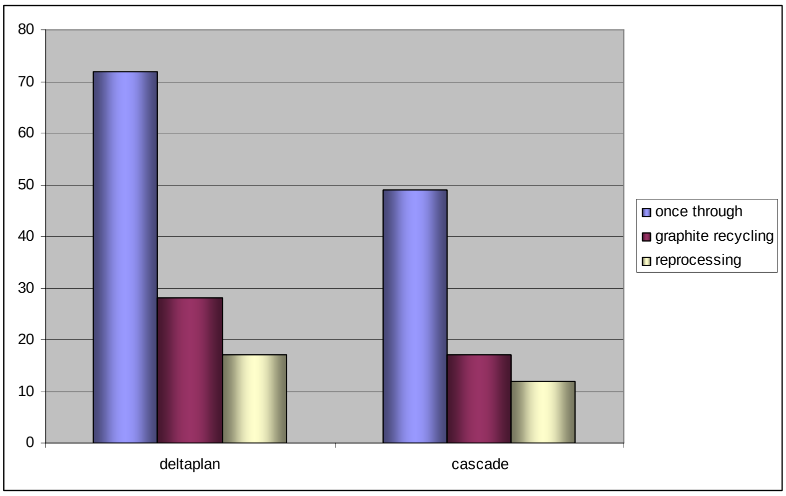

Figure 19 Equivalent number of HABOG volume capacities needed in 2060 for the case without reprocessing, the case without reprocessing but with graphite recycling, and the case with reprocessing.

10 Conclusions

Table 3 summarizes the amounts of waste generated for the high demand ('Deltaplan') scenario and the low demand ('CASCADE') scenario. In the case of the deployment of nuclear reactors, in 2060 the release of CO_2 as a result of electricity generation will be reduced to one-third as compared to the case where nuclear electricity generation is not considered.

Choosing between nuclear and non-nuclear generation parks is a trade-off of waste types. All types of energy mix come with a waste mix. When replacing fossil-generated electricity by nuclear, CO_2 and other gaseous waste is traded for radioactive waste, the CO_2 amount being in the order of a million times the amount of radioactive waste. By signing the Kyoto protocol, The Netherlands obliged itself to reduce CO_2 emissions by 13 Mton/year in 20 years time. By implementing the nuclear/renewable transition scenario, this target could be more than achieved by the electricity generating sector alone (factor 1.8), leaving room for other sectors with less possibilities for CO_2 reduction.

The following conclusions with respect to nuclear waste reduction can be drawn from this study:

- Reprocessing of spent fuel results in a significant reduction of volume that is needed to finally dispose used radioactive materials in geological repositories.
- Reprocessing of spent fuel impels the deployment of capacity to separate the recyclable material from the HLW.
- In case of not reprocessing, most of the space in the interim and final storage facilities is occupied by spent PBMR fuel. By recycling the graphite part of the waste, a significant volume reduction can be achieved.

Table 3 Comparison of amounts of waste of fossil and nuclear origin.

<table>
<tr><th colspan="2"></th><th>No Nuclear</th><th>Deltaplan Once-Through</th><th>Deltaplan Reprocessing</th><th>CASCADE Once-Through</th><th>CASCADE Reprocessing</th></tr>
<tr><td colspan="2">Cumulative CO_2 Release (Mton)</td><td>6700</td><td colspan="2">2430</td><td colspan="2">2130</td></tr>
<tr><td colspan="2">Cumulative NO_x Release (kton)</td><td>5850</td><td colspan="2">2160</td><td colspan="2">5850</td></tr>
<tr><td colspan="2">Cumulative SO_2 Release (kton)</td><td>2075</td><td colspan="2">510</td><td colspan="2">2075</td></tr>
<tr><td rowspan="3">Spent Fuel Inventory (tHM)</td><td>At Reactor Storage</td><td></td><td>1825</td><td>1825</td><td>770</td><td>770</td></tr>
<tr><td>Interim Storage</td><td></td><td>2075</td><td>1794</td><td>923</td><td>808</td></tr>
<tr><td>Geological Disposal</td><td></td><td>6616</td><td>0,0</td><td>2886</td><td>0,0</td></tr>
<tr><td rowspan="2">High Level Waste Inventory (tHM)</td><td>Interim Storage</td><td></td><td>0,0</td><td>193</td><td>0,0</td><td>133</td></tr>
<tr><td>Geological Disposal</td><td></td><td>19,9</td><td>125</td><td>19,8</td><td>60,1</td></tr>
</table>

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Session 18.1.4:

Status of future projects

PRELIMINARY SEISMIC ANALYSIS OF A NEXT GENERATION NPP

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ABSTRACT

The aim of paper is to evaluate and to characterize the structural response behaviour of reactor building internal structures under specific site seismic loading characteristics in order to determine whether these ones satisfy present international safety regulations.
Moreover, the correlations between the horizontal seismic earthquake values recorded on rock site as well as the calculated NRC ones and the in site complex nuclear building structure effects are investigated and discussed, with an application example to near term nuclear power plants (NPPs) concepts (like IRIS or ELSY).
To the purpose finite element method and sub-structures approach were employed for studying the overall dynamic behaviour of the considered system also accounting for the structure and soil interactions. The analysed results, the mentioned effects and the response of internal components (e.g. Nuclear Building, Vessel, etc.) seem to confirm the possibility to achieve an upgrading of geometry and performances of the proposed solutions for the considered NPPs.

1. Introduction

Earthquake response of nuclear structures depends on both the ground motion characteristics and the dynamic properties of the structures. Integrity of structures, systems and components of a nuclear power plant must be ensured in case of any design condition, in particular in the case of seismic accident conditions. In fact, when a structure is subjected to dynamic loads, as seismic ones, the behaviour of structural material may be significantly different from the one characteristic of static load applications. The seismic analysis of a nuclear power plant is one main regulatory requirement for the design and construction approval [1].
The adopted analysis procedure provides minimum requirements and acceptable methods for the evaluation of safety related structures of NPPs. Moreover Soil Structure Interaction (SSI) is considered to be important because take into account the phenomenon of coupling between a structure and its supporting medium (soil, sand or rock) during an earthquake, due to the nonlinear behaviour of several type of soil etc.
In this paper a preliminary application of the proposed analysis methodology to an innovative LWR reactor (**I**nternational **R**eactor **I**nnovative and **S**ecure- IRIS) structure is presented. This preliminary analysis is intended to evaluate the dynamic loads propagation from the ground to the Internals (e.g. Reactor Pressure Vessel (RPV) or Steam Generators (SG)), considering also the mentioned SSIs effects.

2. Model and structural system descriptions

Between the new LWR concept IRIS is one of the most interesting new reactor concept under study at present. The IRIS integral pressure vessel (RPV) is larger than a traditional PWR one, but the size of the IRIS containment system (CS) volume is a fraction of that of corresponding loop reactors, resulting in a significant reduction in the overall reactor size [2-3]. This size reduction, combined with the spherical geometry results in a CS pressure bearing capability at least three times higher than a typical loop reactor cylindrical containment, with the same metal thickness and stress levels (Fig.1). NPPs are always composed of a number of adjacent structures, so in order to ensure adequate treatment of interaction effects the main buildings should be considered, as in the proposed example model, with their real geometry and material characteristics.

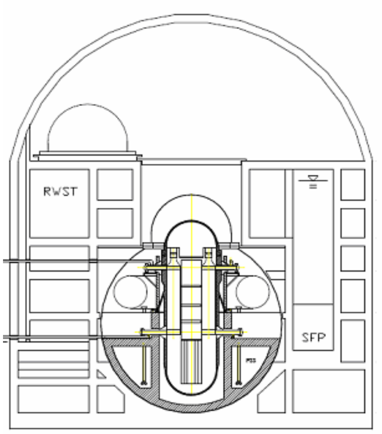


Fig. 1 – Scheme of the whole Nuclear Island

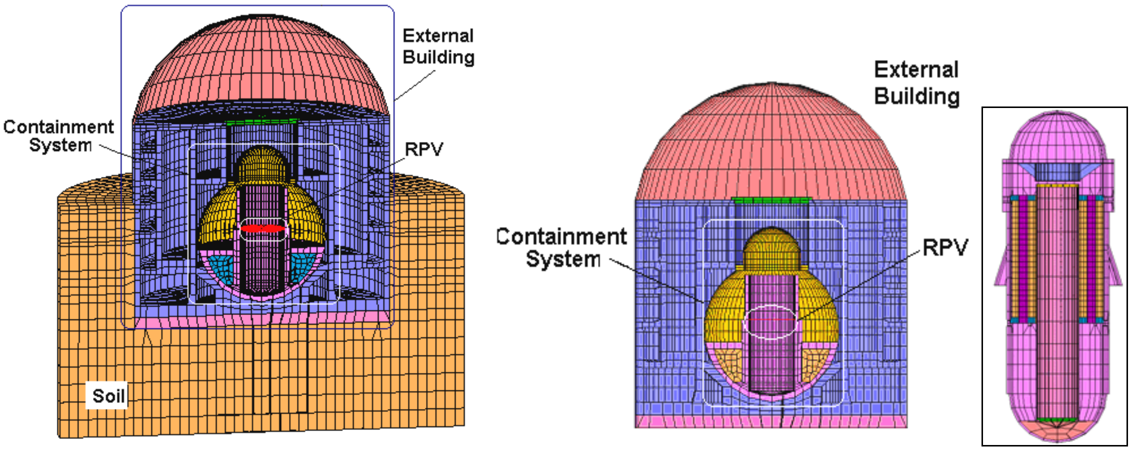
The seismic response of a structure should be determined by means of setting up an adequate mathematical model and calculation of its response to the prescribed seismic input. In this application example, the Nuclear Island may be subdivided into three main structures:

- Auxiliary building including External Building (EB);
- Inner containment structures (CS);
- Containment internals including Reactor Pressure Vessel (RPV).

In the considered EB the reactor type surrounds the CS. The overall structure is assumed to have a rigid foundation, which is the interface between the nuclear island and the soil. The Soil was modelled as a homogeneous loose sand zone, which may influence in different ways the horizontal and vertical propagation wave and the rocking vibration effects [4-5]. The clearly nonlinear constitutive behaviour of soil should be accounted for as an elasto-plastic Mohr-Coulomb material. The CS was one of the main structures studied, which was characterized by different mass and stiffness distribution over the height; due mainly to the upper hemispherical steel structure and to the bottom concrete wall structure. The main internal structures such as the RPV and the suppression water pools content was considered as lumped masses connected to the containment wall nodes. The RPV internals (e.g. Barrel, SG tubes, etc.) are considered as a set of lumped masses linked respectively to the appropriate locations. Moreover the attachments of the SG headers to the RPV internal wall were considered as rigid restrains without mass.

2.1 Method of analysis

In the numerical simulations, three dimensional models (MSC.MARC FEM code) were implemented in order to analyze the seismic behaviour of the whole nuclear island with and without soil-structure interaction (Fig. 2). In order to ensure adequate treatment of interaction effects, firstly the main structures were modelled in only one 3D finite element model with some simplified assumptions, and subsequently the seismic analysis was carried out by means of the Substructure model approach that allows to separate the NPP seismic analysis problem into a series of simpler ones that can be solved each independently. Simplified structural models may be used to provide an adequate representation of considered structures and to generate in-structure response spectra at the reference location or subsystem supports [6].
The performed analyses referred to the same structures coupled with the foundation depth and soil effects. The whole model was represented by a cylindrical structure resting on a shallow cylindrical foundation that was embedded in a homogeneous soil layer. Moreover to simplify the analyses and reduce the calculation time some internal structures (e.g. RPV, SGs, etc.), in each models, were represented like lumped masses distributed at appropriate chosen locations. The Time History approach was used in all analyses, coupled with the before mentioned Substructure method, to evaluate the effects of a Safety Shutdown Earthquake (SSE).



(a) (b)

Fig. 2 (a),(b)- Complete structural model with and without Soil

The seismic excitation was simulated by means of artificial acceleration having the maximum Peak Ground Acceleration (PGA) equal to 0.3 g calculated for an appropriate damping in according to the NRC Regulatory Guide 1.60 only in the horizontal translation direction and for excitation duration equal to 30s.

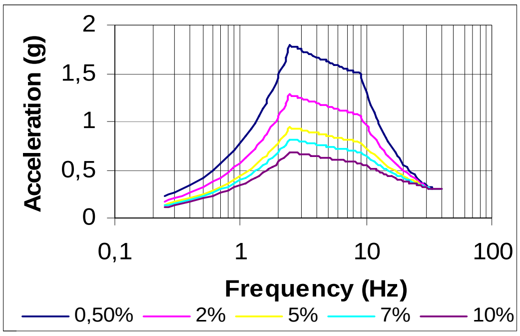
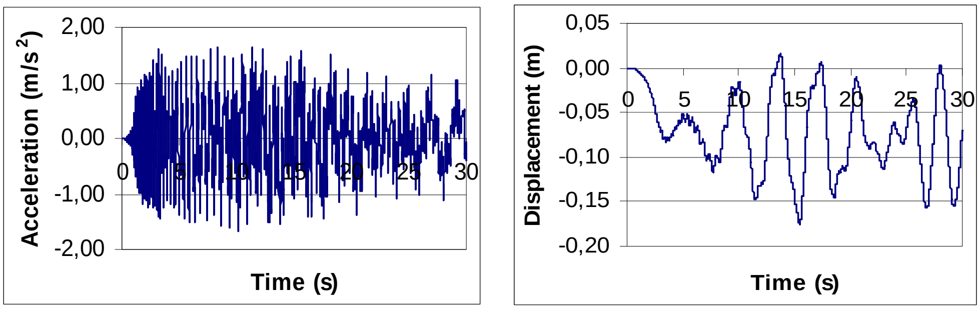


Fig.3 – Input Response Spectra (PGA = 0.3g)

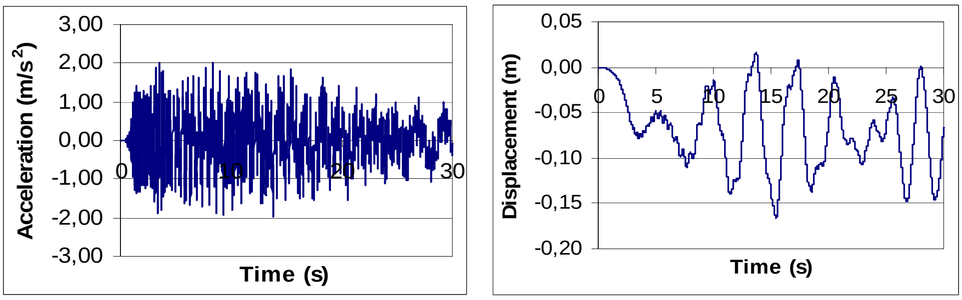
To study the effectiveness of the damping system in mitigating the seismic response of the buildings, the maximum accelerations and displacements of the considered structures at chosen reference point were obtained from the results of each analysis.

3. Numerical results

The input motion of the SSE is used to carry out results referred to the complete nuclear island model including all main structure, with (Model A) and without soil (Model B) highlighting the loading intensity decrease as the seismic input moves from the free field through the soil, EB and CS to the RPV and to the SGs tube restraints. An overview of the acceleration and displacement time histories, through the CS to RPV and through RPV are showed in Figures 4(a) and (b) and 5 (a) and (b) respectively. The response spectra (Figures 6 (a) and (b)) into the frequency domain, indicates that if the soil is considered (Model A) the effect of embedment on structure lead to a reduction in structure response due to the increased amplitudes damping effect. The SSI coupling effect results from scattering of waves from the foundation and the transfer of energy from the structure due to structural vibration [7]. The system damping increases considerably with increasing the foundation embedment and the layer depth, especially for low-rise structures. Deeper is the stratum; greater is the influence of the embedment [8]. If the soil is not considered (Model B), the transfer function indicates that it was a decrease of seismic acceleration from the ground to the tube bundle where it is shown the transfer effect from the ground field to the RPV and from RPV to the SG tubes bundle restraints, highlighting, in this latter case, an amplification of the peak acceleration due to the in-plane internal structures flexibility.

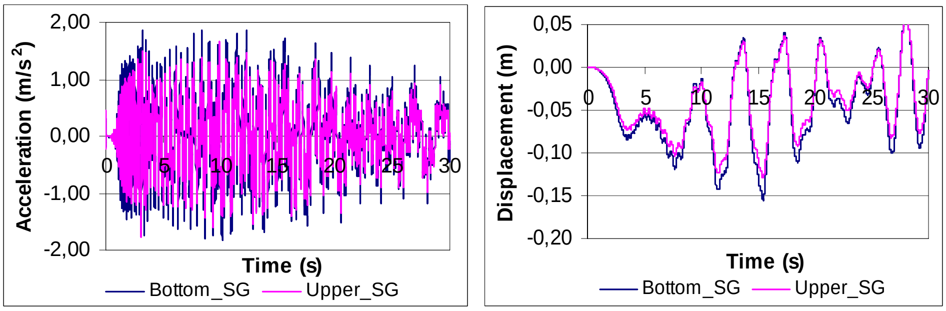


(a)

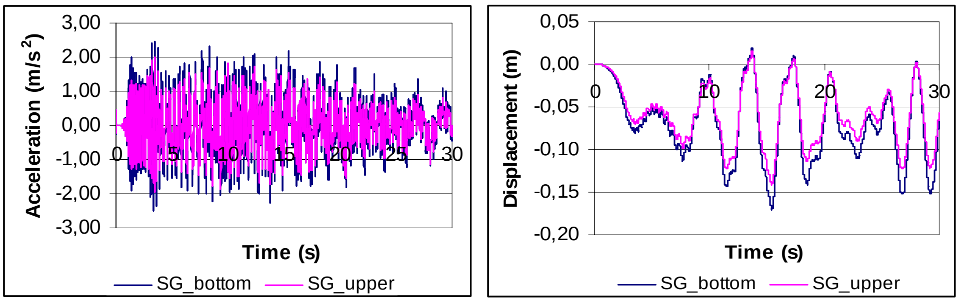


(b)

Figs. 4 (a), (b) – Acceleration and displacement to RPV- Model A and B

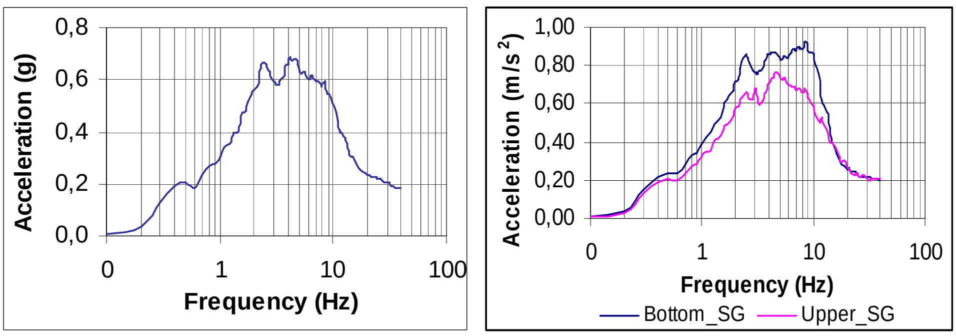


(a)

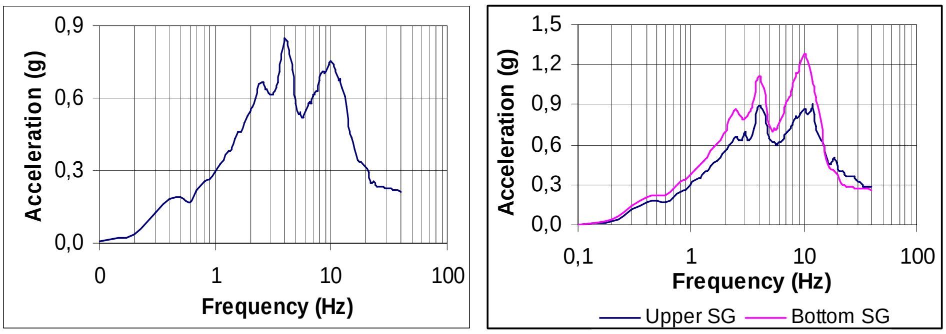


(b)

Figs. 5 (a), (b) – Acceleration and displacement to upper and bottom SG restraints Model A and B



(a)



(b)

Fig. 6(a), (b) – Response spectra to the RPV and SGs restraints- Model A and B

Analysing the response spectra it can be generally observed that there is a slight shift in the fundamental frequency of the building and a reduction in the spectral accelerations when using coupled models.

4. Conclusion

Analysis and design of the NPP structures involve considerations not only on the available geometry but also on the capacity of the most important structural members that transfer the seismic inertial loads from their application points.
An overview of possible seismic analysis approaches has been provided, with particular emphasis on the integral layout of the reactor coolant system accounting for the Soil Structure Interaction as well as Structure-Structure Interaction.
Analysing the calculated response spectra it can be generally observed that the SSI and adjacent building interaction results in rather slight shift in the fundamental frequency of the building. They also depend on the dynamic properties of each structure such as strength, rigidity, and modal characteristics. Soil-structure interaction gives rise to kinematic and inertial effects, resulting in modifications of the dynamic properties of the structure and the characteristics of the ground motion around the foundation. It was shown that the effects of foundation embedment and SSI are extremely important. They increase considerably the effective damping of the system relative to the damping of the structure alone.
On the base of these very preliminary analyses, the effects of the described alternative nuclear building in soil embedment have been considered in order to check the possibility to achieve an upgrading of the NPP geometry and obtain a feed back on the critical design features (if any).

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A NEW APPROACH FOR THE SEISMIC ANALYSIS AND DESIGN OF THE IRIS REACTOR

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ABSTRACT

The ambitious safety goal for the IRIS reactor requires that both internal and external events are duly considered and treated in the PSA, with the Safety-by-Design approach adopted to reduce the overall Core Damage Frequency (in the range of 10^{-8} events/year).
As far as the seismic event is considered, a suitable approach has to be pursued, trying to eliminate unnecessary conservatism. Therefore, an innovative methodology for the evaluation of seismic fragility, applicable both to conventional and innovative reactor concepts, has been developed and is here presented The two central elements of the procedure are the use of the Response Surface Methodology (RSM) for describing the influence, on structural response and integrity, of all the parameters, hypotheses and modelling criteria assumed as uncertain or random and a two-stage approach in the structural modelling of the reactor building.

1. Introduction

The large number of seismic PRA studies performed in recent years on nuclear power plants has shown that earthquakes are among the most important external events affecting NPP safety. In the framework of a seismic PRA, therefore, fragility evaluation of safety related components is a fundamental issue for risk and reliability assessment. The seismic fragilities of individual components and equipments, in fact, are combined with the seismic hazard, i.e. the frequency of occurrence of a given intensity of the earthquake motion, to evaluate the probability of different core damage states.
The main objective of the seismic fragility evaluation is to estimate the capacity and the related uncertainty of a component or a structural element relative to a given earthquake severity parameter, such as peak ground acceleration or spectral acceleration. This capacity is defined as the earthquake severity parameter value at which, for the considered component or structural element, the response exceeds the available mechanical resources, leading to failure.
Two sources of variability need to be incorporated in a structural fragility formulation: inherent randomness and uncertainty. Significant randomness affects many of the parameters describing the mechanical model adopted in structural analysis, such as material properties (including soil). In addition, the earthquake input motion is stochastic in nature, given the extremely large number of parameters affecting the seismic source, the source-to-site transmission path and the local ground response. Uncertainties, on the other hand, arise from analyst's lack of complete and accurate knowledge about models, methods for response analysis, limit-state formulation etc: uncertainties can be reduced, in principle, with detailed studies leading to more sophisticated techniques.

In this framework, a procedure able to reduce uncertainties as much as possible, thus reducing an important cause of unnecessary conservatism has been developed; the procedure is based on the use of the Response Surface Methodology for describing the structural performance, on a simulation approach for facing the random vibration issue and on the Monte Carlo Method for computing the failure probability.

2. Definition of the problem

Seismic fragility is defined as the probability of failure of a component (or structural element) conditioned to the severity (e.g. PGA) of the ground motion and can be written, for each PGA value:

$$p_f = \int_{\{g(\vec{x})<0\}} f(\vec{x})\, d\vec{x}$$

where $f(\vec{x})$ is the joint probability density function of all the variables $\vec{x}$ affecting load and response modelling and $g(\vec{x})$ is the performance function.

In the proposed procedure, the Response Surface (RS) Method is used to provide an analytical formulation to the $g(\vec{x})$ function to allow an efficient Monte Carlo evaluation of the seismic fragility.

Here the performance function of the component is assumed to be expressed in the simple “capacity (C) minus demand (D)” format:

$$g(\vec{x}) = C(\vec{x}) - D(\vec{x})$$

in which the vector $\vec{x}$ lists all input random variables affecting load and response modelling.

In the case of linearity of the analysis, the performance function is expressed as follows:

$$g(\vec{x}) = C^A(\vec{x}) - PGA \cdot y(\vec{x})$$

from which:

$$\tilde{g}(\vec{x}) = \left(C^A(\vec{x}) \big/ PGA \right) - y(\vec{x})$$

where $C^A(\vec{x})$ represents the acceleration capacity of the component under examination and $y(\vec{x})$ is the acceleration response of the structure, computed at the component supports, for a input time history having a unit PGA.

The RS method is used to find an analytical expression of $y(\vec{x})$ and the Monte Carlo method is subsequently employed to find the probability that the acceleration response of the studied component $D(\vec{x})$ exceeds its maximum allowable value $C^A(\vec{x})$, or, more in detail, that $y(\vec{x})$ exceeds the amplification ratio $C^A(\vec{x}) \big/ PGA$. In this way, the behaviour of the building is characterized in terms of the amplification ration of the peak ground acceleration.

3. IRIS test case

The methodology has been applied to the IRIS reactor as a test case.

The IRIS (International Reactor Innovative and Secure) plant is a medium power (~335 MWe) pressurized light water reactor under development by an international consortium which includes more than 21 partners from 10 countries, led by Westinghouse Electric Company.

IRIS plant development is aimed by a Safety-by-Design™ philosophy from the beginning, to reduce as much as possible both the probability of occurrence and the possible consequences of certain severe accidents caused by internal events. The IRIS power plant, in fact, presents some peculiar features, among which a compact design and an integral layout.

The IRIS safety-by-design™ has eliminated many initiators of internal events and consequently the internal events CDF has decreased by at least another decade when compared to passive light water reactors. Still, the external events initiators have not yet been addressed and thus at least for now, the CDF due to external events, such as seismic, is the preponderant factor in the total CDF for IRIS.

For performing initial tests on the procedure for fragility estimation, two structural models of the IRIS rector building have been set; a simplified one for performing the response computation and a refined model for the validation of the previous one by comparison of the eigenproperties. The simplified model will be also used to evaluate the performance of a seismic isolation system.

3.1 Refined Model

A refined model, encompassing a degree-of-freedom number of the order of 10^6, has been set. Modelling has been restricted to the structural system, which has been assumed as fixed at the foundation mat base.

The main structural elements of the building have been introduced according to the criteria summarized in the following. Reference is made to the elements and modelling options available in the ABAQUS structural analysis code.

Foundation mat: 8 node solid elements have been placed in 3 layers. Approximate element size is 0.65 m. Total thickness is 2 m.

Structural walls: All walls directly supported by the foundation mat have been introduced, by means of 8 node shell elements. Element size is about 1 m, while thickness is 1 m.

Shielding wall: It is a cylindrical wall having a thickness of 1.5 m and surrounding the containment system. Has been modeled via shell elements, with approximate size of 1 m.

Slabs: A thickness of 1 m has been assumed for all structural slabs. Shell elements have been used; size is approx 1 m.

Roofing system: a flat slab with stiffening girders was assumed; the slab was modelled via shell elements, 1 m thick. Girders were introduced with a 5 m spacing and a total depth of 3 m. They were represented by 3D beam elements; at each node, the centroidal element is connected to the corresponding shell node with a rigid link.

Containment: A 44.5 mm thick steel sphere is assumed, modelled by means of shell elements. The latter have an approximate size of 0.2 m. The lowest half of the sphere is supportedby a massive reinforced concrete structure, this latter modelled by means of 8 node brick elements. At the interface with the vessel shell, 20 node elements have been introduced. Perfect bond is assumed between steel and concrete at both sides (internal and external) of the shell.

The water contained in the vessel have been treated as a rigid body, which is attached to the shell via a distributed connection (DCC), equally subdividing the inertia forces developing in the water between the selected nodes on the vessel shell.

Sloshing effect in the suppression pools: the suppression pools which are located within the containment have been modelled as rigid bodies, connected to the reinforced concrete structure by means of a DCC connection.

The structure of the RWST pools has been modelled by means of shell elements. For modelling the water content and taking account, though in a simplified way, of the sloshing effect it has been assumed that the r.c. structure can be regarded as rigid in terms of interaction with the fluid.

3.2 Simplified Model

The passage from the refined to the simplified model is founded on both the simplification of some structural components and a suitable mesh optimization, as will be described in this section.

More in detail, the simplified model has been obtained by applying the criteria described in the following.

Walls, slabs and foundation mat: have been all modelled by means of shell finite elements.

Roof: the same discretization as the one adopted for the refined model has been chosen.

Containment: the lower part of the containment, encompassing the lower steel hemisphere and the surrounding reinforced concrete supporting structure, has been modelled as a rigid body. On the contrary, the upper part of the sphere has been represented via an equivalent two degrees of freedom inverted pendulum system.

Vessel: simplified modelling has been suggested by the observation of the lower vibration modes of the refined model, where elastic deformation is confined to a rather limited zone centered on the supporting skirt. On this basis the lower and upper part of the vessel shell have considered rigid,while the central portion of the shell and the skirt have been discretized via shell elements.

All the equipments located in the upper and lower parts of the vessel have been introduced as rigid masses, lumped at the corresponding centres of gravity. The steam generators, which are located along the deformable zone, have been considered as rigidly attached to the upper rigid portion of the vessel. For the water, the same criterion as used in the refined model has been maintained.

Foundation ground model : it has been here assumed that the foundation mat, when stiffened by all structural walls, can be treated as "quasi-rigid". This means that its deformability is taken into account in modelling it as a part of the structural system, but that it can be neglected with respect to soil-structure interaction effects.

3.3 Model comparison

In Table 1 the natural frequencies of the most significant vibration modes are compared for the two models. The modes considered in the table are two cantilever modes, in each direction, a vertical and a torsional mode for the building, two rocking modes and one vertical mode for the vessel and two horizontal translation modes for the containment.
As it can be noted, natural frequencies compare satisfactory for all most significant normal modes.

Mode	Refined model [Hz]	Simplified model [Hz]	Δ [%]	Mode description
11	5.41	5.60	+3.5	1 "cantilever" mode in y direction
12	6.42	6.66	+3.7	1 "cantilever" mode in x direction
15	8.22	8.54	+3.7	Torsional mode
21	12.31	12.70	+3.2	2 "cantilever" mode in y direction
31	15.56	16.4	+5.4	2 "cantilever" mode in x direction
24	13.55	13.52	-0.2	1 vessel rocking mode y-z plane
25	13.56	13.52	-0.2	1 vessel rocking mode x-z plane
92	31.38	30.79	-1.88	vessel mode translation in z direction
90	30.56	30.43	-0.42	1 containment mode y direction
91	31.12	31.06	-0.19	1 containment mode x direction
22	13.06	13.42	+2.75	1 global mode vertical translation
34	16.58	17.01	+2.6	2 global mode vertical translation

Table 1. Natural frequencies of the lowest vibration modes of the FE models

3.4 Estimation of the seismic fragility

Once realized the structural model, a set of 10 time histories has been obtained from a reference spectrum. Then, three random variables have been selected to represent the main sources of randomness for the computation of the response of an equipment located inside the vessel:

- a random variable describing the soil shear modulus G, with mean value of 200 MPa and c.o.v equal to 0.2;
- a random variable for the vessel damping factor; its mean value has been chosen equal to 0.03, and a coefficient of variation of 0.2 has been considered;
- a random variable to describe the viscous soil damping; more in detail, the ratio between the actual value and the mean value of each damping factor associated to foundation modes is considered, named δ, with a mean value of 1 and a c.o.v. of 0.2.

It has to be noticed, with respect to the last two RVs, that damping has been here treated in a simplified way. This was due to the difficulty to deal, by means of the software package at hand, with composite damping within modal superposition analysis. In the case here shown modal damping factor were directly stated and given in input by recognizing, with some engineering judgement, modes dominated by foundation or by vessel movements.
A second degree polynomial function has been chosen to express analytically the both the mean value and the standard deviation of the response $y(\vec{x})$, approximating the demand $D(\vec{x})$.
A Central Composite Design has been selected as an appropriate DoE for the RS generation.
Once found an analytical representation of the response $y(\vec{x})$, the probability of exceeding the given amplification ratio has been calculated, trough the use of the Monte Carlo method, for different values of the PGA amplification.

The Monte Carlo sampling technique have been used to select values of the input variables corresponding to their probability distributions. In correspondence of each set of randomly selected values of the three variables, the structural response has been calculated from the RS and then the condition $C^A(\vec{x}) \big/ PGA < y(\vec{x})$ is evaluated. The obtained results are shown in figure 1.

Procedures are presently under investigation to refine iteratively the RS and the fragility computation.

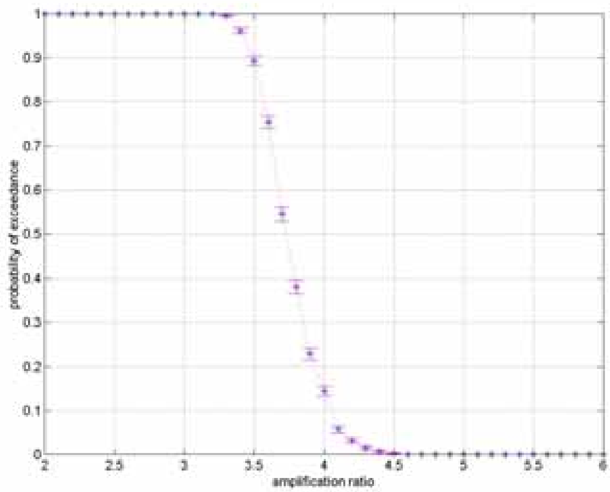

Figure 1. Probability of exceedance as a function of the structural amplification.

4. Conclusions

An accurate methodology to evaluate the seismic fragility of NPP components, trying to reduce uncertainties and conservatism of more traditional procedures, has been developed. Through the use of the response surface methodology, the proposed procedure offers a comprehensive and rational framework for performing parametric studies of the sensitivity of the structural response to the various randomness and uncertainty sources. As already hinted, the proposed procedure represent realistically the uncertainty in the input ground motion. The method is affordable, being the number of simulations to be performed rather limited; in addition it is reasonably easy to deal with, being the procedure straightforward and the tools proposed part of the background of many engineers.

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Status of research reactors for future nuclear research in Europe

ENC 2007 , Brussels, 16-19 September 2007

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Abstract :

During the 1950's and '60's, the European countries built several research reactors, partially to support their emerging nuclear-powered electricity programs. Now, over forty years later, the use and operation of these reactors have both widened and grown more specialized. The irradiation reactors test materials and fuels for power reactors, produce radio-isotopes for medicine, neutrographies, doping silicon, and other materials. The neutron beam reactors are crucial to science of matter and provide vital support to the development of nanotechnologies. Other kinds of reactors serve other specialized services such as teaching, safety tests, neutronic simulation…

The modifications to the operating uses and the ageing of the nuclear facilities have led to increasing closures year after year (ref. 1 and 2). Certain facilities are scheduled for closure, such as the last European fast breeder, Phenix, whose shutdown has been announced for 2008/2009. For others, safety re-evaluations have had to take place, to enable extension of reactor life. However, in the current context of streamlining and reorganization, new European tools have emerged to optimally meet the changing demands for research.

In 2006, in the neutron beam field, the ORPHEE reactor in Saclay returned to the "normal" number of operating days. The FRM2 in Munich has continued power escalation, extension work has continued on the ISIS reactor in Great Britain, and the work undertaken to bring the ILL reactor up to more demanding earthquake resistance levels has reached an end.

In the field of irradiation reactors, the RJH project has continued to advance. After 3 years with an engineering staff of 100, the definition of the reactor was reached in late 2005. The RJH reactor is now a mature pan-European project, selected by the European Strategic Forum for Research Infrastructure as vital to European interests. The construction phase was launched in 2006. The goal of commissioning is set for 2014.

For the European Research Area, the RJH reactor will play a major infrastructure role in the field of fission n, implementing international collaboration. In 2006, already 5 countries committed to the RJH project and contributed to construction. The European Commission supports the project, further strengthening the RJH Consortium.

With respect to Fast Reactors, the future-oriented work developed in GEN4 has demonstrated the strong interest in the fast reactor concept. Several countries, including Japan, Russia, India, China and South Korea, have expressed their preference for sodium-cooled reactors. This commitment has led to tangible actions well underway, including work at Monju for restarting, construction in China of an experimental sodium reactor for divergence in 2009, construction in India of the PFBR (1200 MWth), resumption of the BN 800 construction budget in Russia. In the United States, the announcement of the GNEP initiative includes a program for the ARR sodium reactor. And in France, the satisfactory operation of the Phenix reactor continues, with an availability factor above 80 %.
The Phenix reactor is scheduled to shut down 2008/early 2009, which would leave an absence of fast reactors in Europe as of the Phenix closure date. The 28 June 2006 french law on the

subject of sustainable management of radioactive matter and waste, calls, in its article 3, for the start-up of a prototype prior to 31 December 2020. This paper reports on the current status of the work and organization relating to the objectives for this prototype.

This paper also provides information on the status of other European projects, including the faisability study of an experimental fast reactor in Belgium (MOL) with a technology alternative to sodium, and the Pallas Project in Netherlands to replace HFR reactor in the future.

To conclude, the entire group of research reactors is undergoing significant change in Europe, and moving towards a more streamlined scenario providing for optimization of resources and plant characteristics, for the entire range of users.

EUROPEAN SITUATION

PRESENT SITUATION

In Europe, nuclear electricity plays an important role and will stay for the long term a very substantial part of the energy mix since it contributes to the energy security of supply to the reduction of greenhouse gas production and to the competitiveness of Europe.

Experimental reactors have been used to support many important fields of industry and research in Europe: safety, lifetime management and operation optimisation of current nuclear power plants, development of new types of reactors with improved resources and fuel cycle management, medical applications, material development for fusion reactor…

European experimental reactors have been built in the 60's and most of them have been operated on a national basis. With several Material Testing Reactors (BR2, Halden, HFR, LVR15, Osiris, R2, Siloe), and with demonstration reactors and prototype reactors (Rapsodie, Phenix, PFR, KNK II, and AVR, THTR) for developing the Sodium and Gas cooled reactor technologies, Europe has gained a worldwide leadership.

Some of these facilities are already stopped. The others will be more than 50 years old in the next decade and will face increasing probability of shut-down due to their obsolescence. Such a situation cannot be sustained on the long term.

Other research infrastructures are dedicated for fundamental research application by providing high quality neutron beams: reactors such as ILL-RHF (1971, Laue Langevin institute, France, Germany, Great Britain), ORPHEE (1980, France), FRM2 (2004, Germany). These facilities are commonly operated within European collaborations. In the field of matter science utilizing neutron beams, a set of effective and up-to-date facilities are available within Europe.

Toward renewing some key European Experimental Reactors (EER)

This survey has been discussed in depth and shared in Europe since 2002 (ref. 3).
A first generation of EERs, launched in the 60's, have provided the necessary support to industry and research in Europe (nuclear power plants, actinides management, medical applications, condensed matter physics…). The question is now to define and implement a consistent EER policy

- Meeting industry & public needs, keeping a high level of scientific expertise ;

- With a limited number of EERs, specified within a rational compromise between specialisation, complementarities and back-up capacities ;
- To be put into effective operation in the next decade.

Taking into account the needs of nuclear industry, the strategic importance of future GEN IV reactors developments, the advanced fuel cycle and the public health stakes, an European policy must include a mid-term roadmap encompassing:

- A high performance material testing reactor ;
- A reactor optimised for medical applications ;
- An experimental reactor for innovative fast neutron reactor technology development with capabilities related to test advanced fuel cycles.

RESEARCH REACTORS IN EUROPE

Irradiations in support of present and future nuclear reactors

Nuclear operators are bringing in operational changes and management measures to improve fuel economy and extend lifetime of nuclear power plants. While Utilities implement extended fuel burn-up, optimised fuel cycle, Safety Authorities assess this evolutionary situation through questioning about the safety behaviour of components and systems (Generation 2 and 3).

In parallel, a new generation of reactors Generation 4 will be developed to address key issues related to sustainable development objectives. A variety of aspects will be addressed in this context, regarding economics, safety, better use of natural resources, waste management, non proliferation issues and new utilization of nuclear energy (process heat, hydrogen). This new generation will require important technological advances in material and fuel science.

In the meantime closing the fuel cycle of presently used reactors remains an important topic of research to be addressed through partitioning & transmutation (P&T) and where minor actinides burning will also require reactor developments that can be commonly addressed through the Experimental Reactors (ERs) addressing the Generation 4 issues.

European existing MTRs and ERs

Existing European MTRs are ageing (see table) this leads to a growing discrepancy between their capabilities and the above industrial and public needs.

These reactors have gained a considerable international recognition for their operational flexibility and ability to set up collaborative programmes having broad international participation.
The R2 reactor shut down in 2005 illustrates how fast the situation can evolve in Europe.

Countries	**Reactor**	**Operation**	**Power (MWth)**
Czech Rep.	LVR 15	1957	10
Norway	Halden	1960	19
Sweden	*R2*	*1960-2005*	*50*
Netherlands	HFR	1961	45
Belgium	BR2	1961	60/120
France	OSIRIS	1966	70

With the JHR-CA FP6 coordination-action (2004-2005) and through the ongoing FP6 MTR+I3 (integrated infrastructure initiative), the European MTR community reinforce its scientific capacity by sharing the development of a new generation of experimental devices. There is a need for a high performance MTR to be implemented in Europe in the coming decade in the framework of worldwide competition. Its construction shall cope with the requirement to continuously supply irradiations. Therefore this new MTR will be networked with other facilities involved in material and fuel development programs (hot labs, other reactors).

As far as Experimental Reactors (ERs) in Europe (Rapsodie, Phenix, PFR, KNK II, and AVR, THTR) are concerned, all of them have been shut down except Phenix, a sodium cooled 250 MWe reactor started in 1973 and planned to be shut down by 2009. Although the fuel cycle and associated advanced fuel development as well as Generation IV systems are requesting fast spectrum experimental reactors, Europe will be left without an ER after 2010.

With the ongoing EUROTRANS FP6 Integrated Project (2005-2009), preparing a design of a dedicated experimental minor actinide burner (XT-ADS based on the MYRRHA project initiated by SCK•CEN), with FP6-GCFR (Gas Cooled Fast Reactor), and with FP6-ELSY (European Lead Cooled System), the European reactor research community and nuclear industry are integrating their efforts to put Europe in a position to decide by 2010 on the realisation of an Experimental Research Facility having a fast spectrum and able to address the closing of the fuel cycle. This facility can be conceived with the objective to demonstrate fast reactor technology and effective burning of minor actinides. It should be conceived to serve in a later stage as a fast spectrum irradiation facility.

Nuclear medicine is important for the health of European citizens with about 10 million medical procedures per year and 15 million in vitro analyses. This field is also important in terms of market for the pharmaceutical industry in Europe. For therapeutic and diagnosis activities, respectively 100% and 75% of the radioisotopes are produced by research reactors in Europe more particularly in HFR, BR2 and Osiris.

Perspective for an European Research Area on Experimental Reactors (ERAER)

The Jules Horowitz Reactor (JHR), a mature project meeting nuclear industry and public needs

The need for a new MTR in Europe has been assessed and confirmed by the Feunmarr FP5 thematic network (2002) (ref. 3) :

> « There is clearly a need as long as nuclear power provides a significant part of the mix of energy production sources »
>
> « Given the age of current MTRs, there is a strategic need to renew MTRs in Europe ; At least one new MTR shall be in operation in about a decade from now »

A high performance new MTR is to be built in Europe to meet the industry and public needs related to safety, competitiveness and innovations for the existing generations and the future systems.

More specifically, the JHR shall provide a secured experimental capability to support :

- Plant life time management & extension for Gen 2 & 3.

- Technological evolution for Gen 3, performances improvement.
- Fuel performance improvement and behaviour validation in incidental and accidental situation.
- Innovative fuel & material development for HTR and Gen 4 systems.
- The expertise in the field of nuclear energy, in association with other key infrastructures.

To meet these needs for the coming decades, JHR will be a high performance 100 MWth MTR providing high fast neutron flux in an under-moderated core (10^{15} n/cm²/s perturbed flux above 0,1 MeV) and high thermal neutron flux in the moderator ($5\ 10^{14}$ n/cm²/s). Compared to existing MTRs, JHR will offer advanced experimental capacities such as on line fission product measurements and dedicated cells to manage safety experiments with damaged fuel samples.

JHR – RESULTS IN 2006

2006 was the year the development phase was launched for the Jules Horowitz reactor (JHR). This development phase corresponds to the industrialization of the project, and includes detailed definition of the components and the preparation of the tender documents.

This development phase is an important transitional step between design and construction of the JHR, requiring several readjustments to adapt the teams to the construction objective.

Several files were sent to the Safety Authority in March 2006. These included the “DAC” (Application for Authorization to Create the Installation), the “DARPE” (Application for Water Intake and Discharge), and the “RPRS” (Preliminary Safety Report).

Provision of these documents enabled the public inquiry to be held, which was then completed in December 2006.

Processing of the Preliminary Safety Report also began, with the goal of holding a Permanent Group in 2007.

2006 was also the year of the signature of 6 bilateral agreements between the CEA and its partners, for the construction of the JHR. These agreements are the culmination of several years of European cooperative efforts to define the funding of the JHR, a process which will most likely be applied anew for other joint European research infrastructures in the field of EURATOM-fission.

The agreements were the basis for founding the JHR Consortium in 2006. The agreement proposed to the project partners in late 2006 stipulates their access rights as a function of their financial participation. This agreement was signed in spring 2007.

The Pallas project, securing the production of radio-nuclides for medical applications

In the Feunmarr 5th FP thematic network (2002), the market for radio-nuclide production for medical applications was assessed. Securing the European production capability was stated as an important public health stake.

Nuclear medicine is important for the health of European citizens with about 10 million medical procedures per year and 15 million in vitro analyses. This field is also important in terms of market for the pharmaceutical industry in Europe.

- Nuclear imaging techniques are powerful non-invasive tools providing unique information about physiological and biochemical processes. The gamma imaging activities represent a global annual turnover estimated at more than 1 billion €, and the demand grows each year by about 5%. This requires typically 20 isotopes among which the 99Tc part represents 70%. Other techniques like the positron emission tomography (PET) or Radioimmuno-assay represent an annual turnover of some 450 million €.
- For radiotherapy with radioisotopes, the overall annual turnover is roughly 250 million €. If cobalt therapy is an important but declining market, new technologies appear and are in a growing stage (gamma-knife surgery, alpha immunotherapy, brachy therapy…).

For therapeutic activities (resp. diagnosis), 100% (resp. 75%) of the radioisotopes are produced by research reactors.

The Petten site, in The Netherlands, integrates on the same site the reactor HFR, hot cells and medical-oriented production facilities. The Pallas project replacing HFR after 2015 will reinforce this medical application in Europe. A back up function from other European Research Reactors, especially JHR, is mandatory to secure the continous supply of the medical radioisotopes.

The Pallas power and main technico-economical characteristics are not yet finalised. This thermal power should enable both medical radionuclides production and some complementarity to JHR material programs.

The Fast Spectrum Project, addressing the next generation energy systems and actinides recycling

For the longer term, future nuclear energy systems should contribute, among other energy sources, to secure a sustainable energy development worldwide. Generation IV fast neutron nuclear reactors with the closed fuel cycle shall play a key role to optimise the use of natural resources and minimisation of long lived waste.

Accelerator Driven Systems (ADS) which are the coupling of an accelerator with a subcritical fast neutrons reactor, and the closed fuel cycle, are investigated as a possible alternative to critical fast reactors, in the framework of FP6 EUROTRANS IP (Integrated Project).

Future reactor research is addressing strong expectations related to energy and waste issues. Three basic needs are identified :

- New power plants to be built from 2010 on will use available technologies for Generation 3 and possibly high temperature gas cooled reactors for industrial heat (synthetic fuels…) or hydrogen production. Their development will mainly make use of generic MTRs.
- Development of Gen IV reactors for deployment at the horizon of 2040-2050 requires the realisation of a prototype unit of middle size around 2020 for the most mature technology which is SFR. Nevertheless one cannot secure access to fast reactor technology by considering a single technology only. An alternative track may be

requested at a longer term, being either GFR or LFR. A specific experimental facility will be needed to address technological development and demonstration of the chosen alternative track to support decision towards following step.

- With the closed fuel cycle, fast reactor technology will address a specific concern about waste management to reduce the actinide inventory to be managed in the long term by the above mentioned fast critical reactors and/or by sub critical fast reactors driven by an accelerator.

Based on R&D results there is an important milestone around 2010, to assess viability and performances of GFR and LFR and to decide for an experimental facility of European Interest in the range of 50-100 MWth, either critical fast reactor or ADS.

SCK•CEN volunteers to host this experimental facility on the site of Mol, Belgium.

This project should involve many European countries, who will define a technical roadmap for the selection of a second technology for fast neutron reactor and its implementation, in the framework of the European Sustainable Nuclear Energy Technology Platform (SNE-TP) which will launched in Brussels on September 21th 2007.

Prototype Sodium cooled Fast Reactor (SFR)

Among the fast reactor systems, the Sodium-cooled Fast Reactor has currently the most comprehensive technological basis, thanks to the experience gained internationally from the operating experimental, prototypes and commercial-size reactors (such as the Phenix plant in France, PFR in the UK or MONJU in Japan).

The technological basis gained from these reactors includes key elements of the overall reactor design, fuel types, safety, and fuel recycling. Innovations are sought for a Generation IV sodium cooled fast reactor in order to reduce the costs and to improve the safety. They involve design simplification, improvement of in-service inspection and repair, fuel handling, high performance materials, practical exclusion of high energy release in case of hypothetical severe accident.

Given the maturity of sodium cooled fast reactors, the next facility to be built in Europe will be a prototype reactor with a power conversion system of 250 to 600 MWe to demonstrate innovations with respect to existing sodium cooled fast reactors, and to pave the way for a first of a kind 4th generation commercial reactor.

CONCLUSION

Currently identified large infrastructures of European interest for nuclear research are :

- Jules Horowitz Material high performance Testing Reactor, identified in the ESFRI roadmap as a mature project to replace to a large extent Europe’s aging MTRs (over 50 years old) when it will come in operation in 2014. The JHR, launched recently with the support of several European countries and the European Commission, will In the short term support studies for Gen.-II and Gen.-III Light Water Reactors on ageing and life extension, safety and fuel performances, and support material and fuel developments for Gen.-IV reactors.

- The prototype sodium-cooled fast reactor with a power conversion system of 250 to 600 MWe, to be built through a research-industry partnership, together with a fuel fabrication pilot plant.
- A Fast Spectrum Experimental System with a power range between 50 and 100 MWth to support the development and demonstration of an innovative Generation IV technology.
- A reactor which should replace the High Flux Reactor (HFR) after 2015 as the main European provider of radio-nuclides for medical applications, and as such should be supported by the medical industry.
- Besides these major infrastructures, other experimental facilities are needed to support the technology developments and the safety studies or to demonstrate cogeneration technologies, depending on the market need for hydrogen or synthetic fuel.

Networking of existing facilities, and construction of new ones operated as “European user-facility” are essential for meeting the R&D needs described in the foregoing, for advancing the European Research Area (ERA), and for attracting a new generation of scientists and engineers to contribute to new challenging programs. Modern research infrastructures are essential for enabling the scientific community to remain at the forefront of nuclear energy science and technology, and to support the development of industrial innovations for nuclear reactors, fuels and fuel cycle.

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Session 18.1.5:

Research reactors

DEVELOPMENT OF HUMAN ENGINEERED GRAPHIC DISPLAY FOR THE HANARO RESEARCH REACTOR

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ABSTRACT

From June 2006, the upgrade of the operator workstation for the HANARO research reactor has been started to support modernization of the I&C system and to control the fuel test loop facility in the main control room. The upgrade was done on both the hardware and the software. The software for the monitoring and control was developed and the graphic display was added to the existing system. The new display was designed to meet the style guide that was developed for the designers by the human engineering specialists thru analyzing the control room environment and the hardware. The main policy for the upgrade is the consistency with the existing procedures and displays.

1. Introduction

A control system for the HANARO research reactor facility is under upgrade according to the 10 year refurbishment plan starting from 2002. The control system consist of operator workstations (OWS), networks, controllers and panels. The plan has 5 milestones. The first milestone was the replacement of operator workstations at 2002. The second one is the upgrade of OWS in order to combine the control and monitoring function of the fuel test loop (FTL) into existing system. The development for the FTL controls has been completed and the upgrade for the HANARO controls will be finished by November 2007. Other mile stones are the installation of the cold neutron source (CNS), the replacement of controllers, and the digitalization of reactor protection system. This paper describes the design of the human machine interface (HMI) of the operator workstations.

2. Control system

The upgrade of the operator workstation has been completed to control and monitor the FTL facility by the operators in the main control room. [1] The FTL control system is independent of other systems conceptually. But it is connected with the HANARO control system for sharing information. All controllers for the FTL facility are installed in a FTL control room. But these facilities of the control room are used only during a start-up of the system. Normal operation is performed at the main control room. So that, the control and monitoring functions are integrated to the existing control system. These controllers and networks are duplicated except for the data acquisition system. Each control network is connected to the existing HANARO data severs. The supervision network for duplicated workstations are connected to the tag severs also. The tag server acts as a bridge between the controllers and operator workstations. [2]

The digital control system of the HANARO is duplicated from input modules to output modules. The FTL control system is also fully redundant to ensure reliability. Two independent communication network link controllers are in each channel. The HANARO has a control local area network (LAN) and the FTL has its own control LAN. These two facilities are linked to tag servers. There are two tag servers in the main control room. The tag server is a PC for collecting information from the controllers and providing it to workstations and a data server. The architecture of the FTL and the HANARO is shown in figure 1. The time of all computers and controllers are synchronized with the GPS receiver. Other control systems like the CNS to be installed in 2008 will have the same structure with the FTL control system. [3][4]

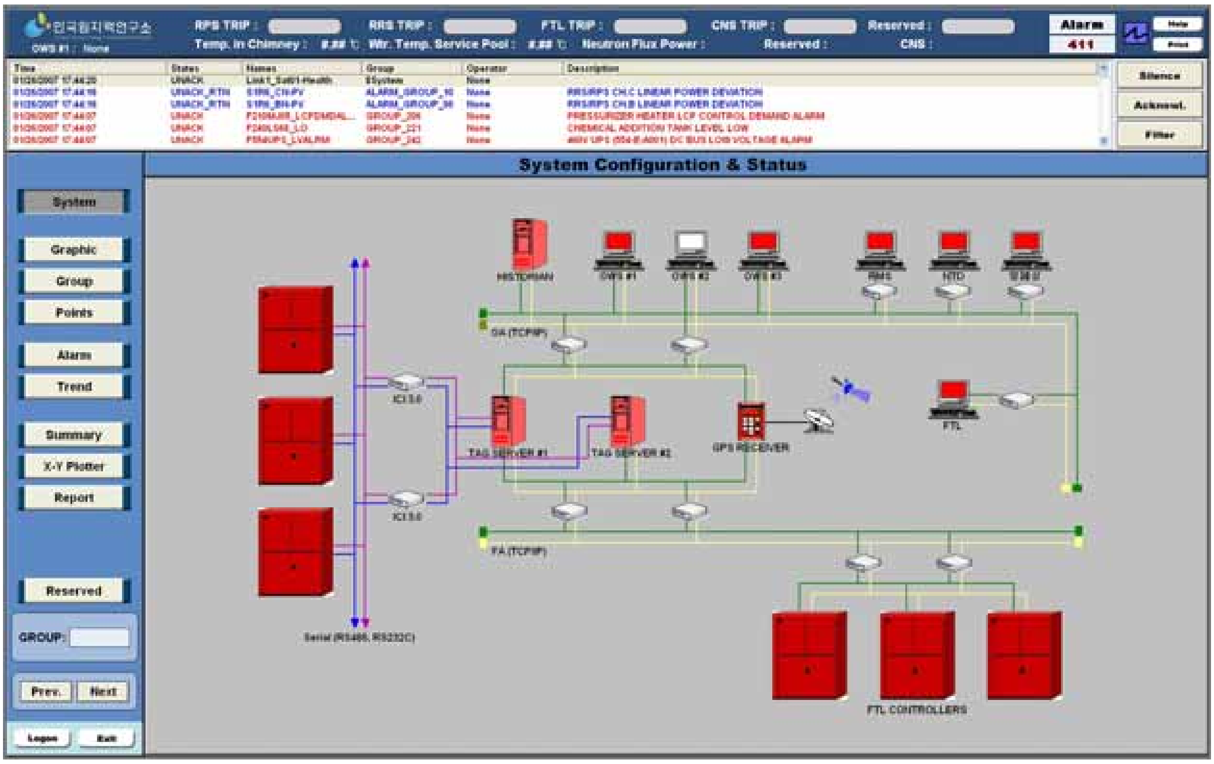


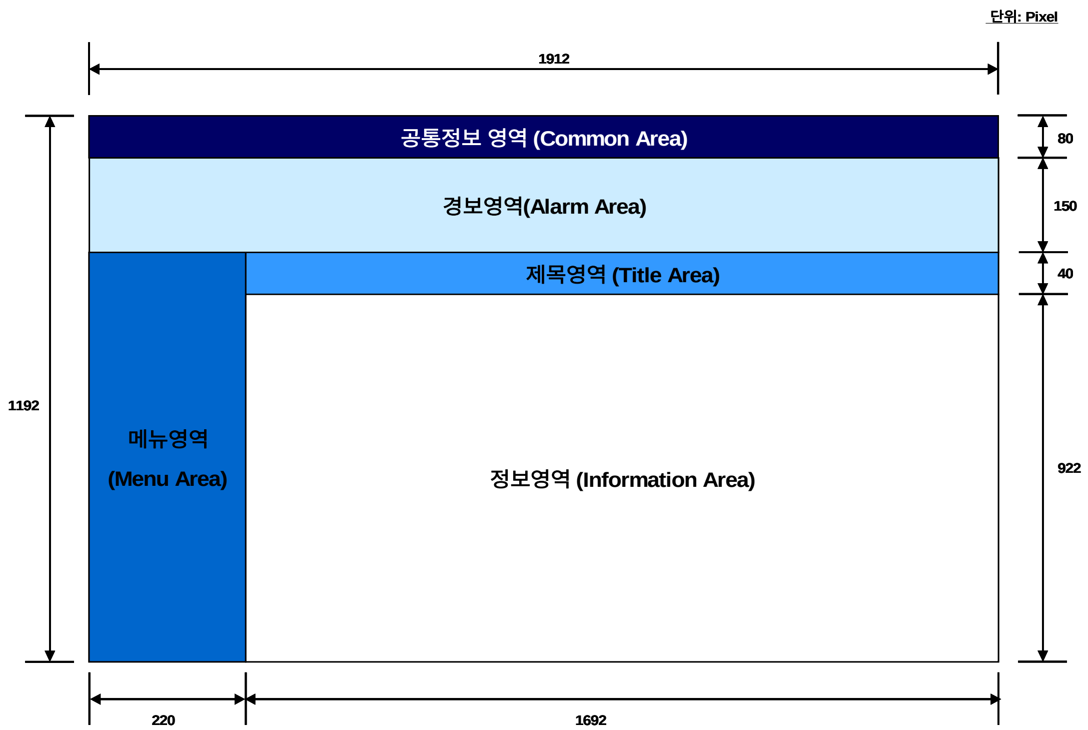
Figure 1. Architecture of the control system

3. Human machine interface

The old display of the operator workstation was not satisfactory in view of human engineering because it was designed to keep the same configuration as the old original workstation. The original workstation produced in 1992 has limitations in the hardware and software capability to perform various requirements. To overcome the limitations due to the old technology, an upgrade of the operator workstation for the FTL facility has been completed. [5] To comply with the human engineering requirements, a style guide was made first by considering the hardware, HMI tools, and operator requirements. Guide for developing of displays of the visual display units (VDU) consists of the general requirements and specific style requirements. The general requirements are representation, size, number, labeling, highlighting, and testing. The requirements for the specific displays were developed by the cooperation of operators and human engineering specialists for the following subjects;

- Configuration of the display area
- Dimension of each area
- Common information area
- Menu area
- Title area
- Alarm area
- Information area
- Icon

The 32 inch wide LCD, 1900pixels x 1200 pixels was selected as the VDU for this project. The area



allocation is figure 2.

Figure 2. Display area allocation

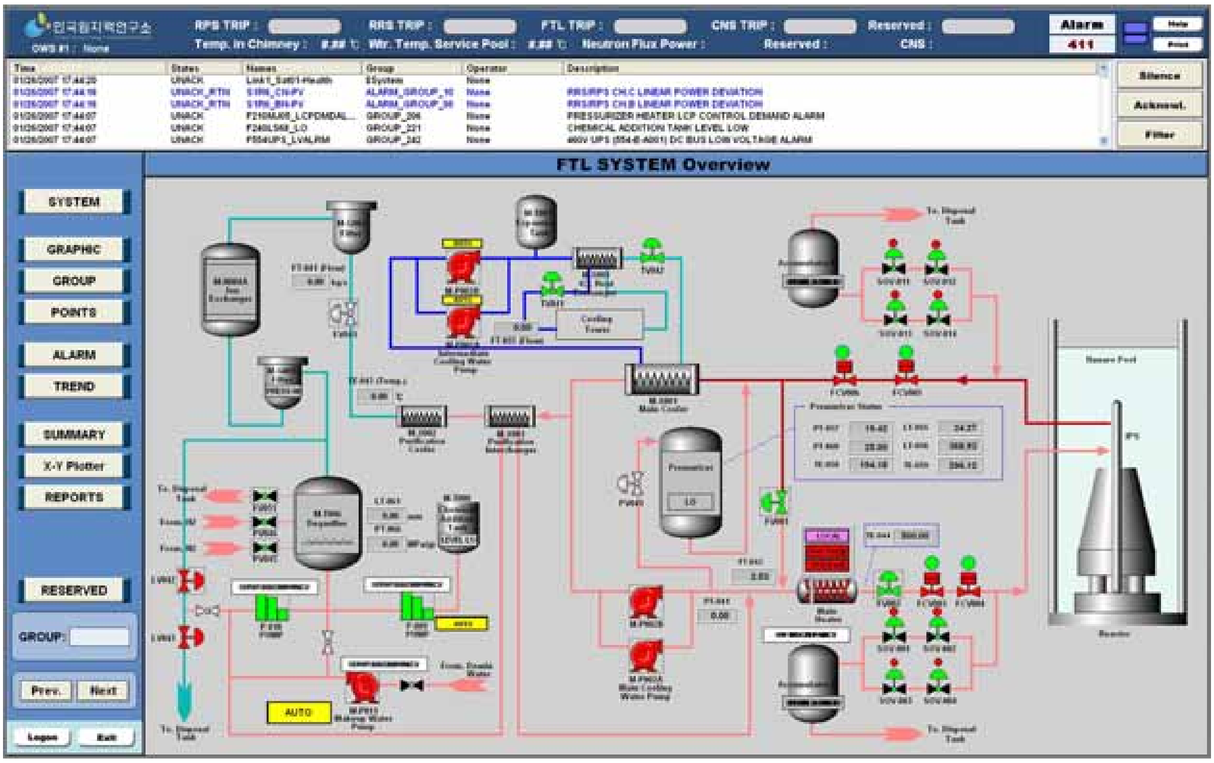


Figure 3. Sample display of the OWS

Menu area is at the left and key parameters are located at the top. These areas are always fixed during navigation for an operator to have quick access and recognition. According to the guide, graphic displays were developed first and were reviewed by the operators. The newly designed main display is shown in figure 3. Various new displays have been developed during this upgrade to support an operator convenient method. The table display shows many parameters in one page and acts like annunciators on the conventional panel. Other usual displays are trend, alarm, historical display, and X-Y plot. One of the new displays is a table display and is shown in figure 4.

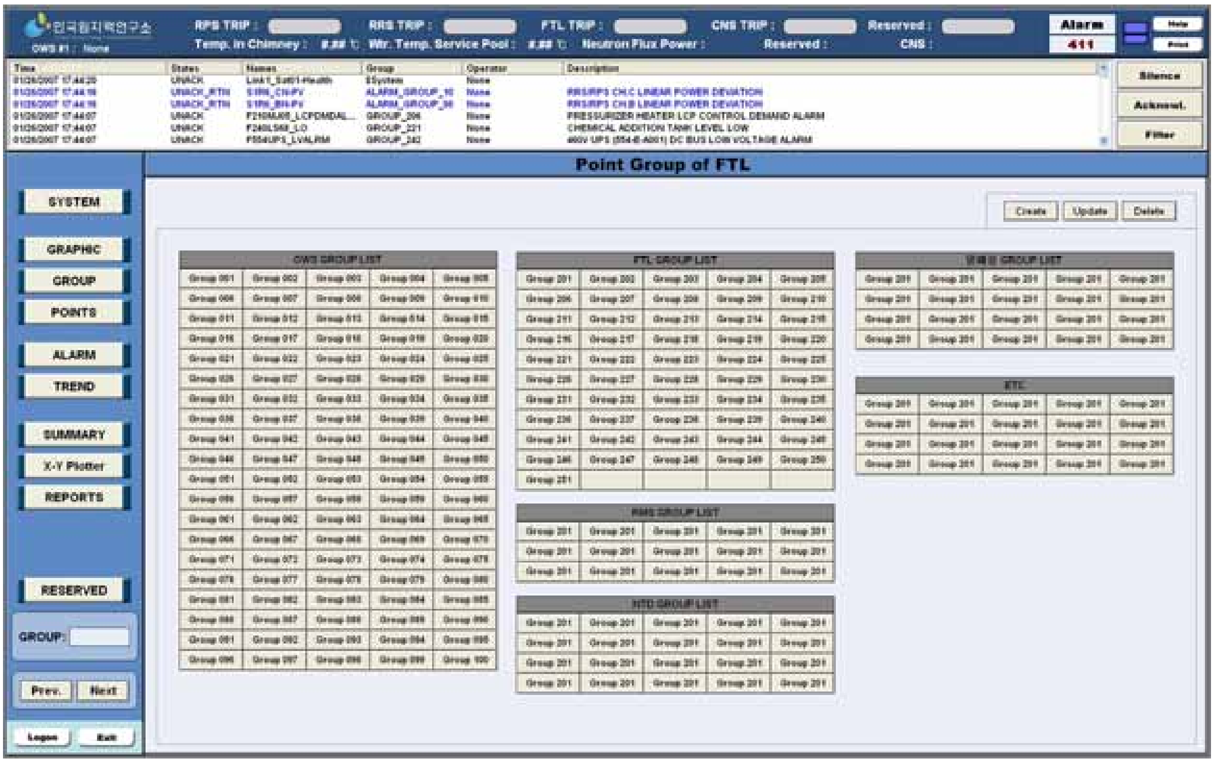


Figure 4. Table display

4. Conclusion

The second upgrade of operator workstations was finished to integrate the FTL control system into HANARO control system at the end of 2006. Requirements from the human engineering aspects and operators comments were incorporated in the style guide for designing of the displays of the visual display units. The human engineered graphic displays were developed and applied to the operator workstations. The new displays help the operators for controlling facilities with the realistic and physical sense easily.

5. References

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IMPROVEMENT OF INTEGRATED MANAGEMENT SYSTEM FOR THE RESEARCH REACTOR IN SOFIA

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ABSTRACT

The main purpose in establishment of Integrated Management System (IMS) is to guarantee safety operation of the nuclear facilities as well as to increase their exploitation effectiveness. To ensure the safety operation of the nuclear facilities the Bulgarian Nuclear Regulatory Agency (BNRA) has created requirements and norms to prevent potential nuclear incidents, overdose irradiations or terrorist attacks opportunities. The IMS of the Institute for Nuclear Research and Nuclear Energy (INRNE) has been developed in a way to create an environment which to guarantee the ways and means for: quality management according to ISO 9001:2000, environmental management in accordance with ISO 14001:2004, management of safety requirements of the BNRA, security and physical protection, management of the safe and health working conditions for the employees. The IMS is based on the concepts recommended by the IAEA: the entirety of work can be structured and interpreted as a set of interacting processes that can be planned, performed, measured, assessed and improved, and, those performing assessing work, all contribute in achieving quality and ensuring safety and environment. The INRNE IMS has been developed in the way to be continuously updateable and additive. The IMS will be added with new instructions, procedures and others formularies and documents, which will correspond to new activities arising during the reactor reconstruction. These instructions and procedures should be in agreement with the quality standard requirements as well as will be harmonized with the environment impacts aspects. The IMS developed on the base of state-of-the-art software ARIS in is developing the way to achieve ease in communication, visualization, possibility for assessment and continuous improvement.

I. Introduction

The Institute for Nuclear Research and Nuclear Energy (INRNE), Bulgarian Academy of Science, with its Research Reactor IRT is the biggest complex in Republic of Bulgaria for conducting research in the field of nuclear science, nuclear technology and energy and in the field of the monitoring of the radioactive influence on the environment.

The INRNE is the host and an operator of this institutes research reactor complex which is situated in Sofia city, and it is responsible for reactor systems maintenance and controls the permanently shutdown Research Reactor. INRNE is responsible also for the reconstruction activities, which include:

- Planning and preparation for partial dismantling of the Research Reactor (RR) equipment;
- Supply of equipment for the IRT Reconstruction;
- Planning of activities and responsibility for reactor modernization;
- Spent fuel control and management;
- IRT radiation monitoring for all implemented activities;
- Radioactive waste (RAW) management and control, for the RAW generated during the reconstruction process.

An Integrated Management System has been elaborated on ISO 9001:2000 requirements for quality management [1], ISO 14001:2004 for environmental management [2], and safety requirements of the Bulgarian Nuclear Regulatory Agency [3], governmental requirements for occupational health and safety and security. The IMS application guarantees the safety of activities as well as reduces of the radiation influence on the environment within the governmental norms. It helps to achieve maximal effectiveness and quality of the performed activities.

II. Processes

The processes, going along with various activities performed on the institute's Research Reactor are described in so called procedures. This procedures are developed in the way to give you an opportunity for simple process control following the Deming's cycle trough plan, do, check, action stages. [4] For the correct functioning of each process a process measuring indicators are formulated. In practice, indicators define the limits within which we would like a certain process to be managed. They specify the qualitative realization of activities and they are serving like comparative measuring process index in the time. For example: they are comparing the process execution taken alone. The measurement periodicity depends from indicator.

Every single process is represented by a complex of activities. A responsible person delegated with peculiar obligations and competence, is appointed for each activity. Work instructions for each activity have been developed together with appropriate formularies where the results of the activity are recorded. These records are applied as documentary evidence for the fulfilment of the safety requirements in front of the Regulatory Body, as well as in case of the internal and external audit, or to give evidence if there is civil interest.

Some of the most important and specific processes and their corresponding procedures applied for our RR management are: "The IRT nuclear and irradiation safety insurance"; "IRT Research Reactor reconstruction management"; "The radiation monitoring insurance on the Nuclear Scientific and Experimental Centre (NSEC) site"; "Radioactive waste management"; "Preparation of documentation for licence and permissions".

There are different instructions attached to a procedure. For example, for the process "The IRT nuclear and irradiation safety insurance" there are instructions as "Instruction for distillate water full up in reactor pool and water pool spent nuclear fuel (SNF) storage", "Instruction for the water technology control in water pool SNF", "Instruction for activity on duty for the mechanic when IRT special sewage is used" etc. The performance of these activities is documented in the records, which proved the IRT safety assurance.

Other basic process is "The Research Reactor reconstruction management". The realisation of that process is graphically shown on Fig. 1 and it is developed in following basic procedures:

- Management of "Investigations, analysis and design of the Research Reactor with 200 kW power";
- The reactor equipment partial dismantling management;
- The IRT reconstruction work project implementation management;

The detailed schedule of the Reactor Reconstruction activities is presented in the table on Fig. 2.

All processes, that are carried out on the NSEC site are accompanied with permanent radiation control and monitoring. These activities are described in the procedures "Securing the radiation monitoring on NSEC site" and "Providing safety radiation conditions for the personal, working with radioactivity sources, on the NSEC site". Monitoring programs and instructions are implemented and being strictly documented by records in appropriate formularies.

The „Radioactive waste management" process includes: RAW generation, collecting, sorting, minimization, and storage up to final transportation from the NSEC site. These activities control is realized according to Bulgarian legislation requirements, which are taken in consideration in the procedure "RAW management". The records performed under this procedure give an account on the RAW quantity and it movement.

To make easier the management of the processes as well as to evaluate them, and to give us a possibility for continuous improvement of the IMS (and in this way to satisfy the ISO 9001:2000 requirements), the ARIS software has been used [5].

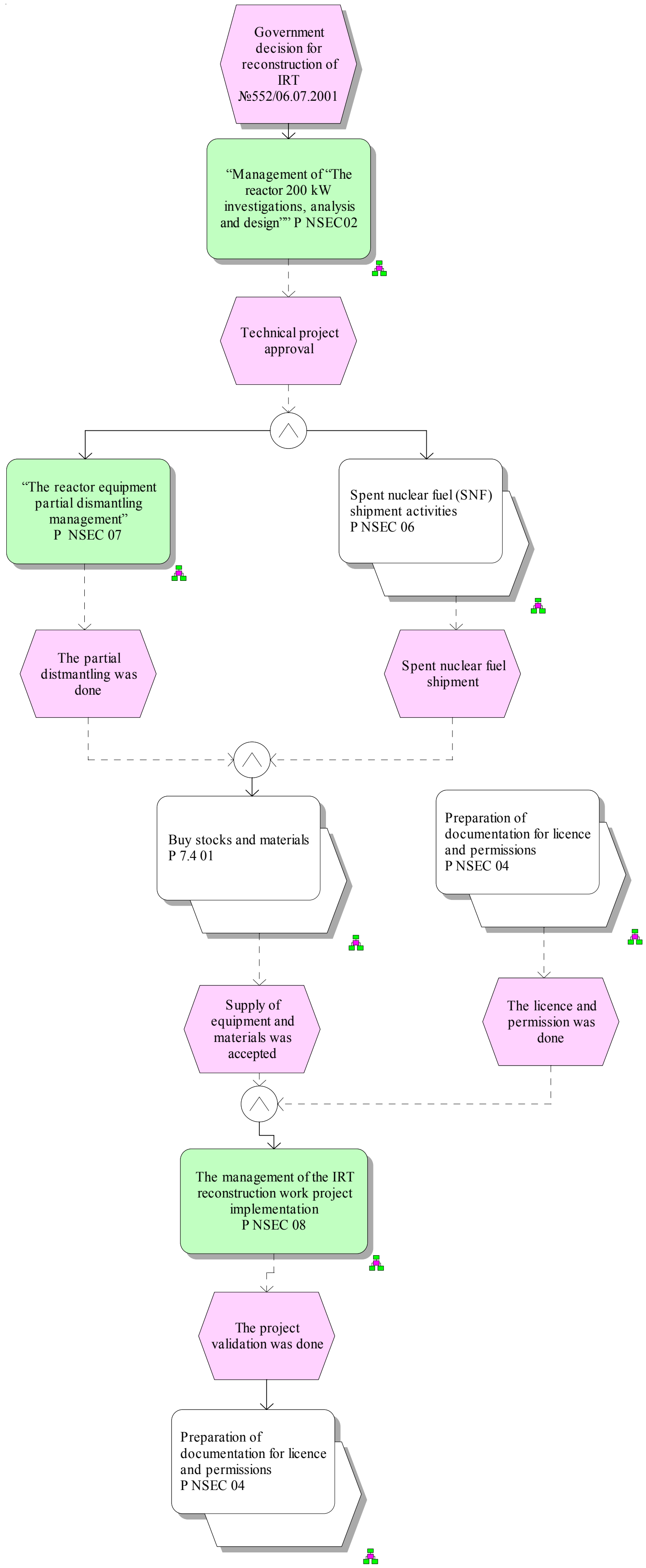
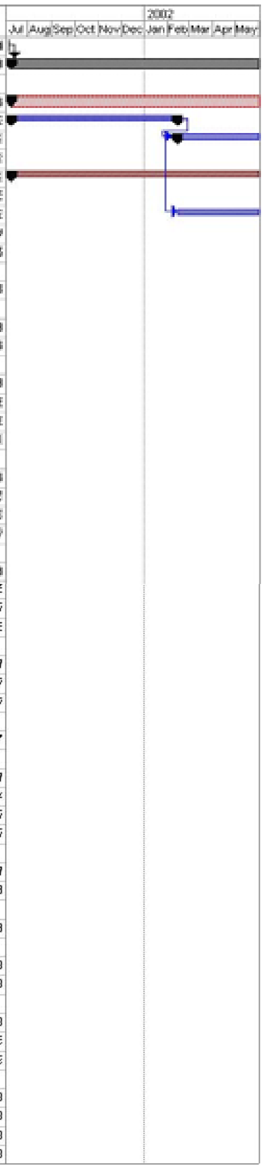


Fig.1. The process chart “The IRT Research Reactor reconstruction management”



ID	Task Name	Duration	Start	Finish
1	**1. Governmental decision for reconstruction of IRT №552/06.07.2001**	**1 day**	**Fri 06.07.01**	**Fri 06.07.01**
2	**2. Realization of the Governmental decision №552/06.07.2001 for reconstruction of IRT**	**1952 days**	**Mon 09.07.01**	**Tue 30.12.08**
3				
4	**3. Elaboration of a technical project for the IRT-2000 reconstruction into IRT-200**	**1333 days?**	**Mon 09.07.01**	**Wed 16.08.06**
5	*- Elaboration and approval of a technical assignment for the IRT-2000 reconstruction*	*157 days*	*Mon 09.07.01*	*Tue 12.02.02*
6	*- Tender procedure according to LPP for selection of the reconstruction project contractor*	*117 days*	*Wed 13.02.02*	*Thu 25.07.02*
7	*- Preparation and entering into a contract for designing with the Interatom consortium*	*3 days*	*Fri 26.07.02*	*Tue 30.07.02*
8	*- Documentation preparation and obtaining of design permission*	*378 days?*	*Mon 09.07.01*	*Wed 18.12.02*
9	*- Design activities*	*184 days?*	*Thu 19.12.02*	*Tue 02.09.03*
10	*- Selection and signing of contract with project consultants /EQE & Belgatom/*	*228 days?*	*Wed 13.02.02*	*Fri 27.12.02*
11	*- Safety analysis report, Version I and Version II*	*295 days?*	*Wed 03.09.03*	*Tue 19.10.04*
12	**3.1. Accounting for the IIRA recommendations and addition to the Safety analysis report, Version III**	**282 days?**	**Wed 20.10.04**	**Thu 17.11.05**
13				
14	**3.2. Technical project of the IRT-200 submition and approval**	**196 days?**	**Fri 18.11.05**	**Fri 18.08.06**
15				
16	**4. Additional activities, accopaning design procidure**	**1439 days?**	**Mon 30.12.02**	**Thu 03.07.08**
17	**4.1. Contract realization with consultants EQE and Belgatom**	**1045 days?**	**Mon 30.12.02**	**Fri 29.12.06**
18				
19	**4.2. Supply of equipment**	**1101 days?**	**Mon 12.01.04**	**Mon 31.03.08**
20	* Preparation of tender packages for equipment supply	710 days?	Mon 12.01.04	Fri 29.09.06
21	*Tenders for supply of equipment	542 days?	Thu 02.12.04	Fri 29.12.06
22	*Supply of equipment according to the implemented tenders and signed supply contracts	782 days	Fri 01.04.05	Mon 31.03.08
23				
24	**4.3. Monitoring equipment supply for the IRT site and radiation control for the IRT building**	**414 days?**	**Mon 31.01.05**	**Thu 31.08.06**
25	**Procedure for supply under the PHARE program*	*195 days?*	*Mon 31.01.05*	*Fri 28.10.05*
26	**Contract for supply and implementation*	*216 days?*	*Mon 31.10.05*	*Mon 28.08.06*
27	**Technical assistance for education in R & M equipment*	*264 days?*	*Fri 01.09.06*	*Wed 05.09.07*
28				
29	**4.4. General plan for partial dismantling (GPPD)**	**1108 days?**	**Thu 01.01.04**	**Mon 31.03.08**
30	*Elaboration of a general plan for partial dismantling	500 days?	Thu 01.01.04	Wed 30.11.05
31	*Development of a detailed partial distmantling plan	433 days?	Thu 01.12.05	Mon 30.07.07
32	*Partial distmantling activities execution, building and construction works realization	108 days?	Thu 01.11.07	Mon 31.03.08
33				
34	**5. Elaboration of a Detailed Design for the IRT-2000 reconstruction into IRT-200**	**358 days**	**Wed 15.02.06**	**Fri 29.06.07**
35	*- Detailed design preparation activities*	*293 days?*	*Wed 15.02.06*	*Fri 30.03.07*
36	*- IRT-200 detailed design acceptance*	*65 days?*	*Mon 02.04.07*	*Fri 29.06.07*
37				
38	**6. Spent nuclear fuel (SNF) shipment activities /detailed timetable/**	***947 days?***	***Mon 15.03.04***	***Tue 30.10.07***
39				
40	**8. Fresh Fuel**	**811 days?**	**Mon 22.11.04**	**Mon 31.12.07**
41	* Return of IRT-2M fuel	25 days?	Mon 22.11.04	Fri 24.12.04
42	*Selection and contracting of fresh fuel	526 days?	Mon 27.12.04	Mon 01.01.07
43	*Fresh fuel IRT-4M supply	149 days?	Wed 06.06.07	Mon 31.12.07
44				
45	**9. Documentation preparation and obtaining of construction permission**	**110 days?**	**Mon 02.07.07**	**Fri 30.11.07**
46	**10. Equipment assembly and construction works realization**	**239 days?**	**Wed 02.01.08**	**Mon 01.12.08**
47				
48	**11. Request and supply for berryllium assemblies provided by IAEA**	**631 days?**	**Mon 03.07.06**	**Mon 01.12.08**
49				
50	**12. IAEA contract "BUL 04/014" for technical assistance realization (specializations,technical assistance, expert missi**	**1057 days?**	**Tue 15.03.05**	**Wed 01.04.09**
51	**13. Staff qualification improvement /education on department programmes, seminars etc./**	**1057 days?**	**Tue 15.03.05**	**Wed 01.04.09**
52				
53	**14. Physical protection on the IRT site**	**1458 days?**	**Mon 01.09.03**	**Wed 01.04.09**
54	*- Physical protection development on the IRT site*	*718 days?*	*Mon 01.09.03*	*Wed 31.05.06*
55	*- Physical protection on IRT site maintenance*	*696 days?*	*Fri 01.12.06*	*Fri 31.07.09*
56				
57	**15. Environment impact assessment report for the IRT-200 commissioning**	**198 days?**	**Wed 30.04.08**	**Fri 30.01.09**
58	**16. Obtaining a permission for the IRT-200 commissioning**	**131 days?**	**Fri 01.08.08**	**Fri 30.01.09**
59	**17. Commissioning and first criticality**	**43 days?**	**Mon 02.02.09**	**Wed 01.04.09**
60	**18. Obtaining of operation Licence**	**43 days?**	**Mon 02.02.09**	**Wed 01.04.09**

Fig.2. Detailed schedule of the Reactor Reconstruction activities

III. Results

From general point of view the good practice is the approach were we plane and conduct training to achieve qualification level. To perform this we can conduct training courses not only concerning nuclear technology and energy knowledge but conducting seminars on IMS also. The necessity, to conduct education courses, is documented in the form – "Training Application". After the end of the education course, the trained person has to write down report.

At daily work we ran into difficulties, connected with establishment and documentation of the suggestions for improvements in separate processes. That's why periodically we are refreshing our courses by the ISO standards requirements.

The INRNE management policy is directed to guarantee high quality developments implementation, which are in agreement with modern world trends of continuously refreshing knowledge, of long standing experience and cooperation with leading European and International institutions. INRNE have a purpose to satisfy the community needs for development and maintenance of the nuclear science, to create necessary knowledge and skills for development of applied methods and research in the area of nuclear technology medical physics and nuclear industry.

IV. Conclusion

The IMS is based on the concepts recommended by ISO standards and the IAEA prescriptions: the entirety of work can be structured and interpreted as a set of interacting processes that can be planned, performed, measured, assessed and improved, and, those performing assessing work, all contribute in achieving quality and ensuring safety. The IMS has been developed in the way to be continuously refreshable and additive. That's why IMS will be added with new instructions, procedures and others, which will correspond to new activities arising during the reactor reconstruction.
The IMS gives strong level of certainty in the Research Reactor safety assurance, environmental protection, reactor physical protection and secure the normal personal working conditions.

V. References

1. Standard EN ISO 9001:2000, 2000
2. Standard EN ISO 14001:2004, 2004
3. Act on the safe use of nuclear energy, Promulgated in the State Gazette No. 63 of June 28, 2002.
4. Kaoru Ishikava, Introduction in quality control, 1989
5. ARIS – software product ARIS toolset, 1997 – 2005, IDS Scheer AG

EVALUATION TESTS OF THE TELEROBOTIC SYSTEM MT200-TAO IN AREVA- NC/LA HAGUE HOT CELLS

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ABSTRACT

The MT200-TAO system for hot-cells, first presented at ENC 2005, transforms a conventional wall-transmission telescopic mechanical telemanipulator (extension 4 m; capacity 20 kg), into an electrical computer-assisted telerobotic system. The working volume is extended to a full hemisphere (a volume approximately three times larger than the original telemanipulator) and operators experience a level of ergonomics and dexterity which sets new standards. This innovative system has been successfully evaluated in a "cold cell" of AREVA/NC/La Hague in order to prepare a complete evaluation in an active hot-cell which is currently in preparation. This paper summarizes the architecture and the components of the system and details the non-active evaluation phase as well as the training of operators and the ergonomical revolution allowed by this system. Finally we describe the planned active mission in the ACR workshop and the awaited benefits for the end-user.

1. Introduction - Project background and objectives

AREVA/NC/La Hague is a reference plant for remote handling technology with 600 telemanipulators installed and 150 operators and is thoroughly involved in developing new tools to improve its plants as well as its workstation ergonomics. Any advantage offered by a new system that can replace an existing one without modifying the facility makes it possible to increase performance in the short term and to prepare changes in new plants in the medium term.

In this perspective, several attempts have been made to actuate an existing disconnectable telemanipulator from the cold side. In the past, CEA and COGEMA have cooperated on a project named "MT200 Numérique" to transform a MT200 into a robot to perform repetitive tasks.

The MT 200 TAO system is the result of the fruitful cooperation between the Interactive Robotics Unit of CEA LIST and AREVA NC. It was designed to address the following specifications:

- guarantee similar performances than the original MT200 telemanipulator
- increase working volume allowing ceiling access
- improve workstation ergonomics of the existing MT200 telemanipulator
- allow distancing of the operator from the controlled zone in certain workstations potentially exposed to contamination or high radiation rate
- ensure safety for difficult tasks and reduces of operator fatigue when located within the control zone
- allow playback of some repetitive tasks that do not require force feedback (robotic mode)

2. Description of the system

This system has been formerly described in more detail [1] but it is useful here to recall its basic characteristics.

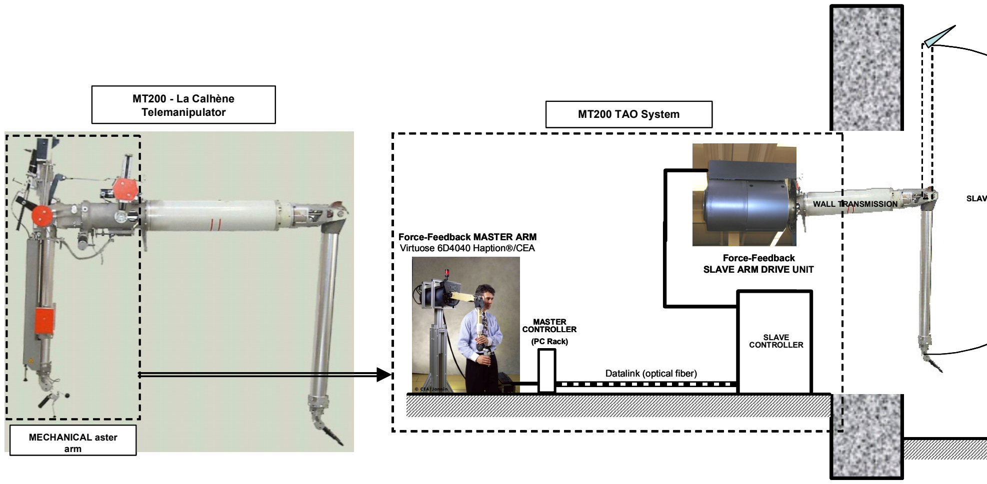

Fig. 1 – Schematic principle of the MT200 TAO system

Left picture: The original MT200 mechanical telemanipulator is disconnectable in 3 parts: the master arm, the wall transmission and the slave arm.

Right picture: The MT 200 TAO system functionally replaces the mechanical master arm. The wall transmission and the slave arm are those of the MT200 La Calhène® a design originating from the early 80's. The slave drive unit can be fitted to any La Calhène® wall transmission model in less than an hour and therefore to any telescopic slave arm produced by this constructor. The compact force-feedback master arm is a Virtuose 6D/4040 constructed by ®Haption on a CEA-LIST patented design using ball screws. It can exert a permanent effort of approximately 40 N.

The TAO 2000 software also a proprietary software from CEA-LIST, allows force feedback master slave mode between the two kinematically different arms in Cartesian coordinates. It also brings a set of powerful functions: Exact balancing and force surveillance, tool weight compensation, adjustable velocity and effort ratio (independently), robotic modes, virtual mechanism modes, and automatic pursuit of the gripper by a telesurveillance camera.

3. Performances requirements and design

Teleoperation is an extension of telemanipulation which has been historically and is still today the essential way to remotely manipulate objects that can't be handled directly because of their potential danger. The operator is always in view of the task (either directly through a window or indirectly through a television system) and master-slave telemanipulators used for this purpose have backdrivable transmissions to ensure that efforts are transmitted whether they are applied at the master or at the slave (property generally called bilaterality). Mechanical telemanipulators (or electrical servomanipulators) are also built with the same number of axis (rotational or translational) and the same architecture.

A teleoperator (or a telerobot) covers a wider variety of master and slave also able to perform master-slave telemanipulation with the same above mentioned properties but through computer control. Thus the slave arm can also playback trajectories just as a robot does and it may also be installed on a transporter (a fixed structure or a mobile platform). Powerful assistance function can be implemented such as "virtual mechanisms" which consists in constraints in position (or force) in certain directions to help guide the tool (for example, to keep it normal to the surface). It is also possible to coordinate more than one master-slave at the same time in various combinations. Master and slave can be heterogenous (mixing rotation and translation), even redundant (more than 6 axis) and therefore more freely optimized to their tasks. The master station may consist in a force feedback joy-stick, a master arm or an exoskeleton. The slave arm may be a dedicated design (like MT200 TAO) or an industrial robot equipped with joint torque sensors or generalized 6 axis force sensor to compensate friction.

Furthermore, telerobots just like telemanipulators, must be "transparent" enough for the user, a combination of low friction and low inertia.

In the case of the MT200 TAO, thanks to a careful design of its drive unit, the operator experiences a similar force sensivity (including for the tong) than with the original MT200 telemanipulator, combined with a much lower inertia. This objective had to be imperatively met in order to pass the acceptation test by La Hague's pilots, trained to daily work with conventional mechanical telemanipulators (principally MT200/La Calhène and A100/Wälischmiller). Moreover force feedback is here performed without any force sensors which is a guarantee of simplicity and reliability for the system.

4. Teleoperation: An ergonomical challenge for AREVA/NC

However, the introduction of teleoperation represents for AREVA/NC La Hague a specific and major challenge in terms of ergonomics due to the cumulated experience on conventional telemanipulators represented by the whole pilot staff (about 150 persons).

Moreover, teleoperation implies here to work in Cartesian coordinates. For the pilot it first means that the handle and the tong are no longer visually aligned. This phenomenon is usually perceived as detrimental for everyone having tested a remote controlled model.

In addition, as a consequence of energy input in the system, force and speed ratios may be selected independently leading to a different behaviour than a simple mechanical transmission.

We can then conclude that a successful introduction of today's teleoperation technology in the plant is highly dependant on the perception of the new ergonomical trade-off offered by the system.

To better understand evaluations, we need to recall that for the operator teleoperation is a sensory motor activity involving visual and force feedback:

- Visual feedback from the cell takes place via a shielding window (direct vision) and/or via cameras (indirect vision)
- Force feedback occurs via the master arm pistol grip handle completed with a trigger.

5. Validation program of the system

Validation will be fully assessed after the termination of a two phase evaluation process. The first phase, finished in 2006, involved testing in a cold cell. Operators were trained to use the machine so that they could assess usability and make suggestions for improvement. After incorporation of the suggested changes, the second phase, consisting in hot cell operations, is now being scheduled (2007 – 2008) under different work conditions and types of workshops such as:

- operating conditions in the alpha waste conditioning facility,
- maintenance conditions in the resin conditioning facility,
- repair conditions in the vitrification facilities,
- exceptional operation conditions (workshops to be defined).

5.1 Cold-cell testing phase

Teleoperator experts first underwent a 2 day training course. This "practical" course was based on exercises involving routine works to be carried out, enabling them to get gain experience with the new tool. They also used an ergonomical test bench, a device that allows the operator to perform several standard teleoperation tasks, to record the operation time and measure the exerted forces, the latter being an important parameter to evaluate the impact on the environment.

Fig. 2 –MT 200 TAO at AREVA/NC La Hague's training and evaluation cold-cell

Left picture: slave arm drive unit replacing the conventional mechanical master arm and its counterweights
Right picture: slave arm in a "work at the ceiling" configuration

The training proved conclusive and the new tool received the operator's approval at the end of the course. Trained operators expressed their desire to use the MT200 TAO in actual work situations. An evaluation of the prototype was then carried out using a questionnaire with the criteria (equipment, activity, etc.) that are important to teleoperation at the end of this course. Recommendations for improving the tool and a report were given to the AREVA NC project manager. Thanks to the flexibility of the software, most recommendations were followed by corrections and an important two functions have been finally added: an automatic control of the telescopic offset and a "screwdriver" function. This means that the pilot no longer needs to regularly adjust the length of the telescopic movement using handle knobs and can thus more easily concentrate on his task.

After this first phase of evaluation, positive conclusions regarding the benefits for both the plant management and the operator have already been drawn in connection with each technical characteristics or function of the system. They are summarized and classified in Tab 1.

	Technical features of the MT200 TAO teleoperator	Performance/Quality benefits	Mental workload benefits	Physical benefits
Replacement of mechanical transmissions by electrical transmission	**Replacement of the mechanical transmission by a flexible ombilical**	Direct vision improved as not linked to the master arm		Improved operators' position (facing the shielding window)
		Distanciation of the operator from the hot-cell		Reduction of exposition to: irradiation, contamination Elimination of electrical shock risk
	Hemispherical slave working volume	Increased working volume (3 times greater) + Continuous free displacement	Decrease attention to joint limits management	
	Displacement homothetical ratio	Effective use of slave working volume		
	Force homothetical ratio	Increase operator's sensivity/Dexterity	No necessity to anticipate parasitic efforts and their consequences on trajectory excursions	Decrease the force to deliver
	Automatic force saturation of the slave arm	Preserve the slave arm and the environment (increase safety)		
	Master force capacity adapted to the operator	Increased security		
TAO software intrinsic benefits	**Accurate balancing of the weight of the master and slave in all positions**	Higher force fidelity=>Better dexterity		
	Partial or total balancing of the weight of the tool or manipulated object			
	Operator visual coordinates	Coherence between vision and action	Absence of mental compensation (no inversion phenomenon)	
	Generalized offset in Cartesian mode in positions/orientations (suppression of telescopic offset "Z electric")	Single push button (instead of three for a conventional telemanipulator)	Handle and tong trajectories occur in the same coordinate (operator visual coordinates)	
	Virtual mechanisms	Improved gesture accuracy/Implementation of complex procedures	Operator only controls useful efforts on the tool	Suppression of guiding efforts on the tool
	Robotic modes	Suppression of repetitive tasks		

Tab 1 : Relation between teleoperation functions/performances and practical benefits for the operator/end user

5.2 Hot-cell testing phase

Due to successful testing in the cold cell testing, an intervention mission has been programmed in the ACR (in English, resin conditioning facility). It will predominantly consist of preventive maintenance operations (cleaning, checking and dismantling/reassembling in the event that equipment needs replacing). During the intervention, the telerobot will be used at its maximum capacity. Moreover, the task to be performed inside the cell is not aligned with the shielding window. Altogether these constraints represent excellent reference conditions to evaluate the benefits of the system. The ACR process is used to condition the resin in a cement matrix is described in the scheme below:

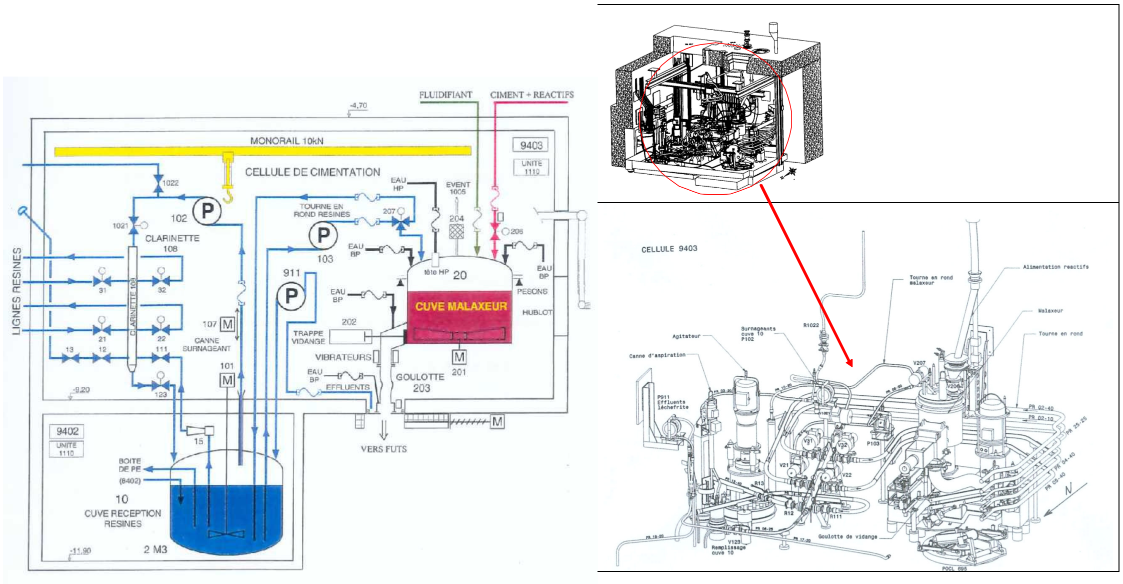

Fig. 3 –The ACR process and the cementing cell 9403 NPH/ACR

The Fig. 4 shows the particular arrangement of the telemanipulator layout.

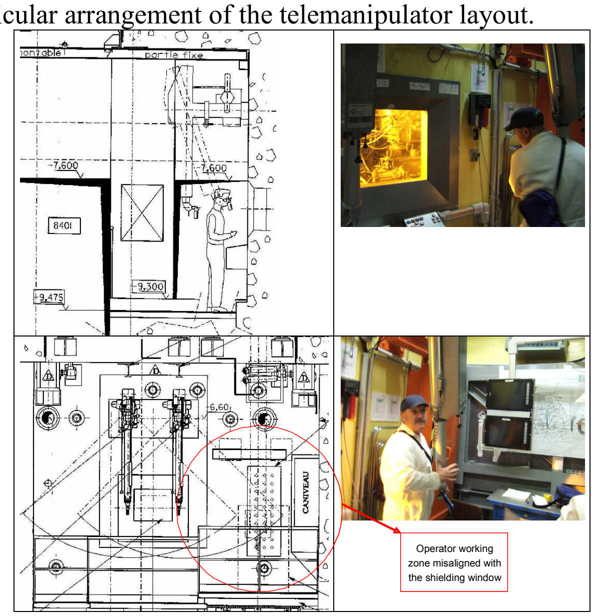

Fig. 4 – Specific telemanipulators layout in the cementing cell

<u>Operations to be carried out in the predominantly preventive maintenance:</u>

- Checking of the equipment, (frequency: 3, 6, 12 months),
- Cleaning of the equipment (frequency: 6 months, 12 months),
- Changing of faulty equipment, which leads to teleoperations that predominantly involve Dismantling/Reassembling mechanical equipment:
  - valves, plugs, connectors,
  - large pieces of equipment such as the mixer, stirrer, hatch, chute,
  - Mechanical process cleaning such as tapping

6. Conclusion

The MT 200 TAO (CAT) is an ergonomically adapted tool as witnessed by the ready acceptance by trained operators, regardless of the change in work habits that this tool will require. It is well suited to the work to be carried out and enables the telemanipulation activity to be significantly improved in terms of efficiency and reliability. The system will be brought into use on the AREVA NC site over the next few months depending on the gains in performances obtained during hot-cell operations. The MT2OO TAO opens a new era in teleoperation; a technical revolution such as those during the 1990's for control stations (migration of the wall mimic diagrams to computerized control rooms).

The partners are now considering the development of an optimized telescopic arm exhibiting an extended lifetime and a higher reliability. The increased use of interactive simulation tools is also forecasted (hot-cells virtual mock-ups, workstation simulators…) to improve Man Machine Interface ergonomics, training methods and task planning.

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Plant Life Extension

MAINTENANCE ISSUES IN RELATION TO PLANT LIFE MANAGEMENT MODELS

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ABSTRACT

Due to current social and economical framework, in last years many nuclear power plant owners started a program for the Long Term Operation (LTO)/PLIM (Plant Life Management) of their older nuclear facilities. A PLIM framework requires both a detailed review of the features of the main safety programs (Maintenance, ISI, Surveillance) and a complete integration of these programs into the general management system of the plant. New external factors, such as: large use of subcontractors, need for efficient management of spare parts, request for heavy plant refurbishment programs demand for updated techniques in the overall management of the plant. Therefore also new organisational models have to be developed to appropriately support the PLIM framework. Last year a network of European Research Organisations carried out many R&D tasks aiming at capturing the aspects of the maintenance programs where research is mostly needed and at developing suitable optimised maintenance models. Using the outcome of these initiatives, this paper aims at identifying the technical attributes of the maintenance program more directly affecting the decision for a long-term safe operation of a nuclear facility, and the issues related to its optimal implementation.

1. Introduction

Due to current social and economical framework, in last years many nuclear power plant owners started a program for the Long Term Operation (LTO)/PLIM (Plant Life Management) of their older nuclear facilities [1,2]. This process has many nuclear safety implications, other than strategic and political ones. The need for tailoring the available safety assessment tools to such applications has become urgent in recent years and triggered many research actions. In particular, a PLIM framework requires both a detailed review of the features of the main safety programs (Maintenance, ISI, Surveillance) and a complete integration of these programs into the general management system of the plant.

New external factors, such as: large use of subcontractors, need for efficient management of spare parts, request for heavy plant refurbishment programs demand for updated techniques in the overall management of the plant. Therefore also new organisational models have to be developed to appropriately support the PLIM framework, integrating both safety related and non safety related issues.

In 2003, the JRC-IE (Joint Research Center, Institute for Energy) launched a network of European Organisations operating Nuclear Power Plants, SENUF (Safety of European Nuclear Facilities). The SENUF Working Group on "Safety of Nuclear Facilities in Eastern Europe dedicated to Nuclear Power Plant Maintenance", hereinafter referred to as SENUF-WG-NPPM, was founded with the following objectives:

1) Review and identification of the remaining open (generic/specific) maintenance related issues,
2) Promotion of well designed and prepared maintenance plans for systems, structures and components,
3) Support for the implementation of advanced maintenance approaches, including implementation of preventive (condition based) maintenance as well as preventive mitigation measures,
4) Evaluation of the advanced risk based maintenance approach and provision of assistance in its implementation.

A background report was developed by the network in 2004 on Maintenance optimisation issues in the EU, supported by a detailed questionnaire in the EU countries [3]. The report collected and evaluated the available and applied maintenance methods at NPPs of acceding and candidate countries to the European Union (ACCs) as well as of the wider Europe (covering Russian Federation and Ukraine), and based on this evaluation, preliminary identified areas for further collaboration with them.
A very successful workshop was organised in Madrid on June 19-21, 2006 on “Maintenance rules: improving maintenance effectiveness”, by the JRC-IE (SENUF network), UNESA, EPRI, Iberdrola, Soluziona and Tecnatom [4]. The workshop confirmed that improving the maintenance program is one of the best tools to improving the overall plant performance and the cost control, even improving the overall plant safety.
A second Workshop was organised by the JRC-IE (SENUF network) and by the International Atomic Energy Agency (IAEA), in Petten on October 2-5, 2006, on “Advanced Methods for Safety Assessment and Optimization of NPP Maintenance” [5]. The workshop addressed the application of advanced probabilistic methods to the optimisation of the maintenance programmes at the European NPPs.
On the basis of the outcome from the SENUF activities in the last years, the objectives of this paper are the following:

- To analyze and summarize the existing strategies on nuclear power plant (NPP) maintenance optimization, i.e. predictive maintenance based on monitoring component condition, reliability centred maintenance, and risk-informed maintenance in the NPPs of the collaborating parties
- To identify the technical attributes of the maintenance programs more directly affecting the decision for a long-term safe operation of a nuclear facility, their implementation issues and safety review.
- To identify differences and commonalities in the Western and Eastern European practice, and based on this evaluation, to identify areas for further research and development (R&D).

2. The maintenance program in the Long Term Operation perspective

There is a generic convincement in the nuclear community, also confirmed by the SENUF questionnaire carried out in 2004 [3], that the maintenance program should have specific attributes in order to support a long term operation (LTO/PLEX) program for the plant. In this sense, the International Standards (e.g. the IAEA) can be seen, but also the national experience of USA, Spain, Hungary, etc. More specifically, the maintenance programs based on standard preventive maintenance (time based), not oriented to the monitoring of its effectiveness, are not considered suitable to support the LTO/PLEX programs. Crucial attributes for maintenance programs in order to support LTO/PLEX are considered: the verification of the performance goals, the root cause analysis of failures, the feedback from maintenance to the ISI program, and the feedback on the OLC (operational limits and conditions).

All Countries implementing an LTO program applied extensive modifications to their requirements on maintenance at first step, setting up mechanisms to monitor the effectiveness of the maintenance activities. In particular, the following features are believed to be indispensable for a maintenance program in a PLIM framework:

1) Monitor the performance of the SSCs (structures, systems and components) which may have impact on safety during all operational statuses of the plants;
2) Assess and manage the risk that may result from the proposed maintenance activities in terms of planning, prioritisation, and scheduling.

In order to implement these requirements, some issues have to be addressed [6,7], namely:

1) The identification of the **scope** of the condition based maintenance rules: typically the Countries choose the safety related SSCs, SSCs which mitigates accidents or transients, SSCs interacting with safety related SSCs, and SSCs that could cause scram or actuation of safety related systems. Therefore, many non-safety related SSCs may see the application of such maintenance rules, with augmented efforts in monitoring their performance and planning their reparation.

2) The setting of the performance **goals** for every component in the scope of the maintenance rules, ranking them according to their risk significance for the plant safety. This task may end up very challenging as, when industry experience is not available, either dedicated PSA tasks have to be developed (with special requirements on PSA quality) or special qualification programs for the evaluation of the component reliability.
3) The performance **monitoring** techniques for the very broad categories of structures systems and components in the scope of the rules.
4) The assessment of the safety **during implementation** of maintenance actions.
5) The **feedback** from the result of the monitoring of the component reliability back into the inspection, surveillance and maintenance procedures. Root cause analysis, equipment performance trend analysis and corrective actions have to be developed on a case by case basis.

In this sense for example the experience of the USA and Spain (where a LTO/PLEX program is well established), Hungary, and Finland (where a PLIM model is in place at the Loviisa NPP) are a confirmation of this generic statement: all these countries modified their regulatory requirements or practice on maintenance, in the direction mentioned above, as one of the preconditions for the LTO/PLEX of their plants.
The SENUF WG carried out a detailed analysis on the experience of some of the above mentioned countries on the interfaces between LTO/PLEX and maintenance programs, as a background for the development of a state-of-art approach to modern maintenance programs [4,5]. The most relevant conclusions are summarised in the following chapters.

3. The RCM programs in the experience of the European Countries

The objectives of the Reliability Centred Maintenance (RCM) and Maintenance Rule [8,9,10] programs as they are usually defined, are listed as in the following (with some differences according to the country framework):

1) Need to control the maintenance cost, particularly in liberalized energy markets, through reduction of unnecessary tasks and optimized maintenance periodicity
2) Improvement of plant safety through better scheduling of maintenance activities
3) Optimization of the management organization, more suitable to control plant safety
4) Development of pre-conditions for the plant life extension
5) Support the production through minimization of outages duration and optimized work control
6) Minimization of the radiation doses
7) Optimized integration among existing safety programs, such as: ISI, AMP, configuration management, design basis reconstruction, etc.

In relation to the operating cost reduction as a consequence of RCM application, the SENUF WG recorded the following reductions [5]:

- In SWE, 10 - 20% of the effort, especially for I&C calibration intervals
- In SP, 20% in work, 30% in number of tasks
- In HUN, expected, not quantified
- In CZ, 30% on a restricted number of systems selected for a benchmark (according to the implemented Phare project in Dukovany NPP)
- In SKR, expected, not quantified.

In relation to the Scoping process applied in the RCM, the WG noted that the approaches are quite different in the Countries:

- In SWE RCM is applied only to non-safety related SSCs. Safety SSCs are analyzed only to get a documented base for the preventive maintenance (PM) program. Analyses of safety system seldom result in any changes of the existing PM-program. The process to get a change of the Technical Specification requirement are very strict and in most cases not worth the effort.

- In HUN RCM is applied to 70% of the safety related SSCs and to 30% of other systems
- In SKR RCM is applied to 44 systems (100-500 components) selected on the basis of different criteria, including safety significance.

The quality of the maintenance documentation was recognized as crucial to feed a proper feedback mechanism. The culture of communication (including the “no blame”) may play a major role in ensuring all failure mechanisms have been properly identified and all actual equipment failures have been recorded.

It was noted that in the current dynamic industry an optimized maintenance system should be adaptive. In particular mechanisms should be put in place to deal with configuration changes, changes of suppliers, emerging results from the aging management programmes (AMP), etc. The need for implementation of a living RCM program under the responsibility of the system engineer was highlighted.

More in detail, the following difficulties and challenges were identified during the implementation of optimised maintenance systems in different EU Countries:

1) The implementation of the MR poses major challenges to the organization: in some cases the interfaces among existing departments were so many that new structures had to be developed. In other cases (Spain) the organization did not change at all and only the coordination was improved. Also in the US, the objective of the action was the re-definition of the interfaces. It was pointed out how the interfaces are very sensitive to the changes in plant configuration and should be promptly updated in such cases.
2) The development of suitable performance criteria is a crucial task. In Spain three years of historical data fed the statistical analysis, complemented by the PSA. In the USA the process was also reviewed by the regulator. The digital I&C cannot be monitored easily in time. Therefore the failure rate usually is provided by the supplier who can derive it on the basis of the whole population of the installed equipment.
3) There is no shared data base on maintenance among European NPPs. Only INPO and WANO provide a worldwide service to their members, though limited to some issues. There are confidentiality issues attached to it, national factors and plant dependent issues that still prevent such communication. Neither non-nuclear plants are involved in this exchange of experience. Some maintenance forum (such as EPRI/NMAC) provides a certain level of experience exchange, however again restricted to members.
4) The interfaces between ISI databases and MR databases are still poor, due to their history: ISI data bases are mainly related to passive components, MR to the active ones.
5) There are objective difficulties in the implementation of the RCM due to the required change in mentality of the personnel and amount of extra work in some cases (particularly when the RCM is not fully computer assisted)

In general, the Ukrainian, Slovenian, Czech, Russian representatives expressed their interest to adopt a MR-like approach in their Countries, even starting on a voluntary bases, most probably closer to the ”equipment reliability” model (INPO/AP-913, [11]). Many of them already created some training centers which are developing procedures in this direction.
The “equipment reliability” program is not mandatory in most of the Countries (including the US). However, it is gaining growing interest for its systematic approach to the management of the plant safety. In particular, the correlation among the many existing safety related programs and the consistent classification of items (important, critical, run-to-failure) seems to be very attractive and practical.

4. Tools for measuring maintenance performance

Recent statistics carried out in the USA (INPO) [4] show that 40% of the failures are related to human factors: among them, 30% are related to engineering deficiencies an 30% to work performance. Most of the significant events in the latter category have been triggered by the supplemental workers. Therefore the contractor performance becomes a crucial issue where many utilities are investing large effort for their reduction. Also supplier reliability is an issue: in many cases equipment were delivered with wrong or different specifications.

Performance Indicators for maintenance effectiveness are considered very useful. However it was recognized that some research work is still needed in this field. It was felt important for the International organization to provide assistance in this field and set up some benchmarking studies.
Maintenance performance indicators are typically based upon: ownership, time from exceedance of the performance criteria and setting of new goals, use of MR to drive performance, etc. Many Countries use the availability and reliability concepts defined in the MR also to monitor the performance of the ageing management programs (AMP).
The WG developed a special set of indicators [14] under testing at many European NPPs.

A special group of indicators are now made available [4] on the “supplemental workers” and the “supplier reliability” in general, by INPO. They are recognized as very useful to monitor one of the main causes of deficiencies in the maintenance systems (they are included for example in the AP-930)

The techniques for the risk monitor during maintenance are also crucial, mainly in relation to the NUMARC 93-01 [12,13] proposal. The use of panel of experts and/or PSA for the construction of the risk matrix or of the risk monitor (real time) are apparently the only two available techniques.

Some data bases are available on component reliability in Europe: for example the experience of DACNE for PSA failure probabilities and for MR performance criteria (by Tecnatom), the EPIX (by INPO) and the PKMJ (by EPRI). However, most of them remain country specific and/or restricted to the contributing users.

The WG recognized that no tools are available yet to manage the maintenance process in a comprehensive manner, even if the EPRI proposals are excellent in some fields. The user groups (EPRI/NMAC, EPRI/MRUG, etc.) are providing an invaluable contribution to this concern to their subscribers.

5. Use of PSA for maintenance optimization

In case the maintenance optimization is supported by the application of PSA results and models [5], the quality of the PSA becomes an important issue for the success of the maintenance optimization. As any PSA application, the maintenance optimization has crucial requirements for the PSA quality. The scope, completeness, modelling details and used data should be such that allow the PSA to be used for adequate support of maintenance optimization. In order to ensure an appropriate PSA quality, as minimum the following actions should be implemented:

- Use appropriate guidelines during development of PSA and review of PSA
- Involve both PSA experts and NPP maintenance staff in the development of PSA models
- Keep in mind the intended applications at the time of scope definition and if possible take into account the available standards.
- Perform PSA regulatory review before maintenance optimization is implemented.

Basically two guidance for qualification of PSAs for specific applications are available, namely: the ASME RA-S-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications and the IAEA TECDOC 1511 [15]. These documents facilitate determining how suitable a given PSA is for a specific application and in particular for supporting maintenance optimizations.

In particular, maintenance related special PSA needs may include the following:

- Separation of the maintenance related basic events in the component unavailability models, like unavailability due to repair, planned maintenance, test, human errors etc.
- Modeling of maintenance activities in each of the safety system trains to correctly reflect actual maintenance activities
- Use of more detailed reliability models for modeling of PSA basic events, e.g. to identify failure modes of components affected by different type of maintenance
- Additional special models to support ISI, On-line maintenance, RI configuration control, etc...

In addition, it was noted that risk monitors are useful tools to support maintenance planning off-line and on-line restoration strategies in case of equipment failures during the plant operation.

6. Conclusions

The workshop identified some areas where some R&D effort is needed to support the full implementation of RCM models in European Countries. These areas cover research tasks and call for an initiative at the International Organizations level.

In the field of regulatory practice, support would be needed in the licensing of advanced maintenance optimization applications and information on the regulation in the countries with good practices in the field. In particular, the following recommendations for future support from international organizations were identified:

- Develop detailed guidelines for regulatory review of specific maintenance optimization applications such as: RI TS, RI ISI, On-line maintenance, etc.
- Provide training and/or training material, tutorials for regulatory review of maintenance optimization applications.
- Promote benchmark exercises.

In relation to the PSA quality issues, need for support was identified in the following tasks:

- Disseminate the available PSA quality guidelines (for example the IAEA TECDOC-1511) and promote their development towards Level 2 PSA and at least internal floods and fires in order to facilitate the regulatory use of the PSAs
- Provide support for establishment of WWER specific component reliability database

In terms of research tasks able to make the RCM more broadly applied, the following was identified:

- Clarification of the reliability target for the different groups of components and reliability parameters calculation
- Integrated management of the data bases available at the plants: many sources of data are available at the plants (ISI, maintenance, AMP, PSA, operation, etc.) but often they are not integrated and they do not support an integrated approach to component reliability.
- Development of criteria for “good” performance of SSCs (acceptance criteria)
- Identification of representative maintenance effectiveness indicators
- Understanding of the impact of the RCM on the workforce: in relation to different competencies needed and overall reduction of the workforce at the sites
- Comparison of the available methodologies for RCM: the available proposals are very much affected by the national frameworks where they have been developed. Benchmarking on selected systems and commodity groups would be very useful to this concern
- Exchange of information at the EU level, despite of the national differences and plant issues, would be very useful in the following areas:
    - Methodologies for RCM
    - Organizational aspects
- Derive failure rates for commodity groups (with some assumptions on anchoring, environment, etc.)
- Develop guidelines for training of personnel and use of training centres in the field of optimised maintenance programs oriented to PLIM.

The WG concluded that there is a potential, very important role for the IE network on safe operation of nuclear installation (in the research field) in the coordination of the efforts among the European Countries to promote a full implementation of maintenance optimization programs.

In fact the implementation of RCM methods requires the availability of component data, well established probabilistic techniques of appropriate quality etc. that cannot be developed at the Country level only. In this framework, any future action in the EU/FP7 [1] would be most probably very welcome and will provide concrete support to the enhancement of the safety of the European Plants.

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PLANT LIFE EXTENSION (PLEX) & REPOWERING STUDY FOR EMBALSE NUCLEAR POWER PLANT

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ABSTRACT

Embalse (Cordoba) PLEX pre-project work has started and is being executed by AECL and Ansaldo Nucleare, for NPP and BOP systems respectively.
The job is organized in order to get within two Plant Outages the residual life evaluation of the as built configuration of the Plant; within the outage 2010 (end of the Plant Design Life) it is planned to produce the intervention planning to be implemented in the years 2009 and 2010 to get the required life extension.
Through this strategic approach it would be possible to get the life extension without being forced to stop the Plant for updating implementation.
This PLEX for Embalse BOP first phase consists, as minimum, of Condition Assessment and Residual Life Evaluation of the BOP, and – in parallel – investigation of potential Power Uprate of the BOP, based on evaluation of the possibility to increase the power associated to thermal cycle design optimization.

1. Introduction

Nucleoelectrica Argentina S.A. (NASA), the Argentine Utility owner and operator of the Embalse CANDU Nuclear Power Plant, has the firm intention of refurbishing and extending the design service life of the Embalse Nuclear Power Plant (Cordoba) up to 2035, providing a Plant Life Extension equal to 25 years. Embalse Plant Life Extension (PLEX) pre-project work has started in 2006 and is being executed by AECL and Ansaldo Nucleare (ANN), for reactor building and non nuclear systems respectively.
ANN is performing the first study activities on the Balance of Plant (BOP) equipment and systems, as part of an overall Plant Life Management program. Further, in the frame of PLEX program, NASA has assigned a contract to ANN to perform a Power Uprate study on Thermal Cycle, that will include thermal cycle efficiency improvement evaluation.

2. Embalse Plant Life Extension Project - Overview

First Phase of the Refurbishment and Life Extension project for the Embalse Nuclear Power Station consists of all preparatory activities that are required to define the refurbishment scope and costs, for input into the utility business case for the Refurbishment and Life Extension project.
One of these activities is the Pre-Project Plant Condition Assessment Project as part of an overall Plant Life Management program, that provides for the systematic assessment, timely detection, mitigation, recording, and reporting of significant aging effects in Systems, Structures and Components.

Main elements of the project are:

i) **Aging Assessment**

Systematic aging assessments of Systems, Structures, Components (SSCs), or groups of components with similar characteristics (Commodities), selected according to a priority process. Aging assessment generally entails a review of data in order to assess the effect of aging degradation on SCCs; it establishes their current condition and provides a prognosis for attainment of design life and/or long term operation with associated recommendations.

They include:

- **Condition Assessment (CA):** Typically applied to SSCs or Commodities. The methodology entails a general review of design, manufacturing, installation, operations and maintenance at a component level.
- **Life Assessment and Residual Life Assessment (LA & RLA)**: Typically applied to critical and complex components and structures, that are designed not to be replaced as part of normal maintenance program, and that are subject to long term degradation mechanisms. The methodology entails a detailed review of design, manufacturing, installation, operations and maintenance at a sub-component level.

ii) **Implementation**

Conclusions and recommendations emerging from aging assessment studies provide input into the implementation stage carried out by the Plant.

Aging assessments will be executed to determine which of the selected SSCs are recommended for inspection, replacement or repair during the Refurbishment Outage and which may be done during normal maintenance outages. They will also provide a health prognosis for continued operation of the SCCs for life attainment and life extension beyond the refurbishment outage, and may identify changes which are necessary and sufficient in order to deal with issues related to equipment obsolescence and aging effects.

3. Embalse PLEX – BOP Assessment

ANN approach to Embalse PLEX program is divided in two main phases:

- **PHASE 1**: engineering activities which will govern and address the walkdown inspections in the BOP to assure Plant Life evaluation and extension up to the Customer requirement; it is part of this stage also the Plant Power Uprate evaluation working on the Thermal Cycle parameters and re-evaluation of existing components in the frame of Plant Life Extension;

- **PHASE 2**: Detailed / Constructive Design, Hardware Procurement and Installation together with in field potential supervision and assistance considering the results of the different walkdowns addressed between the years 2007 and 2008 and to get the goal of 25 years of PLEX as well as of potential identified Nuclear Power Plant Repowering.

According to the above approach, a Technical Study is now on going for the Phase 1 Engineering Activities, aimed to get the following objectives:

i) Evaluation of the as-built configuration of the Thermal Cycle and Essential Systems in Embalse BOP, taking advantage of the Plant Cognitive Walkdowns;

ii) Through the above Cognitive Walkdown visit results, identification of all the systems which will need inspection and checks in order to get a residual life evaluation based on the as built configuration is performed; for all these systems and for the identified and selected inspection / checks activities, an action planning to be addressed among the 2007 and 2008 planned Outages (respecting and not affecting the already planned duration of each outage) is going to be issued in order to assure the extension of the life of the Thermal Cycle as well as of the BOP essential systems for 25 (twenty five) years, as required by the Customer;

iii) At the completion of the In service inspection / Non Destructive Examinations / Tests actions among the planned outages, it will be issued a Global Integration Report, organized into two sequential sections:

  - Residual Life Evaluation (RLE) of the BOP in as built configuration;

- Intervention Planning (integrated with technical procurement specifications, bill of materials, costs-benefits analysis and budgetary economical estimate of the updating / repowering engineering, supply and installations activities) addressed to get the plant life extension for the BOP systems up to 25 years.

iv) Evaluation of a potential Repowering of the Plant considering as variables potential modifications of the as-built Thermal Cycle configuration and replacement of aged components of the cycle itself; implementation feasibility in hardware in the Plant BOP will be part also of the "Cognitive Walkdown" visit .

At the completion of the steps described here above and following the results of the walkdown in the site integrated in the Global Integration Report, the Phase 2 Study activities will be produced for:

- Engineering, Hardware procurement and assistance in equipment replacement in the field, so as to get the Plant Life Extension goal of 25 years;
- Engineering, Hardware procurement and assistance in equipment replacement for Thermal Cycle in the field to get the Plant Repowering;
- Planning of the Engineering and Procurement Activities in order to produce specific activities and implement some modification during the outages prior to the end of the design Plant Life.

3.1. Plant Life Extension Study - Phase 1

Independently of their implementation approach in different sequential walkdowns in the Plant, the following activities are on going, as part of the whole **Phase 1** job:

- Investigation of the as built configuration of the Embalse BOP
- Investigation of potential Power Uprate of the Embalse BOP

3.1.1. Investigation of the as built configuration of the Embalse BOP

Based on the available Embalse documentation in terms of Process and Instruments Diagrams / Systems Technical Descriptions for BOP BSI as well as of BOP General Arrangement / Composite Drawings, the following documentation / information are in progress to be processed:

a) Systems identification to be investigated in the current aged configuration;
b) Mechanical / Electrical Components identification to be in field investigated to check / monitor their aged life;
c) Piping systems lists and associated design classification to be in field investigated to check / monitor their aged life;
d) Supports lists (and associated piping systems) to be checked during the in field visit;
e) BOP Control system as built configuration analysis.

For each item to be processed through the walkdown visit in order to get information on the current aged life, dedicated **check lists** have been prepared, covering the following information sections:

- **General information** (i.e. Consistency between design and as built configuration, integrating potential modification implemented during the spent Plant Life, Identification of experimented monitored thermal/pressure transients of the NPP which might have affected the mechanical / electrical components / piping systems life, Identification of external accidental events which might have affected SSCs design life, Identification of internal accidental events which might have affected BOP residual life, Components identification and current availability of the same typologies in the market, Investigation on availability of in field documentation like "Embalse Plant Operational Transients experimented during the whole Plant Life", "Periodical Maintenance Operations registration for the most critical electrical and mechanical components", and "In Service Inspection registration" notes, Direct contact with Plant Operators and Maintenance / In service inspection personnel to catch direct experience of problems arised during the Plant design life, Analyses of existing operational log books for determining the utilization degree of the various components, along with the history of their maintenance, preservation, and use of spares, Identification of the industry codes and standards applied for the original Embalse Project, and proposal of an equivalent set of codes and standards that could be used in case of components refurbishment/replacement. The basic idea is to demonstrate that the application of the least stringent codes and standards is still acceptable to the National Safety Authority, thus simplifying the market investigation on the same equipment types)

- **Detailed information through investigation**
  As built configuration for SSCs belonging to well identified characterization (selected Electrical Systems and Mechanical Components in 2007 outage and for example piping systems in 2008 Outage); Ageing as built configuration for SSCs belonging to a well selected identified BOP Plant Area; in this way all the disciplines might be covered simultaneously in the Residual Life evaluation of the essential systems, as a function of SSCs available in that selected Plant Area.
  Non Destructive Examinations (NDE) for mechanical components / piping systems in order to check residual structural thickness or evaluate weldings configurations in critical joints.
  Specialistic checks (insulation tests, fire resistant tests, short circuit test) will be performed on electrical components in order to evaluated the aged configuration of each electrical component;
  Specialistic checks on currents as-built status and aged life of instrumentation for systems and components will be performed in order to judge properly on their potential systematic replacement;
  Specialistic check of current available BOP Control system configuration in order to properly evaluate the opportunity of **integration of a Distributed Control System (DCS)** as implemented for Cernavoda Unit 2 BOP.

Based on the Integrated Multi-disciplinary Walkdowns Visit Report, specialistic analytical evaluations to check the residual life of the Plant in BOP Portion for specific items will be performed.

For **piping systems** analytical evaluations (piping stress analyses) will be made in order to check structural behavior simulating the residual thickness as per NDE investigation, to simulate potential local permanent plastic deformation or non correct structural support configuration (springs working as rigid support or out of the spring allowed range) range; through these analyses, the following aspects will be properly evaluated:

- RLE for each simulated portion of piping system in the as built configuration and identification of the “critical” items which dictate the residual life for the investigated piping portion;
- Applicability of Leak Before Break (LBB) theory for high energy Lines ;
- Items / Spools / Supports to be replaced with new ones in order to assure a life extension consistent with the Customer requirements, assuming in terms of design loads, a similar loads / design thermal / pressure transients distribution as per the spent life;
- Replacement Planning for each Piping System will be prepared and submitted to a cost / benefits analysis in order to assure the life extension required by the customer;
- For each item to be replaced it will be supplied a delivery time schedule and a reference procurement specification integrated with the design data sheets.

For **mechanical components** analytical evaluations (components stress analyses) will be made in order to check structural behavior simulating the residual thickness as per NDE investigation, to simulate potential local permanent plastic deformation or incorrect structural support configuration (springs / shock absorbers working as rigid support or out of the spring allowed range) range; the job will have two standard approaches addressed one for static components (tanks, heat exchangers, pressure components, chillers) and the other for dynamic components (pumps, fans, compressors, valves); through these analyses, it will be properly evaluated the following:

- RLE for each simulated component in the as built configuration and identification of the "critical" items which dictate the residual life;
- Items / Spools / Supports to be replaced with new ones in order to assure a life extension consistent with the Customer requirements, assuming in terms of design loads, a similar loads / design thermal / pressure transients distribution as per the spent life ;
- Replacement Planning for each Components or portion of itself will be prepared and submitted to a cost/ benefits analysis in order to assure the life extension as required by the Customer;
- For each item (or portion of the same) to be replaced it will be supplied a delivery time schedule and a reference procurement specification integrated with the design data sheets.

For **electrical components** analytical evaluations (components stress analyses) will be made, in order to check structural behavior simulating the residual thickness as per NDE investigation or other damage got through visual inspection; the job will have an approach similar to the two standard approaches as defined for mechanical components; additionally the job will be addressed to check primarily the component electrical functionality for a potential life extension as required by the Customer; through these analyses, the following aspects will be properly evaluated:

- RLE for each simulated component in the as built configuration and identification of the "critical" items which dictate the residual life, mainly in terms of electrical functionality;
- Items / Supports / Components (Transformers, switchgears, circuit breakers Etc.) to be replaced with new ones in order to assure a life extension consistent with the Customer requirements , assuming in terms of design loads, a similar loads / design current / voltage transients distribution as per the spent life ;
- Replacement Planning for each Components or portion of itself will be prepared and submitted to a cost/ benefits analysis in order to assure the life extension as required by the Customer;
- For each item (or portion of the same) to be replaced it will be supplied a delivery time schedule and a reference procurement specification integrated with the design data sheets.

For the **BOP Control System**, following the results of the different walkdowns investigation, it will be properly evaluated the possibility to implement a new distributed control system , replacing the current existing one cabled control system; the job will be performed according to the latest ANN experience gained in the construction of the Nuclear Power Plant Cernavoda Unit 2 providing for the same BOP a Distributed Control System (DCS).

At the conclusion of the Phase 1, but its preparation can be addressed in parallel, will be issued the "**Safety Report**" for the PLEX activity, covering all the investigations studies addressed to finalize system by system the residual life evaluation as well as the intervention planning to get its required life extension.

3.1.2 Investigation of potential Repowering of the Embalse BOP

The possibility to increase the Power of the Embalse Nuclear Power Plant will be evaluated, as an integration of the Plant Life Extension activity. It is ANN intention to evaluate the possibility of a potential Embalse NPP Repowering into two different stages belonging nevertheless both to the **Phase1** job:

° Stage 1 – Repowering associated to thermal cycle design optimization, without considering any modification to the current configuration of the turbine blades;

° Stage 2 – Repowering associated to updating of the turbine blades lengths or modifications to be implemented on the Turbo-generator system.

In the frame of Stage 1, the following variables of the Thermal Cycle will be investigated:

1 Moisture separator efficiency in the range 95 – 97 %;
2 Pre-Heaters number in the thermal cycle;
3 Decreased friction losses value in the main steam interconnecting lines turbine / pre-heater;
4 Decreased head losses in high and low pressure inlet valves;
5 Condenser vacuum evaluation;
6 Double ReHeaters (RH) integration;
7 Drainage calculation for turbine;
8 Discharge Pressure at the high Pressure Stage of the Turbine;
9 Efficiency of Electrical Generator

Results of potential power increasing related to each one of the selected listed variables will be shown and justified.

3.2 PROJECT STATUS

ANN has dedicated a project team composed by engineering specialists and on site technicians in order to perform cognitive walkdowns and data gathering during normal operation of the plant and in Outage period, inspections and survey walkdowns during Outages. In particular, for data gathering activity, some resident specialists have been involved for some months.

3.2.1 Cognitive Walkdowns during plant operation: STATUS

Two cognitive walkdowns have been performed on site in December 2006 and February 2007, by two specialists teams and some preliminary visual inspections have been done. A third one is now on going. All the systems have been identified and a **first priority system list** has been defined, as per Table 3.2.1.

BSI	System	Priority
36100	Main Steam Systems (inside TB)	**2**
41120	Steam system including Moisture Separator	**2**
41130	Reheater Drains and Vents	**2**
42100	Main Condenser	**1**
42120	Extraction Air	**2**
43100	FW Heating & Extraction Steam (preheaters)	**1**
43210	Condensate Systems	**1**
43230	Feedwater Systems (inside TB)	**1**
43350	Steam to Feed-water Heaters	**2**
45100	Condenser Leak Detection & Sampling Systems	**2**
45400	Chemical Addition Systems	**2**
45510	Turbine Building Inactive Drainage	**4**
51100	Transmission Line 500kv + MOT	**3**
51300	Transmission Line 132kv + SST	**3**
51400	Phase Bus 22kv + main breaker + UST	**3**
52000	Standby Emergency Generators	**3**
53000	MV Distribution (Class IV & III)	**3**
54000	LV Distribution (Class IV & III)	**3**
55000	UPS, Class II & I (EPS excluded, AECL scope)	**3**
66200	BOP Control Centre Instrumentation Racks	**4**
66300	BOP Control Centre Logic Panels	**4**
71100	Pumphouse Common Systems (screen house)	**1**

71210	Circulating Water Supply System	**1**
71310	Service Water Systems (Outside R/B)	**1**
71400	Fire Protection System	**4**
71900	Chilled Water Systems (Outside R/B)	**2**
72130	Auxiliary Steam Condensate System	**2**
72140	Auxiliary Steam Distribution	**2**
73010	Hot Water System	**4**
75100	Compressed Air System	**1**
75120	Instrument Air System (Outside RB)	**1**

Table 3.2.1: first priority system list

During this phase ANN has searched and involved some original manufacturers in order to have technical consultation related to condition assessment of Main Components.

3.2.2 Walkdown during Outage 2007: STATUS

Some Plant Areas and Main Components have been selected for investigation during Outage 2007, for Cognitive and for Surveillance purposes, as described hereafter:

- **Valves** inspectable during Outage
- **Electrical Equipment** inspectable during Outage
- Piping (**Expansion Joints** and **supports** inspectable during Outage)
- **Feed Water Pumps** (FWP) and **Circulating Cooling Water pumps** (CCWP)
- Some Main components (**Condenser**, **MSR**, **exchangers**, filters, ecc.)
- Galvanic protection measurements

Most of the planned activities have been performed; some activities have been planned for Operation and for next Outage period.

3.2.3 Repowering study on BOP: STATUS

The relevant possible scenarios of Repowering have been found and presented to NASA.
Considering the impact of each modification on the Thermal Cycle As-built configuration, four main scenarios have been studied, as described hereafter:

- modification of Steam Supply conditions to the 4th Feed Water preheaters
- addition of a Feed Water High Pressure preheater (5th preheater) with new steam extraction from HP turbine
- addition of a second Steam Reheater
- Moisture Separator / Reheater efficiency improvement

At the same time each scenarios has been evaluated with the modification of Low Pressure Turbine last stage blades.

3.3 Conclusions

It has been developed a global view of ANN approach in RLE and related PLEX study, including the current project status, the performed activities and the repowering study.
First main difficulty has been the managing of such great quantity of information from archives (ANN & Plant), from Plant Staff Interviews, from Inspection Reports, etc. that forced the team to build the software component/pipeline database immediately.
It has been demonstrated that a good coordination between ANN team and NASA Plant team, helped by efficient communication procedures, gives the project a very important effort to reach the main goal of Plant Life Extension, goal that seems to be reachable in a mid-term planning.

CHANGE OF THE COMPONENT IMPORTANCE VALUE ACCORDING TO THE INITIATING EVENT MODELLING AND A PROBABILISTIC SAFETY ASSESSMENT QUANTIFICATION METHOD

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ABSTRACT

This paper presents the elements that affect importance analysis result and it also provides a method to obtain an accurate component importance ranking by considering those same elements in a Probabilistic Safety Assessment (PSA). According to a change of an operational method and regulation method on Nuclear Power Plants (NPPs) by using risk information, it is important to select the risk-significant or safety-significant components to make the right decision for a risk-informed regulation and application. Thus, we reviewed an importance ranking change according to the cutoff value and the Initiating Event (IE) modelling method. The results of this study revealed that a failure inducing an IE and the cutoff value affect importance analysis results. Therefore, to obtain accurate importance estimation results, it is recommended that the cutoff value be lower than 1×10^{-13} and that IE Fault Trees (FT) be used during a PSA quantification process.

1. Introduction

In this paper, we present the change of a component importance ranking according to the methods used to handle the Initiating Event (IE) frequency and the cutoff value during a PSA quantification process.
In the Risk-Informed Applications (RIA) and the Risk-Informed Regulations (RIR), the importance measures for Structures, Systems, and Components (SSCs) provide some of the most useful PSA information. Therefore, an importance analysis is an important tool for providing appropriate PSA information for an NPP. Recently, risk information has been used as a method to make a decision on the selection of the risk-significant or safety-significant in the field of the RIR and RIA such as Risk Informed-In Service Inspection (RI-ISI), Graded Quality Assurance (GQA), and maintenance rule, etc [6][7].
Section 2 briefly details the method used in this study for an identification of the elements affecting the importance estimation results. Section 3 explains the method used in our study. Section 4 presents the results of the comparison of importance value according to the elements selected in this study. The concluding remarks are given in section 5.

2. Identification of the elements affecting the importance estimation results

To obtain more accurate importance analysis results, it is necessary to identify the elements affecting an importance ranking.
We have investigated whether the IE modelling method affects the importance analysis result or not in this study. In general, there are two approaches to estimate the frequencies of IEs. They are a Fault Tree (FT) modelling method and the Bayesian analysis method when using historical data [1][2][3]. However, the IE frequency is usually handled as a value during a quantification process even though it is obtained

through an FT analysis. Thus, we have evaluated the fact that there will be an effect on the importance value of components if we use an IE FT directly instead of a frequency value during a quantification process. It means that we might obtain an incorrect importance estimation result by not considering the initiator effect on the components.
However, if the same component failure event is used in an IE FT and a mitigating system FT simultaneously, it is not easy to evaluate the importance value for a component. Thus, most importance values for components are estimated by considering the role as the mitigating system. Therefore, we should include not only the role of a mitigating system but also an IE to obtain a realistic component importance value.
In addition, we have evaluated the fact that the importance estimation result might be altered according to the cutoff value used during a quantification process.
All PSA analysts have found that the Core Damage Frequency (CDF) value is different according to the cutoff value used during a quantification process. However, most PSA analysts have overlooked the effect of the cutoff value on the importance estimation result. Generally, we have used a value of about $1x10^{-10}$ or so as a cutoff value because of a time constraint and a quantification engine's limitation. Thus, we reviewed an importance value change according to a cutoff value change.

3. The method used in this study

In previous PSAs, most IE frequencies have been obtained from historical data. However, if a plant has its own specific design of a system for triggering an IE, it is not appropriate to use the generic IE frequency value estimated from historical data. In this case, researchers usually estimate the IE frequency value through an FT analysis by considering the specific design of a system. They generally use an FT analysis method to estimate an IE frequency value for the support systems such as the Component Cooling Water System (CCWS), Service Water System (SWS), Instrument Air system (IAS), and the Electrical Power System (EPS), etc. However, once they have obtain the IE frequency value of a support system through an FT analysis, the IE frequency is treated in the same way as any other IE is treated in an Event Tree (ET). To review the change of an importance value according to the IE modelling method, we revised the IE FT for Loss of CCW (LOCCW) to use it during a quantification process. The major importance measures for the components relevant to an LOCCW were estimated and the importance results were compared with the importance estimation results obtained by using a frequency value.
To review the effect of a cutoff value change on the importance result, we compared a change of the number of basic events classified as safety-significant by changing the cutoff value from $1x10^{-8}$ to $1x10^{-15}$.
We performed an importance analysis for the one top model for Level 1 of Ulchin 3&4, Rev. 1 by using the FT quantification S/W FTREX [10, 11]. The one top model of Ulchin 3&4 is composed of 49 IEs, 2,807 gates and 2,498 basic events.

4. Comparison of the importance estimation results

4.1 Importance estimation results according to the IE modelling methods

Several components in the CCW system have different importance results in the case where the IE FT is used when compared with the case where the IE frequency value is used.
Table 1 shows a part of the importance estimation results of the CCW components when the IE frequency value and the IE FT are used, respectively. As presented in Table 1, the results show that the importance value of the components is altered according to the IE modelling method. From the viewpoint of the Risk Achievement Worth (RAW), the importance value for the CCW system components, 3461CC-V0142, V0905, 3461M-PP01A, PP02A, and the Essential Chilled Water (ECW) system components, 3633M-CH02A, 3633WO-V1014A, were changed considerably. However, the importance value for the other components has not changed that significantly.
Through the results of a component importance value for the CCW system and the ECW system, we found that component importance value for the ECW system is altered more when compared to the CCW component importance value. We concluded that this result is because the ECW system is more

dominant against a CDF than the CCW system. That is, it is understood that the importance value change is not considerable since the minimal cutsets induced by the LOCCW are not dominant against a CDF. Therefore, if an IE that is dominant against a CDF was modelled as an FT, the importance value for the components relevant to that IE would be changed considerably.
In the example of this study, the chiller unit, 3633M-CH02A, was categorized as a non-safety significant component for the case of a quantification process by using the LOCCW frequency value. However, the same chiller unit was categorized as a safety-significant component after we performed the quantification process by using the LOCCW frequency FT. If a component's RAW is above 2, it is classified as a safety-significant component [6], [7]. Figure 1 graphically displays the importance vale change for some components when using the two IE modelling methods.

	Using IE Frequency Value			Using IE FT		
Component	FV	RRW	RAW	FV	RRW	RAW
3461CC-V0073	0.000137	1.000137	2.24	0.000135	1.000135	2.26
3461CC-V0074	0.000899	1.0009	2.91	0.000417	1.000417	2.49
3461CC-V0095	0	1	1	0.00027	1.00027	1.19
3461CC-V0105	0.00003	1.00003	1.21	0.00003	1.00003	1.21
3461CC-V0106	0.00005	1.00005	1.22	0.000035	1.000035	1.21
3461CC-V0141	0.000186	1.000186	2.66	0.000182	1.000182	2.68
3461CC-V0142	0.001774	1.001777	4.06	0.000714	1.000715	3.14
3461CC-V0905	0	1	1	0.0048	1.004823	1.83
3461CC-V0906	0	1	1	0.000056	1.000056	1.28
3461CC-V1001	0	1	1	0.000002	1.000002	1
3461CC-V1002	0.000004	1.000004	1	0	1	1
3461M-HX01A	0.00002	1.00002	1.83	0.00002	1.00002	1.84
3461M-HX01B	0.002536	1.002543	106.67	0.002426	1.002432	102.08
3461M-PP01A	0.00033	1.00033	403.77	0.000376	1.000376	465.36
3461M-PP01B	0.000328	1.000328	382.75	0.000353	1.000353	443.59
3461M-PP02A	0.00033	1.00033	403.77	0.000385	1.000386	465.36
3461M-PP02B	0.000352	1.000352	403.79	0.000384	1.000384	465.36
3633M-CH02A	0	1	1	0.051079	1.053829	4.13
3633M-CH02B	0	1	1	0.000389	1.000389	1.44
3633M-PP01A	0	1	1	0.000008	1.000008	1
3633M-PP02A	0	1	1	0.000074	1.000074	1.02
3633WO-V1010A	0	1	1	0.00044	1.00044	1.44
3633WO-V1014A	0	1	1	0.000063	1.000063	1.28

Tab 1: Comparison of the Importance of the components for the CCW and ECW Systems

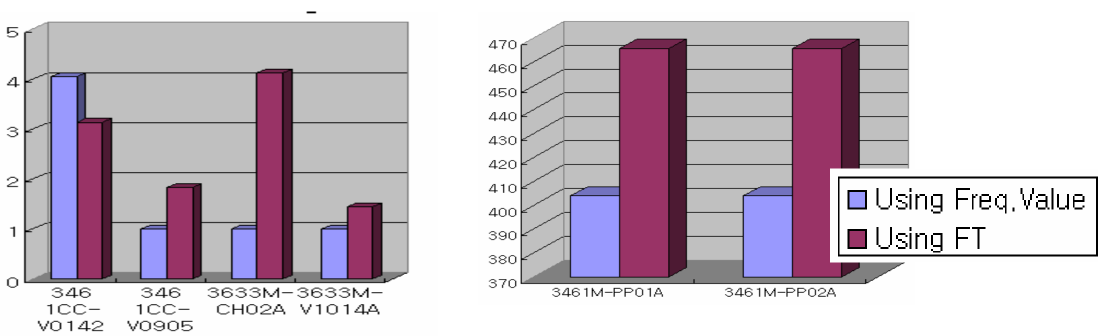

Fig 1: Comparison of the RAW

4.2 Importance estimation results according to a change of the cutoff value

Figure 2 shows that the RAW is considerably underestimated in case where the cutoff value is high.

We can not solve the FT in the case where the cutoff value is lower than 1×10^{-15}. Importance analysis results are summarized in Table 2, when the cutoff value is 1×10^{-15}. Figure 2 displays that the cutoff value should be lower than 1×10^{-13} so as not to overlook the safety-significant SSCs or to prevent a RAW underestimation.

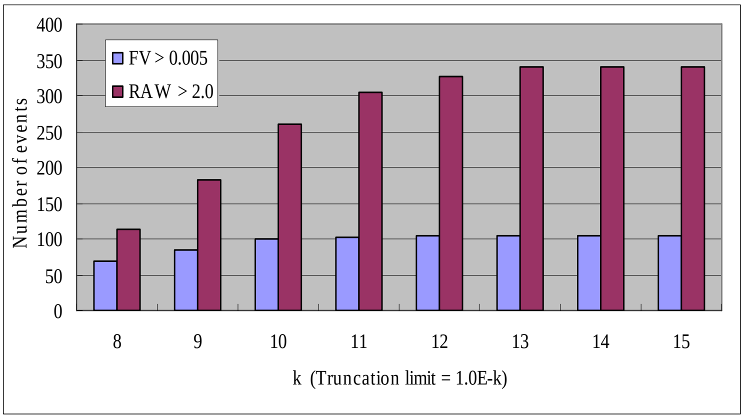

Fig 2: A comparison of the number of safety-significant basic events according to the cutoff value

Elements	Number
Cutoff value	1×10^{-15}
Minimal Cutsets	27,000,850
Basic Event of Minimal Cutsets / Basic Event	1,754 / 2,498
Basic Event (FV>0.005)	105
Basic Event (RAW>2.0)	341
Basic Event (FV>0.005 or RAW>2.0)	373
Basic Event (FV>0.005 and RAW>2.0)	73

Tab 2: The summary of importance estimation result for the cutoff value, 1×10^{-15}

5. Concluding remarks

An importance value change for a component is very important in the RIR and the RIA since these importance measures are used as a method to select a safety-significant or risk-significant component.
Through this study, we identified that a failure inducing an IE affects an importance estimation result for components. Through a review of a component's importance estimation results for the CCW system and the ECW system, we found that the ECW component's importance value was altered more when compared with the CCW component's importance value. From the importance value comparison, we concluded that this result is because the ECW system is more dominant against a CDF than the CCW system is. That is, it is understood that the importance value change is not remarkable since the Minimal Cutsets (MCSs) induced by an LOCCW are not dominant against a CDF. Therefore, if an IE which is dominant against a CDF such as a Loss of Offsite Power (LOOP) was modelled with an FT, a component's importance value would be change considerably. Here, we only reflected the enabler events of the IE FT events to estimate a component's importance value because of a technical limitation. If we could reflect the effect of an initiator, the importance value for the components would be changed considerably. Thus, the development of an FT for other IEs is necessary for an estimation of a more exact importance value for components.
Also, we identified that the RAW was underestimated when the cutoff value is high. Moreover, we found that we can prevent RAW underestimation when the cutoff value is lower than $1\text{x}10^{-13}$. However, the Fussel-Vesely (F-V) value was not affected by the cutoff value.
Therefore, to obtain an exact importance value estimation result, it is recommended that the cutoff value be lower than $1\text{x}10^{-13}$.

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Session 19.1.2

Maintenance and operation

THE EFFECT OF SEA SALT AEROSOLS IN THE JAPAN SEA COAST FACILITIES

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ABSTRACT

All of 15 nuclear power plants in Fukui Prefecture are located in the Japan Sea Coast. Corrosion is strongly influenced by material and environmental factors. We installed the salt damage experimental yard at Awara sea coast in March, 2006. We are doing the open air test, sheltered test without filter and with filter.

The carbon steel standard specimens were exposed under three kinds of test conditions mentioned above. The corrosivity of the open air test specimens was higher than it of Miyako Island, Okinawa. The corrosivity of carbon steel of winter was higher than it of summer.

The measurement of chloride deposition rate was done by the dry gauze method. The correlation between chloride deposition rate and the average velocity of the wind was found. The correlation between chloride deposition rate and the average rainfall was a little complicate. It means the chloride deposition rate has maximum value at a certain rainfall amount. No other clear relation was found.

1. Introduction

Fukui Prefecture presently hosts 15 rector units, all of which are located along the Sea of Japan. Under such a circumstance, corrosion due to sea salt aerosols is one of the major factors causing ageing degradation of nuclear power plants facilities. Many scientists have been engaged in research on the corrosion of structures due to sea salt aerosols[1-8)]. In pursuing sea salt aerosol-induced corrosion research, it is necessary to perform corrosion tests under a certain set of environmental conditions since corrosion of structures is highly sensitive to environmental factors. In this respect, we installed the outdoor exposure test facilities along the seacoast in Awara-city, Fukui Prefecture. To obtain data which can be compared with the past research results, we performed 3 types of exposure tests; placing standard test pieces (carbon steel) in the open air environment and placing the same test pieces in the shielded environment with and without a filter in the air intake.

2. Testing method

The outdoor exposure test facility located on the seacoast. This facility was manufactured and installed according to JIS Z 2381 (2001). The facility has 2 air intake openings (on the front and rear sides) in parallel with the coastal line. One facility has no filter on its air intake another has a filter. The standard test pieces are installed on the facility roof while other test pieces installed inside the test facility. Both the standard test pieces and other test pieces installed inside of the exposure test facility were manufactured according to JIS Z 2383 (1998). These test pieces are sized in 10cm×10cm. After going through the specified test period, we weighed the test pieces after removing corrosion products to determine the corrosivity. In the short-term testing, we changed test pieces every day and evaluated the corrosivity in the same manner.

To measure airborne chlorides, the "dry gauze method" specified in JIS Z 2382 (1998) was adopted. We placed a dry gauze plate inside of the outdoor test facility and replaced the plate with a new one once a month to measure the sea salt aerosols by ion chromatography. The area capturing sea salt aerosols was set at 100cm^2 according to JIS Z 2382. In short-term testing, we replaced the dry gauze plate everyday, which was installed outside the test facility (in front of the filter) to eliminate the effect of the filter, and measured sea salt aerosols by ion chromatography in the same manner.

3. Test results and evaluation of the results

3.1 Sea salt aerosols

Figure-1 shows monthly changes in sea salt aerosols and the filter trapping efficiency R from March 31, 2006 through January 10, 2007. The filter trapping efficiency R is defined as below:

$$R = \frac{A - B}{A} \times 100$$

Where,
R : Filter trapping efficiency
A : Sea salt aerosols entering inside of the test facility without filters
B : Sea salt aerosols entering inside of the test facility with filters

The solid line in Figure-1 indicates sea salt aerosols entering inside of the test facility without filters. As can be seen in the figure, amounts of sea salt aerosol tend to be lower in the summer season and higher in the winter season with a difference by a factor of over 30. The difference between seasons is expected to relate to the wind speed as will be mentioned later. Accordingly, in the winter season during which the wind speed is high, greater amounts of sea salt aerosol are dispersed in the air. The dashed line in Figure 1 shows the filter trapping efficiency. Figure 2 plots the filter trapping efficiency versus amounts of sea salt aerosol.

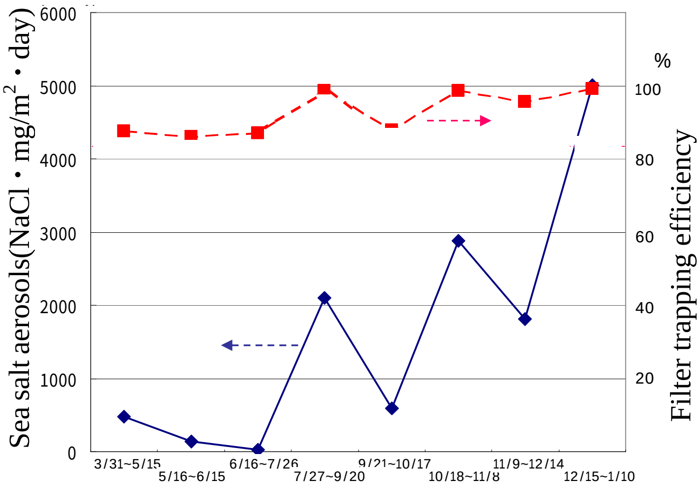

Figure 1 Seasonal changes in sea salt aerosols

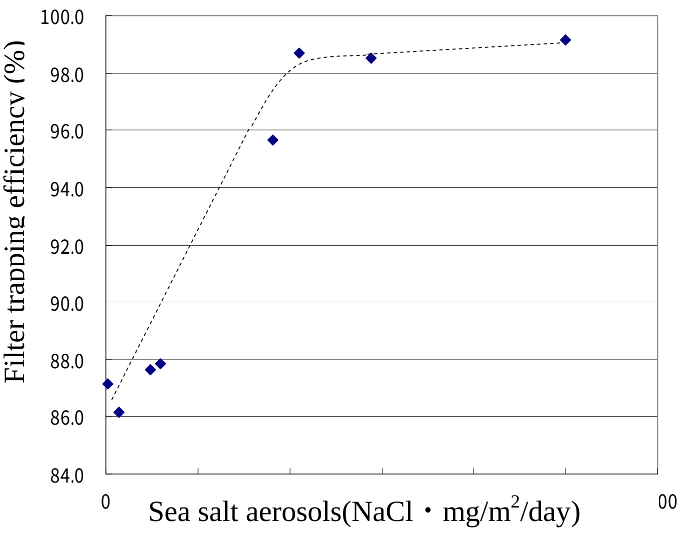

Figure 2 Sea salt aerosols and filter trapping efficiency

As shown in Figures 1 and 2, the filter trapping efficiency remains almost constant (i.e., 86-88%) when the amount of sea salt aerosol is small due to a lower wind speed. On the other hand, as amounts of sea salt aerosol increase, the filter trapping efficiency also increases getting closer to 99%. This may be because the filter has clogged and the pressure loss for replacement or the capacity of wind to be treated is exceeded.

We measured the amount of sea salt aerosol every day to clarify the relationship between the sea salt aerosol amount and environmental factors. The dry gauze method was also used in the daily measurement by placing the dry gauze plate in front of the test facility opening. In the short-term testing, we call the plate placed in the ocean side as the front gauze plate and that placed in the mountain side as the rear gauze plate. Figure 3 shows the daily measurements of sea salt aerosols. As can be seen in this figure, amounts of sea salt aerosol in the air largely vary depending on the day with the largest difference of about 40 times. The amount of sea salt aerosol measured in the front filter was about 20 times larger than that measured in the rear filter. However the ratio is not consistent. When sea wind blows, amounts of sea salt aerosol increase while land wind blows, amounts of sea salt aerosol are reduced. It is expected that the front gauze plate is mainly exposed to sea wind while the rear gauze plate is to land wind.

We evaluated the correlation between sea salt aerosol amounts and environmental factors. Figure 4 indicates the monthly average wind speed versus average amounts of sea salt aerosol. In this figure, average amounts of sea salt aerosol are represented by the star mark for each range of wind speed which is categorized every 0.5m. The figure suggests a clear correlation between the wind speed and sea salt aerosol amount. In this figure, the amounts of sea salt aerosol were measured inside of the outdoor exposure test facility without filters.

Figures 5 show the rainfall level versus sea salt aerosol amount. The sea salt aerosol amount marks the optimal value at the rainfall level of about 5mm. This suggests a hypothesis that seawater (sea salt aerosols) is dispersed in the air when rain drops hit the sea surface, as the rainfall increases, the amount of sea water aerosols dispersed in the air increases; in the beginning, an increasing amount of sea salt aerosol is dispersed in the air because only a limited amount of generated sea salt aerosol returns to the sea surface accompanying rain drops. When the rainfall exceeds a certain level, most of generated sea salt aerosols return to the sea surface accompanying rain drops and thus amounts of sea salt aerosol in the air are reduced.

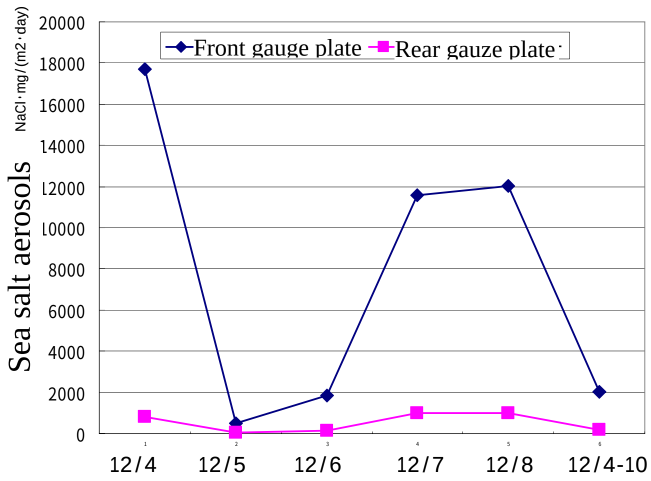

Figure 3 Daily changes in sea salt aerosol

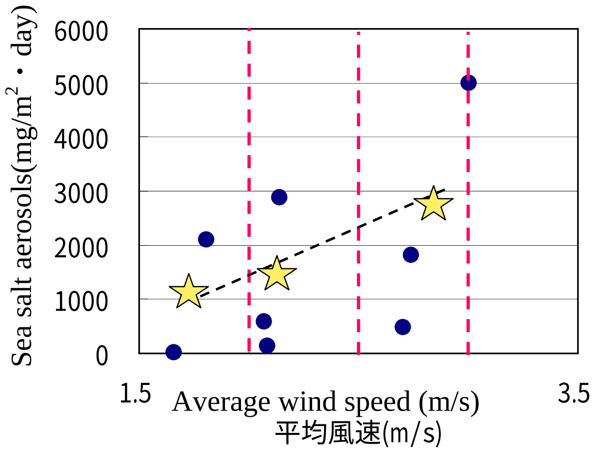

Figure 4 Wind speed vs. sea salt aerosols

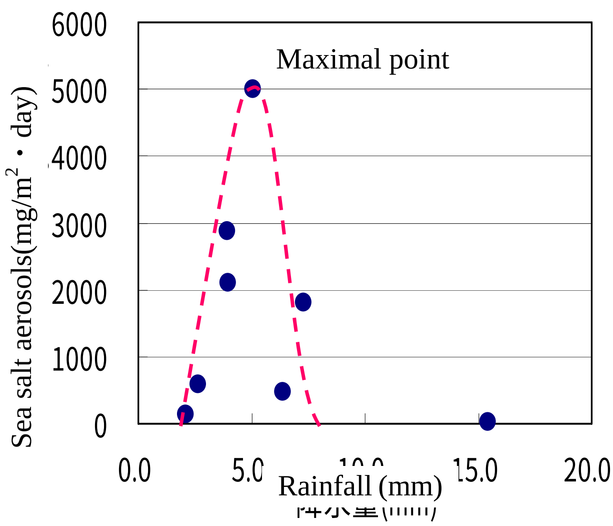

Figure 5 Rainfall vs. sea salt aerosols

3.2 Corrosivity in standard test pieces

We measured corrosivity in the standard test pieces, which were placed outside of the outdoor test facility, by the method in accordance with JIS-Z 2383. Figure 6 compares the measurements with those measured in Miyako Island, Choshi and Nishihara, which are described in JIS Z 2383. Our measurements reveal that the corrosivity measured during the period from March 31, 2006 through December 15, 2006 including the winter season was 1.7 times higher than that measured from March 31, 2006 through Sep. 21, 2006 which does not include the winter season. This difference is suspected to be because amounts of sea salt aerosol in the air increase during the winter season (See Figure 1).

It was found that the corrosivity we measured using the outdoor test facility installed Awara city, Fukui Prefecture was several times higher than that measured in Miyako Island. It is suspected that the location where the outdoor test facility was installed is highly corrosive. Accordingly, we believe that performing corrosion resistance tests using the specific test facility was worthwhile.

Figure 7 shows the corrosivity in the standard test pieces placed in the open air, test pieces placed under the shielded environment with a filter and test pieces place under the shielded environment without a filter. The x-axis shows the number of days elapsed since March 31, 2006. The y-axis represents the corrosivity which is standardized per year. As shown in this figure, the corrosivity per year for 259-day exposure including the winter season is higher than that for 174-day exposure. These results suggest that regarding the standard test specimens, the corrosivity of carbon steel increases under the environment in which a lot of sea salt aerosols are dispersed in the air.

Corrosivity under the shielded environment are much less than those in the standard test pieces which were subject to open air exposure. Furthermore, the corrosivity under the shielded environment with filters are lower than those under the shielded environment without filters. Although the use of filter reduces the amount of sea salt aerosol to 12% or less (See Figures 1 and 2), the reduction of corrosivity is about 50%. In addition, the corrosivity in standard test pieces subject to open air exposure increased by about 70% from 174-day exposure to 259 exposures. However, the increase in corrosivity was about 17% in the test pieces placed under the shielded environment with filters and about 15% without filters.

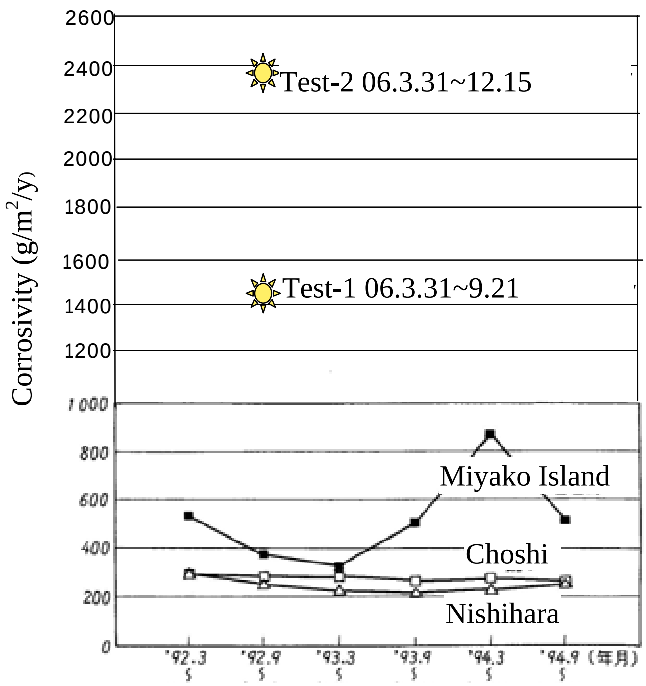

Figure 6 Corrosivity Comparison in standard test pieces

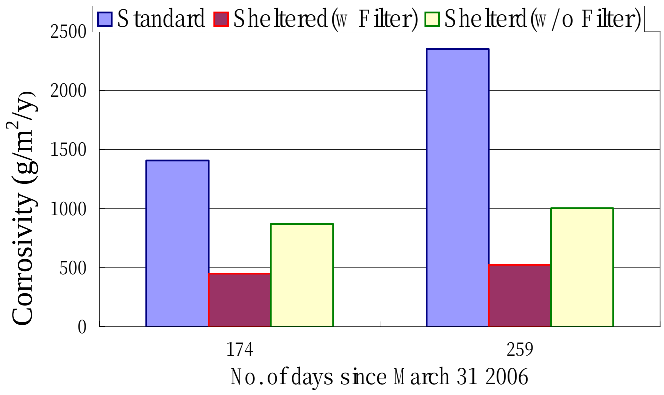

Figure 7 Corrosivity vs. exposure period

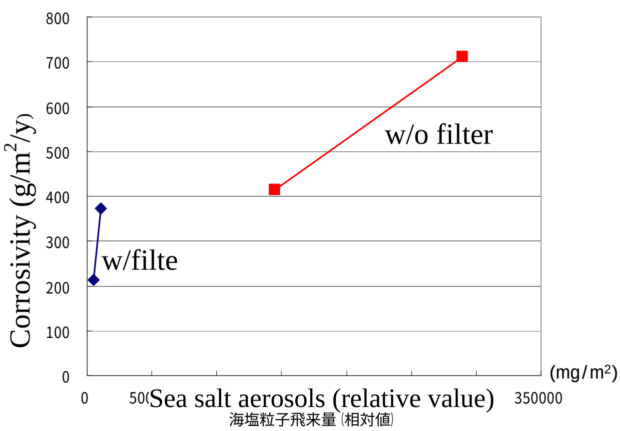

Figure 8 Amount of sea salt aerosol vs corrosivity

Figure 8 shows the correlation between amounts of sea salt aerosol and corrosivity, both of which are described in the cumulative amount not being standardized. As can be seen in this figure, the amount of sea salt aerosol increases by a factor of 2.1 during certain period when filters are used and by a factor of 2.0 during same period when filters are not used. The corrosivity also increases by a factor of 1.7 in both cases with or without filters. This suggests that although corrosivity in the initial stage varies depending on the environmental conditions.

4. Conclusion

We installed the outdoor exposure test facility on the sea coast in Fukui Prefecture, which belongs to the seashore district in the southern part of the Sea of Japan, and implemented both open air and shielded exposure tests. As a result we confirmed that:

- Amounts of sea salt aerosol in the air vary among the season.
- There is an approximate linear correlation between the amount of sea salt aerosols and average wind speed.
- Regarding the relationship between the amount of sea salt aerosols and rainfall level, a maximal value of sea salt aerosols exists against a certain level of rainfall.
- It is difficult to clearly define the relationship between the amount of sea salt aerosols and environmental factors including the sunshine hour and temperature.
- The sea salt aerosol trapping efficiency of the filter, which is about 88% when the amount of sea salt aerosols is small, increases to almost 99% when the amount of sea salt aerosols is large.
- The corrosivity on the seashore of Awara city was higher than that in Miyako Island.
- Although the use of filter reduced the amount of sea salt aerosol in the air to 12% or less, the reduction of corrosivity was about 50%.
- In both cases with and without a filter, the cumulative corrosivity increases by a factor of 1.7 as the cumulative amount of sea salt aerosols in the air increases by a factor of 2.

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COLLABORATIVE MACHINING SOLUTION EXTENDS THE OPERATING LIFE OF A NUCLEAR POWER PLANT

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ABSTRACT

Examination of a CANDU 6 nuclear power plant's steam generators during a scheduled maintenance outage revealed that the manway ports, part of the ASME Section III, Class 1 pressure boundary, needed repair. The port's inner cover gasket was not seating properly. Integrity was at risk. It was determined that this operation would required a specialized machine to successfully repair the man-way port.

The solution included the modification of a standard portable boring machine with a custom mounting option to enlarge the counterbore in the primary head shell from a round shape to an obround shape (76mm of shell thickness, 16mm radially). The shape change was needed to accommodate the new obround cover and gasket seal design. Once the new major shape was machined, the repair was finished with a Computer Numerically Controlled (CNC) machine developed by the service team to achieve the necessary gasket face location and sizing.

The final result met all of the plant's expectations and was completed well within the time allotted during the maintenance shut down. This success was due to the positive partnership and collaboration of the service team and the machine tool manufacturer working together to successfully extend the operating life of the nuclear power plant.

1.0 Background

Examination of a CANDU 6 nuclear power plant's steam generators during a scheduled maintenance outage revealed that the manway ports, part of the ASME Section III, Class 1 pressure boundary, needed repair. The port's inner cover gasket, which measured 355 mm by 456 mm, was not seating properly. Integrity was at risk. The solution team needed to determine how to balance the geometry of the port and redesign the covers' gaskets.

In addition, the steam generator's hemispherical head was developed out of forged steel and while the exact properties of the material were unknown, it was obvious that the metal would provide substantial machining challenges. It was determined that this operation would require a specialized machine to successfully repair the man-way port.

2.0 Machine Design Specifications

The service team turned to a specialized manufacturer of on-site machine tools for initial engineering guidance and solution recommendations. A team comprised of service team engineers and project managers working with the machine tool manufacture's engineering specialists developed a cutting repair solution to quickly and accurately machine the difficult material.

Figure 1: The boring machine with 115V variable electric drive and variable feed with special fixturing for mounting to the bottom of a steam generator.

First, the team needed to determine the project requirements. They knew the tool had to operate in a tight space and would need to withstand cutting tough material. Second, the solution had to perform a dry cut. Third, the tool needed to be rigid, accurate, and quick to set up. Finally, with a tight 24-hour time allotment to complete the machining, the machine also needed to cut speedily.

The tool's mounting requirements were another consideration. The team needed one stationary bracket to mount the tool because it would be in a crowded area where the primary head of the generators was surrounded by process piping and system components. This made access difficult. In addition, to accommodate unforeseen machining adjustments, on-site, the tool needed to include several interchangeable cutting heads.

3.0 A Standard Machine Redesign

Meeting all these machining requirements limited the engineers' tool options. After much assessment and analysis, the team decided to modify a standard boring machine. The tool's ability to program precise cutter movements provided the flexibility necessary to perform onsite cutting of material with unknown qualities. The machine also offered precise control of the spindle RPM, allowing feed rates and cutter movements to be fine-tuned by the operator at the repair area.

The boring machine incorporated changeable tooling and cutting technology to provide accurate cuts and inserts. An adjustable mechanical stop and an incremental adjustment process were machined into the tool head and bit. The cutting head was automatically fed axially on a traveling bar using the standard axial feed screw with mechanical stops. The radial feed was manually adjustable using a tapered locking mechanism. The machine also included a 108 mm diameter x 1219 mm long chromed bar, a rotational drive unit and an axial feed unit. The boring machine also featured an electric drive motor, 115v 50/60 Hz, with two-speed gearboxes and an 115v remote control pendant with variable speed and stop start.

To meet the job's difficult attachment concerns, a special bearing mount was developed into a slide mechanism and attached to a modified version of the hydraulic chuck supplied by the service team. The modified chuck incorporated a slide, which held the bearing and allowed it to move from one side of the bore to the other. The chuck was mounted using stops and screws at either end of the slide. A passage bore cut into the chuck enabled the hydraulic line to connect and move to either side of the bore.

The service team engineers supplied the machine tool manufacturer with a 3-meter hydraulic hose with a quick connect for attaching to a hydraulic pump for activating the chucking system. A service team manifold attached to the top of the slide mechanisms, and the lines were routed away from the bar using hose brackets and looms to provide stability to the machine once it was mounted on-site.

Figure 2: The boring machine with special I.D. mounting chuck and tool head for accurately positioning the cutting tool for boring a round hole to make it obround.

A special standoff bracket supported the bar's other end and used the manway pivot block as the primary mount. The fixture was anchored against the wall of the steam generator and the plant's structural steel. Adjustable legs with jacking feet were expandable to secure the fixture in place. The team designed a slide mechanism into the standoff bracket for positioning the bar on both ends.

Once the new boring tool was completed, the service team's operational staff visited the machine tool manufacture's training center for a full education on the tool's capabilities. Machinists tested the boring tool by simulating the repair on a replication of the repair site. This testing and training process provided the service team with a level of confidence that the tool could complete the work to specification within the allotted time. Having the opportunity to learn how to use the machine before on-site work began also helped reduce any on-site guesswork by machinists.

4.0 Approval Cycles

Before deploying the tool, the machine tool manufacturer proved the boring machine's capabilities to a validation committee comprised of the service team and personnel from the nuclear plant. The acceptance criteria were based on set-up times, cutting rates and reliability of operation over what was required at the site. This was deemed necessary to ensure that the equipment could complete the machining functions successfully, considering the uncertainties of the vessel material properties.

The final solution met expectations and passed a test plan before any work was completed.

5.0 The Repair

The machinists removed approximately 76 mm of shell thickness, 16 mm radially to enlarge the counter bore in the primary head shell from a round shape to an obround shape. The shape change was needed to accommodate the new obround cover and gasket seal design.

Once the new major shape was machined, they completed the repair by using a service team developed Computer Numerically Controlled (CNC) machine to achieve the necessary gasket face location and size. This machine also accurately produced the desired smooth finish of the obround shape.

6.0 The Result

The final repair results met all of the nuclear power plant's expectations. Together, the teams exceeded the exacting cutting requirements and beat the time requirements.

By working together, the service team and the machine tool manufacturer efficiently and collaboratively defined the requirements and customized the tool according to specifications including space and location criteria. Additionally, they shortened the development process and accounted for all potential problems during the machining process. As a result, the machine tool manufacturer completed the machine from quote to finished job in 12 weeks — well within the targeted outage time. Together, the service team and the specialty machine tool manufacturer formed an effective partnership that was able to quickly complete the repair and insure continued operation of the nuclear power plant for many years to come.

ENC 2007

European Nuclear Conference
16-20 September 2007, Brussels, Belgium

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Layout and Design: Marion Brünglinghaus, ENS