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Sustainability of Nuclear Energy System Based on Evolutionary Reactors

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INPRO
International Project on Innovative Nuclear Reactors and Fuel Cycles

Organized by the IAEA Department of Nuclear Energy

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1. INTRODUCTION

1.1. Overview

The 7th INPRO Dialogue Forum on the '*Sustainability of Nuclear Energy Systems Based on Evolutionary Reactors*' was convened on 19-22 November 2013. The activity was implemented under Project 4, Activity 1.1.4.4/3 of the IAEA Programme & Budget 2012–2013. The 7th Dialogue Forum was designed to develop and promote INPRO assessment services to Member States to help in their long range and strategic nuclear energy planning. The Dialogue Forum brought in technology holders of evolutionary reactors to demonstrate exemplary INPRO Methodology assessments of their designs to familiarize Member States with the INPRO Methodology and NES assessment (NESA). Preparatory meetings were held with participating vendors in the 3rd quarter of 2013 to prepare assessment materials for the Dialogue Forum.

The event was attended by representatives from 32 technology holder and technology user countries and 12 reactor designers / experts of evolutionary reactor technologies from AREVA, Candu Energy Inc., GE Hitachi Nuclear Energy International, KEPCO Engineering and Construction and KHNP, Mitsubishi and Rosatom. The reactor vendors voluntarily prepared sustainability assessments of safety aspects of their reactor technologies using the INPRO Methodology. Member States had the opportunity to discuss availability of design information for a NESA and also approaches to engage designers in a NESA.

1.2. Opening

The Dialogue Forum was opened by Mr Alexander Bychkov, IAEA Deputy Director General for Nuclear Energy and INPRO Project Manager. Mr Bychkov described the forum as an unusual and excellent opportunity for Member States to meet and discuss issues related to sustainability of nuclear energy systems directly with vendor experts. Member States could use the event to discuss the possibility of vendor expert cooperation in nuclear energy system assessments (NESA) using the INPRO Methodology.

INPRO Group Head, Mr Zoran Drace, provided an overview of INPRO activities in support of transition to sustainable nuclear energy systems. He started with reiterating the establishment of INPRO in 2000 as authorized by the IAEA General Conference resolutions (GC(44)/RES/21), followed by INPRO basic characteristics as being a membership-based and funded mainly from extra-budgetary contributions dealing with international project with cross-cutting issues with all relevant technical Departments in the Agency. INPRO cooperates with Member States to ensure that sustainable nuclear energy is available to help meet the energy needs of the 21st century. INPRO also brings together technology holders and users to jointly consider international and national actions required for achieving desired innovations in nuclear reactors and fuel cycles. The service provided by INPRO is part of the integrated services of the IAEA for Member States considering initial development or expansion of nuclear energy programmes. Mr Drace discussed INPRO membership from 2001 until 2013, and then discussed the four INPRO Projects in the current budget year of 2012-2013 as well as plan for 2014 dialogue forum. More details of INPRO Projects can be viewed on http://www.iaea.org/INPRO/.

2. PROGRAM OVERVIEW AND OBJECTIVES

The Dialogue Forum was chaired by Mr Juergen Kupitz (Germany) who conveyed the objectives of the Dialogue Forum which were to:

- Familiarize Member States with the INPRO Methodology, focused on the sustainability assessment of NES, to support long-range and strategic planning of a successful nuclear power program.
- Discuss assessment of evolutionary nuclear reactors in the areas of safety of nuclear reactors.
- Discuss case studies and lessons learned as well as support to the on-going and future projects on national NESAs.
- Discuss salient features of evolutionary nuclear reactors relevant to sustainability of NES, and the important role of vendors in a NESA – particularly in providing necessary data and familiarization with the design to support the assessment.

Mr Kupitz noted participation of vendors to deliver sample assessments of their evolutionary reactors and their important roles in the meeting to validate the INPRO Methodology, provide design information and help to reduce assessor's efforts in INPRO assessment in the area of reactor safety. The vendors had selected a number of criteria and evaluation parameters for their 2-hour lectures. The Chairman put an emphasis that

the forum would focus on progress of evolutionary design for each vendor, and not on the comparison of designs of different vendors. Participation of Member States in the forum was in the form of responding to the questionnaire, providing questions throughout the forum and delivering a short statement on the country's plan for deployment of NPP in the next 10 to 20 years, the extent of discussion of sustainability of national NES, overall comments on the event and some recommendations. Overall, it was expected that the forum would promote a dialogue between MSs and vendors on NES assessment (NESA) using the INPRO Methodology. INPRO should carry out follow up activities based on the recommendations submitted by the Member States, e.g. on topics for the next Dialogue Forum, regional workshop.

The agenda was outlined as follows:

1. Presentations from the IAEA on the INPRO Methodology and Safety of Nuclear Reactors
2. Presentations from Vendors on assessment of evolutionary reactor designs for selected INPRO criteria in the area of safety of nuclear reactor
3. Q&A session on Friday, 22 November 2013 to be addressed by Vendors based on selected questions from Member States. *Question Template was provided in the folder.*
4. Short Statements from Member States on Friday, 22 November 2013. *Statement Template was provided in the folder.*
5. Summary of Dialogue Forum by Chairman. *To be supported by Rapporteurs.*

3. SUSTAINABILITY ASSESSMENT OF NUCLEAR ENERGY SYSTEM (NES) USING THE INPRO METHODOLOGY

3.1. INPRO Methodology for Sustainability Assessment of Nuclear Energy System (NES)

Mr Jon R. Phillips from the INPRO Group delivered a lecture on INPRO Methodology. He explained that INPRO methodology was derived from the concept of sustainable development within the UN system. This concept is defined in four dimensions dealing with societal, economics, environmental and institutional aspects, leading to the need for sustainable energy supply and energy system planning which brings down to the role of nuclear power in sustainable energy supply mix. This results in the derivation of a NESA using the INPRO Methodology.

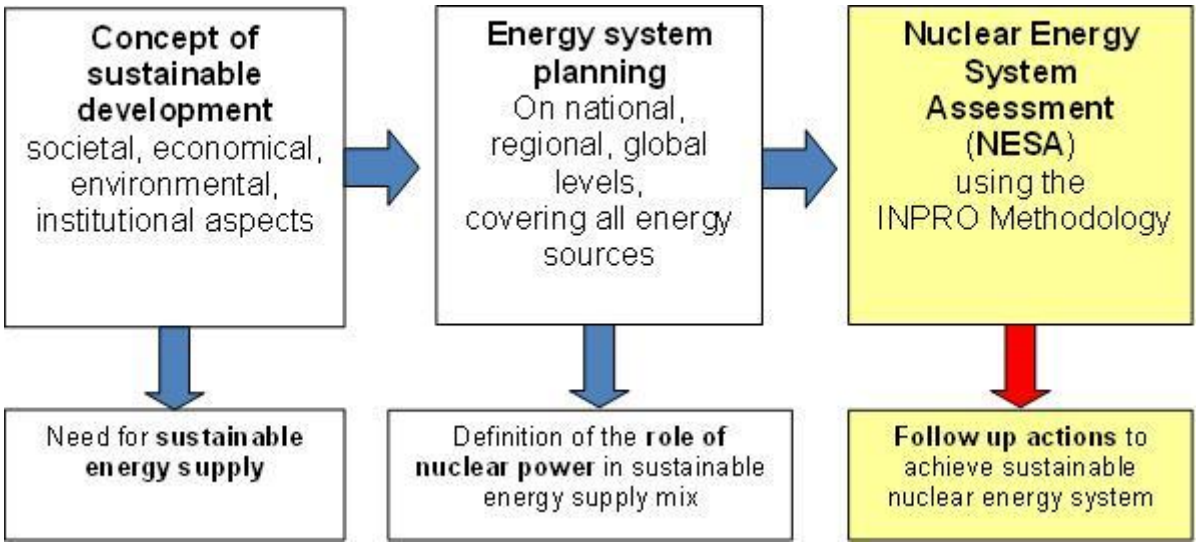


FIGURE 3.1.1. Sustainable Development, Energy System Planning and NESA

He described the structure of methodology, indicating seven key issues that influence sustainability of nuclear power, namely safety, cost, proliferation, security including sabotage prevention, waste and environment and resources. Additional key issue is the overarching issue of infrastructure to support all of the others. They are all translated to the 7 areas of INPRO Methodology assessment guidance manuals.

Mr Phillips showed the holistic nature of the NESA, i.e. that NESA covers all areas relevant to NES sustainability, all reactor types and fuel cycle facilities, all facilities of a NES, and all phases on a NES, from

cradle to grave. But he explained that NESA is not always implied as a full scope NESA down to criteria level, but sometimes it is applied in partial or specific way.

The methodology and criteria driven assessment comprise 3 levels: (i) 14 Basic Principles (BPs) as goals for development of sustainable NES, (ii) 52 User Requirements (URs) indicating what should be done by designer, operator, industry and State to meet goal defined in Basic Principle, and (iii) about 150 Criteria (CR) with indicators (IN) and specific acceptance limits (ALs) to check whether a User Requirement is met. Mr Phillips further described at the very highest level, the Basic Principle level message sustainability metrics for each area.

1. Economics: Nuclear energy products must be competitive within the energy portfolio available in the country.
2. Waste Management: Nuclear waste must be managed so that human health and environment are protected and undue burdens on future generations are avoided.
3. Physical Protection: Efficient graded PP regime protecting all materials, facilities and operations implemented for whole life cycle of NES.
4. Proliferation Resistance: NES must remain unattractive to nuclear weapon program through implementation of technical features combined with effective and efficient institutional measures.
5. Environment: Impact of stressors from future NES must remain within performance envelope of current NES. Resources must be available through end of 21st century.
6. Safety: Near term deployments of NES facilities should show demonstrable improvements in safety over existing facilities
7. Infrastructure: Assure adequate infrastructure and efficiently create and maintain infrastructure. This includes legal and institutional frame work, supporting industrial economic infrastructure, and socio-political infrastructure (public acceptance, human resources).



FIGURE 3.1.2. Architecture of INPRO Methodology

The INPRO user requirements are directed at different audiences: (i) State (government institutions), (ii) Operator of nuclear facilities, (iii) Designer or developer of nuclear facilities, and (iv) National industry involved in the nuclear power program. The input data needed for the evaluation of the URs should be provided by the responsible organization. He noted that NESA is very difficult for academic to perform because of the breadth and depth. It can be done by the right subject matter experts.

There are 3 different types of NESA users: (i) Newcomers as first time users, (ii) Experienced users, and (iii) Developers/designers of nuclear technology. When requested, developers/designers may assist newcomers / experienced users in a NESA.

There are 3 different levels of depth and scope in a NESA:

- NESA as a learning tool: Increase awareness of long-term nuclear sustainability issues (newcomers)
- Limited scope NESA: Selected areas of INPRO methodology and/or selected components of NES (needs driven)

- Full scope NESA: All areas of INPRO methodology, full depth of assessment, complete NES development of sustainability strategy and long-term planning

The main outcome of a NESA is a sustainability metric of defined NES through identification of gaps and creates the 'gap map'. The purpose is not to criticise the system, but work in the area where improvement is useful. Sustainability is defined as finding the areas and making progress, i.e. finding direction to sustainability, and it is incremental in nature. The best use is follow-up actions to close the gaps, a discovery process for the Member States even for experienced countries. The gaps found help define a path to a sustainable NES.

On effort to perform a NESA, the most important step is to have the correct subject matter experts so it can be done very efficiently. It is hard to make an estimate.

Mr Phillips discussed 3 options on how NESA is performed by technology user as self-assessment: (i) Support from IAEA/INPRO, (ii) Support from IAEA/INPRO and Technology Holder to provide collection input for design related URs, and (iii) Support from IAEA/INPRO and Technology Holder to provide access for design related URs (engaging in the process). The first approach can be challenging unless the technology users have tremendous experience with the technology that they are proposing in the NES.

A total of 6 national assessments (Argentina, Brazil, India, Republic of Korea, Armenia, Ukraine) and 1 multinational assessment (sodium reactor, closed fuel cycle) have been documented respectively in IAEA Tecdoc-1636 and Tecdoc-1639. The Belarus NESA presents a full scope assessment of simplified NES consisting of power plant and spent nuclear fuel / waste management facilities, and it has been documented in Tecdoc-1716. There are currently 3 on-going NESAs: (i) NESA in Indonesia (full scope – reactor and fuel cycle; technical draft is to be completed in 2014), (ii) NESA in Ukraine (limited scope, to be completed in 2014) and (iii) NESA Romania (full scope, training has been provided, to be completed in 2016). One NESA project is under discussion with Kazakhstan.

In the final part, Mr Phillips informed NESA support package available to assessors based on feedback from assessor as documented in Tecdoc-1636, training on INPRO Methodology application, examples of input data for INPRO assessment in the area of Economics, Infrastructure, Waste Management, and Safety of Nuclear Reactor, e.g. safety data for AP1000 from US NRC website. There are also examples of full scope assessment by Belarus, NEST tool for Economics analysis, and List of design data to be provided by designer. Continuous access to IAEA and MS expertise is available via INPRO Group.

The discussion session is summarized below:

Questions	Answers
Mr Hj Khalid (Malaysia) asked if NESA experience would help with public acceptance	**Mr Phillips (IAEA)** explained that NESA contains resourceful detailed information that can be used for those responsible for public acceptance. It is up to Member States who perform a NESA how they want to use the NESA.
Mr Burkart (USA) asked if the methodology is far more useful for countries with already established strategic partnership rather than countries who are still in the bidding process. This limits the use of INPRO Methodology. So he asked if there is a way of broadening the methodology without becoming complicated, and being useful for countries who are in (Milestones) Phase 2 rather than country with already selected vendor.	**Mr Philips (IAEA)** had the opinion that it will depend in the scope of a NESA. A beginner country is not advised to conduct a full scope NESA considering of the technicality aspect and instead to opt for the first level NESA (as a learning tool). So INPRO will not propose a country to immediately jump into a full scope NESA to answer sustainability. Rather, NESA is ought to be seen as a learning tool to understand what questions/technical questions to be asked. If Member States want to do a full scope NESA, then it is far more useful for countries with already established strategic partnership.
Mr Burkart (USA) indicated that efforts are needed to explain to countries not doing a full scope NESA, that it would help their job in earlier process.	**Mr Drace (IAEA)** commented that INPRO Methodology may overlap but is a complementary to the Milestones approach. At the early stage, INPRO Methodology can provide preliminary information without the need to collect technical design, to understand what a country is getting

	into, and what it takes to attain sustainability. It was noted that INPRO methodology does not discuss how to do the bidding, deal with government organizations, etc as these are the scope of Milestones. Rather Member States are advised to perform a limited NESA in between Milestones, e.g. before entering into construction, to learn more details. If Member States still cannot understand the different layers in IAEA, they should look for supports from the IAEA.
Mr Silva (Thailand) asked about the key issues that a country like Thailand should select if it wants to perform a limited scope NESA. All areas seem are important.	**Mr Phillips (IAEA)** suggested economics (macroeconomics, knowledge of energy situation), waste management and infrastructure. For infrastructure, he noted the difference between Infrastructure provided by INIG and INPRO. The INPRO Manual serves as a matrix to measure the preparation, but not an active service like INIR mission. The INPRO Manual provides guidance on the kinds of questions to ask and answers to have, so it is more like a map. He considered safety of nuclear reactor is very technical, which cannot be done by superficial assessment. There are several areas in which it is not realistic to provide simple answers. The manual, however, can be used for awareness from the point of view of sustainability.
Mr Tjahjono (Indonesia) stated that in INPRO assessment in the area of safety, a lot of data required for the analysis, and sometimes the data is available in the internet. He wanted to know how to select the best data in the internet, given the much information from many sources.	**Mr Korinny (IAEA)** indicated the assessor should be the expert with certain expertise. He suggested that national experts should communicate continuously with INPRO experts, in which based on past practice the assessor succeeded in the selection of data. **Mr Phillips (IAEA)** added that presence of designer in the Dialogue Forum served as a very good example to obtain such data.

3.2. Safety Assessment and NPP Design Requirements

Mr Hussam Khartabil from the Division of Nuclear Installation Safety (NSNI), Department of Nuclear Safety, began his presentation by providing some observations on the INPRO Safety Manual. The INPRO BPs, URs and CR were developed based on the following IAEA Safety Standards: (i) Fundamental Safety Principles, (ii) Requirements for the Safety Assessment for Facilities and Activities, and (iii) Requirements for the Safety of Nuclear Power Plants. These requirements are applicable to existing and new reactors as well as innovative designs to be built later on. In the INPRO Methodology, the assessment is performed on a new design against a reference design, which according to the INPRO manual should be in operation by 2004. Both reference and assessed designs are expected to comply with applicable requirements whether the IAEA standards or national requirements. Therefore, INPRO Methodology is not looking at compliance with the IAEA safety standards requirements, but more at long term sustainability as defined by INPRO. INPRO asks for continuous improvements of the assessed design over a reference design, as more knowledge or operating experience is gained. Because the INPRO assessment uses the results of a formal safety assessment (e.g., safety analysis report) as input, it is necessary for an INPRO assessor to be familiar with applicable requirements that both reference and new designs must comply with.



FIGURE 3.2.1. Structure of the IAEA Safety Standards

The structure of the IAEA Safety Standards comprising Safety Fundamentals, Safety Requirements and Safety Guides, was briefly discussed. The Safety Fundamentals states the main safety objective which is to protect people and the environment from harmful effects of ionizing radiation. From the safety objective, there are 10 Safety Principles are derived as given in the Fundamental Safety Principles (SF-1). INPRO uses six of these Safety Principles to define INPRO Basic Principles in the area of safety of nuclear reactors, namely:

- Leadership and Management of Safety
- Optimization of Protection
- Limitation of Risks to Individuals
- Protection of Present and Future Generations
- Prevention of Accidents
- Emergency Preparedness and Responses

Mr Khartabil then introduced documents on safety assessment and NPP design requirements, in the context of design and review process for NPP. The design requirements, i.e. safety requirements and other requirements such as performance and economic, serve as inputs. Based on the requirements, the initial design is assessed: if the result is acceptable, then the assessment continues to the next stage; if not, then the design must be modified. For the safety assessment part, there are a number of safety guides to help interpret the requirements.

He then went on to discuss the two sets of documents. First is the Safety Assessment for Facilities and Activities (GSR Part 4), which is a generic document that is applicable to all activities and facilities that have the potential to result in radiation hazards. The facilities include NPPs, research reactors, critical assemblies, fuel cycle facilities, irradiation facilities, etc; whereas the activities include production, use, import and export of radiation sources, transport of radioactive material, decommissioning and dismantling of facilities, etc. Second, the requirement documents have a number of requirements, each with an explanation. The first document (GSR Part 4) contains 24 requirements, in terms of the safety assessment process. The first four requirements are the starting point for doing the safety assessment, namely, the graded approach, scope, responsibility and purpose of the safety assessment. There are then features that must be assessed. The features are assessed using, among others, safety analysis (e.g. deterministic and probabilistic analysis). There are also requirements for assessment of defence in depth, documentation, independent verification, management of the safety assessment, use of the safety assessment (e.g. establishing limits for operation) and maintenance of the safety assessment (e.g., periodic safety review for the duration lifetime of facilities). The IAEA Safety Assessment document (GSR Part 4) is supported by a number of Safety Guides, namely: Deterministic Safety Analysis, Development and Application of Level 1 and 2 Probability Safety Assessment, and Format and Content of the Safety Analysis Report for NPPs.

The second set is the design requirements for NPP containing more specific requirements. The set entitled ‘Safety of Nuclear Power Plants: Design’ (SSR-2/1) replaces the previous NSR-1, incorporating several changes. The content includes an Introduction section and a section on Applying Safety Principles and Concepts, which refer to the Safety Objectives, Radiation Protection, and Defence in Depth. The remaining sections contain a total of 82 requirements: Management of Safety in Design (3 requirements), Principal Technical Requirements (9 requirements), General Plant Design (5 categories with a total of 30

requirements, and Design of Specific Plant Systems. The first three categories are applicable to any kinds of reactors. The last category is more specific, and influenced by experience of large water-cooled reactors. Overall, the document is valid for any reactor design but certain requirements may need reinterpretation for innovative designs, e.g. design with a coolant other than water. The following are examples of IAEA Safety Guides that can be used to interpret the safety requirements: Software for Computer Based Systems, Instrumentation and Control Systems, Design of Fuel Handling and Storage Systems, External Events Excluding Earthquakes, Seismic Design and Qualification, Protection Against Internal Fires and Explosion, Design of Emergency Power Systems, Design of the Reactor Coolant System and Associated System, Design of Reactor Containment Systems, Protection against Internal Hazards other than Fires and Explosions, Design of Reactor Core, and Radiation Protection Aspects.

The IAEA has established a Generic Reactor Safety Review (GRSR) service, looking at whether requirements from the safety assessment (GSR Part 4) and design (SSR-2/1) are addressed in the safety analysis report. The service can be carried out at any design stage using a graded approach, for a new reactor from conceptual design to a design supported by a comprehensive safety analysis report. In order to do the review using the two documents, it is necessary to combine the requirements from GSR-4 and SSR-2/1. To organize the review process, one can start with GSR-4 (with smaller number of requirements), combined with relevant requirements from SSR-2/1, and group them for simultaneous assessment. The remaining requirements from SSR-2/1 are then assessed as supplementary requirements specific to nuclear power plants. The process of the GRSR service involves looking at all the requirements and grouping them, interpreting the requirements using appropriate safety guidance and other documents, looking at the safety analysis report (SAR) and making judgement if the requirements are adequately addressed. The review is done by a group of experts, given the many different areas to be covered in SAR.

There are 7 examples of safety requirements provided to further illustrate the process:

- GSR-4 Requirement 1: Graded Approach
  For NPP, the status of development stage needs to be taken into consideration. If the design is fully mature, the SAR should cover everything. If not, there are minimum things to be included in the SAR, e.g. plant layout and systems, acceptance criteria are defined, preliminary analyses given the available tools, recognize areas needing further work, etc. For a mature design (e.g. designs presented in Dialogue Forum 7), all information should be included.
- GSR-4 Requirement 13 and SSR-2/1 Requirement 7: Defence in Depth
  GSR-4 states adequate provisions have been made at each level of DID without defining them for a NPP. SSR-2/1 requires more details, e.g. 5 levels of DID, statement on independence of DID, etc.
- Plant States (changes in new design requirements):
  The new document removes the term Beyond DBA (BDBA) and introduces the term Design Extension Conditions, where the Design Extension Conditions have to be addressed in the design of the NPP.
- SSR-2/1 Requirement 19: Design Basis Accidents
  The requirements for DBA are the same and the analysis is similar with method used in the past using the conservative manner.
- (SSR-2/1 Requirement 20: Design Extension Conditions
  For Design Extension Conditions, the design should further improve the safety of the NPP, e.g. to consider more severe accident than DBA or accidents that involve multiple failures; and the analysis methodology used is best estimate as normally used for BDBA.
- Qualitative success criteria for Design Extension Conditions
  The first criteria used for success in the Design Extension Conditions are significant releases have to be practically eliminated. The requirements document gives definition for practical elimination. There are two conditions, either one to be satisfied: (i) physically impossible to happen, or (ii) one has to be sure with the high degree of confidence that any conditions that could lead to large releases are extremely unlikely to happen. For the latter, if one wants to use probabilistic criteria, the number obtained must be accurate enough. But the requirements document does not propose any values, so it is up to the national regulators to decide. Another success criteria for DEC is that integrity of containment must be maintained with conditions that cannot be practically eliminated should result protective measures that are limited in terms of area and time needed to deal with the situation. The requirements document is mostly applicable to new NPPs. For existing NPPs, the document states the existing NPPs are not expected to fully comply since they are based on the previous

requirements. However, the document recommends to use the new requirements as benchmark if any operational / feasible improvements can be made to bring them closer to compliance with the new requirements.

- GSR-4 Requirement 10: Assessment of Engineering
  The requirements have several paragraphs, e.g. use of proven design and components, application of DID, classification of systems and components. One paragraph of interest to the INPRO Group is about the use of innovative improvements to a certain design. The requirement is that it has to be supported by appropriate R&D program, which is consistent with the requirement that proven system has to be used. The INPRO Methodology also has some reference to this.

In conclusion, an INPRO assessor in the area of safety needs to be familiar with relevant IAEA Safety Standards to conduct INPRO assessment, namely: (i) Safety Fundamentals (SF-1), (ii) Requirements for safety assessments (GSR-4), (iii) Requirements for NPP design (SSR-2/1), and (iv) Applicable safety guides. The GRSR service can be used to assess compliance with IAEA safety assessment and design requirements for new designs and can be applied at any stage of development of the new design.

The discussion session is summarized below:

Questions	Answers
Mr Zhou (China) asked about gaps - not reach the criteria. When doing safety assessment, many requirements are qualitative parameters. So how to define gaps or to understand gaps?	**Mr Korinny (IAEA)** explained that gap is used in INPRO assessment safety, and it is not the same with assessment using the IAEA Safety Standards. Gaps in INPRO mean that certain requirement or criterion is not fulfilled, either logic or quantitative criterion is not fulfilled (not get necessary number). For 'yes' or 'no', if the answer is 'no' then the criterion is not met. Again, one has to distinguish between INPRO assessment in the area of safety and safety assessment using the IAEA Safety Standards.
Ms Yang (China) asked about meeting the criteria: 'yes' and 'no'.	**Mr Korinny (IAEA)** stated that if the criterion is not met, then it is a gap.

3.3. INPRO Methodology Requirements on Safety of Nuclear Reactors

Mr Andriy Korinny from INPRO provided an introduction discussion in the area of safety of nuclear reactors so as to provide better understanding to participants on the presentations to be delivered by invited vendors. He described briefly the hierarchy of INPRO methodology, comprising basic principles, user requirements and criteria. BPs set out goals which need to be met for a NES to be sustainable. URs define actions which need to be done to achieve the goals. CR can be used by the assessor to check whether the given URs are fulfilled or satisfied. The INPRO assessment is a bottom-up approach, meaning that the assessor checks all criteria to define whether URs are satisfied, when URs are satisfied, the goals defined in the BPs are also achieved.

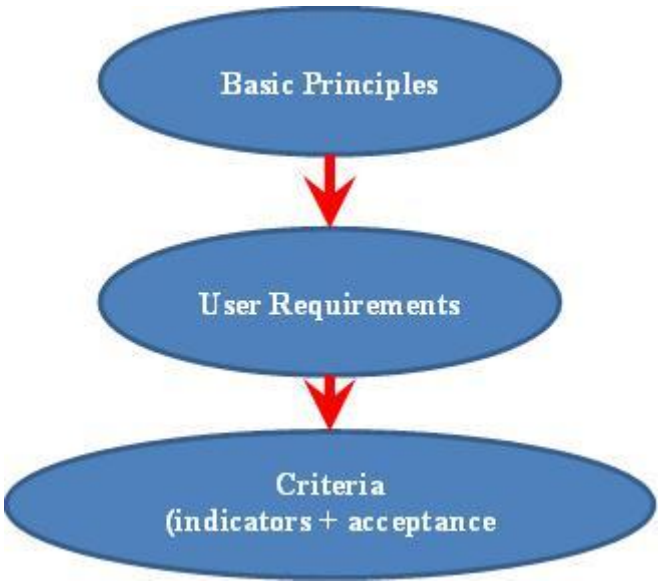


FIGURE 3.3.1. Hierarchy of INPRO Methodology

INPRO assessment in the area of safety is based on comparison between new/planned reactor being assessed and operating reactor of the same design/origins. Thus INPRO does not compare different technologies but rather looks at the progress within a given technology. Safety is the largest and the most complicated INPRO area, and accounts for a third of the total assessment items in the INPRO Methodology. The current operating reactors are operating at high standards, whereas the new evolutionary reactors are even more superior than the operating ones. So the assessment becomes quite challenging and detailed study is necessary to find potential gaps. For this, NESA support package is developed to reduce the efforts, and the materials from the Dialogue Forum 7 meeting are to be included in the support package.

Mr Korinny noted that INPRO requirements in the area of safety are based on the contribution from hundreds of national experts, and the requirements comply with the IAEA Safety Standards using same fundamental safety objective to protect people and the environment from harmful effects of ionizing radiation (SF-1) and the following measures to meet the objectives:

- Control radiation exposure of people and the release of radioactive material to the environment;
- Restrict the likelihood of events that might lead to loss of control over the reactor core and other sources of radioactive material associated with the reactor core;
- Mitigate the consequences of such events if they were to occur.

The methodology is also based on extrapolation of current trends assuming a large increase of nuclear power in the 21st century.

The INPRO Methodology uses the fundamental safety principles in the IAEA Safety Standards, introduced in INSAG-12 (Fundamental Nuclear Safety Principles), which are distributed into 3 groups:

- Group 1: Management responsibilities
  These include safety culture, responsibility of operator, and regulatory control.
- Group 2: Strategy of defence in depth
  These include defence in depth, accident prevention and accident mitigation.
- Group 3: General technical principles
  These include proven engineering practices, radiation protection, quality assurance, human factors, peer reviews, etc.

Group 1 is covered in the INPRO area of Infrastructure, whereas Group 2 and Group 3 are covered in INPRO area of safety of nuclear reactors, i.e. the safety area only discusses design-related safety principles, which was the scope of the meeting.

The INPRO Methodology in the area of safety focuses on the confirmation the long term sustainability of a NES, i.e. INPRO requirements may differ from licensing requirements or IAEA Safety Standards. Mr Korinny stressed the difference between INPRO assessment in the area of safety and the assessment using the IAEA Safety Standards. Safety assessment using the IAEA Safety Standards includes formal safety

analysis and the assessment of all aspects of a practice relevant to protection and safety. The INPRO assessment in the area of safety uses available results of safety analysis and it does not include analysis itself. It focuses on sustainability relevant safety issues, with tougher requirements compared with the IAEA safety standards. In his slides, Mr Korinny described safety analysis using the IAEA Safety Standards and INPRO assessment as two consecutive steps. The safety analysis is a necessary prerequisite for INPRO assessment in the area of safety, i.e. INPRO assessor needs results of analysis as input to perform judgements whether INPRO criteria are met.

The INPRO assessment comprises two basic steps:

- Step 1: Collection of necessary input data
  This is a challenging task, and normally takes more than 90% of total effort. The input data includes both for NES to be assessed and a Reference Design (defined as newest reactor operating end of 2004):
  - Information on NPP components and system design (e.g., thermal, mechanical, electrical, neutronics, I&C, etc.)
  - Information from safety analysis (deterministic and probabilistic) of all initiating events, e.g., AOO, DBA and BDBA frequencies and consequences
  - Information on R&D results for innovative features of a NPP
- Step 2: Check whether an INPRO Criterion (CR) is fulfilled.
  This step involves making judgement on the fulfilment of the criteria.
  - CR is met if Indicator (IN) meets its Acceptance Limit (AL)
  - If CR is not met, follow-up action(s) to meet the CR are to be defined

Mr Korinny then described the structure of the INPRO area of safety, which comprises:

- 4 Basic Principles (BPs)
- 14 User Requirements (URs)
- 38 Criteria (CR), some of which are further split into Evaluation Parameters (EPs) – which gives a total of 55 CR and EPs.

The structures for the Basic Principles BP1-BP4 are provided, as shown below, along with structures of some URs for better understanding how the INPRO Methodology works in the area of safety.

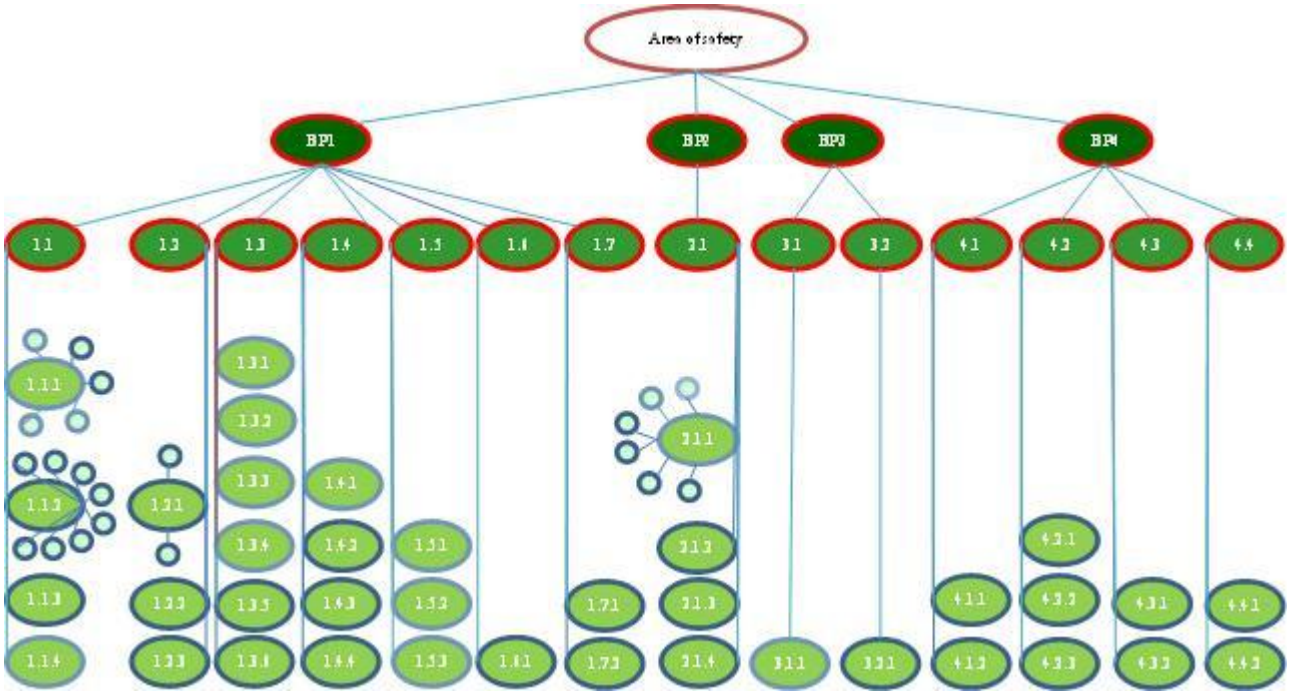


FIGURE 3.3.2. Structure of the INPRO area of Safety

Mr Korinny explained BP1 with 7 URs, in which the first 5 URs correspond to five levels of DID.

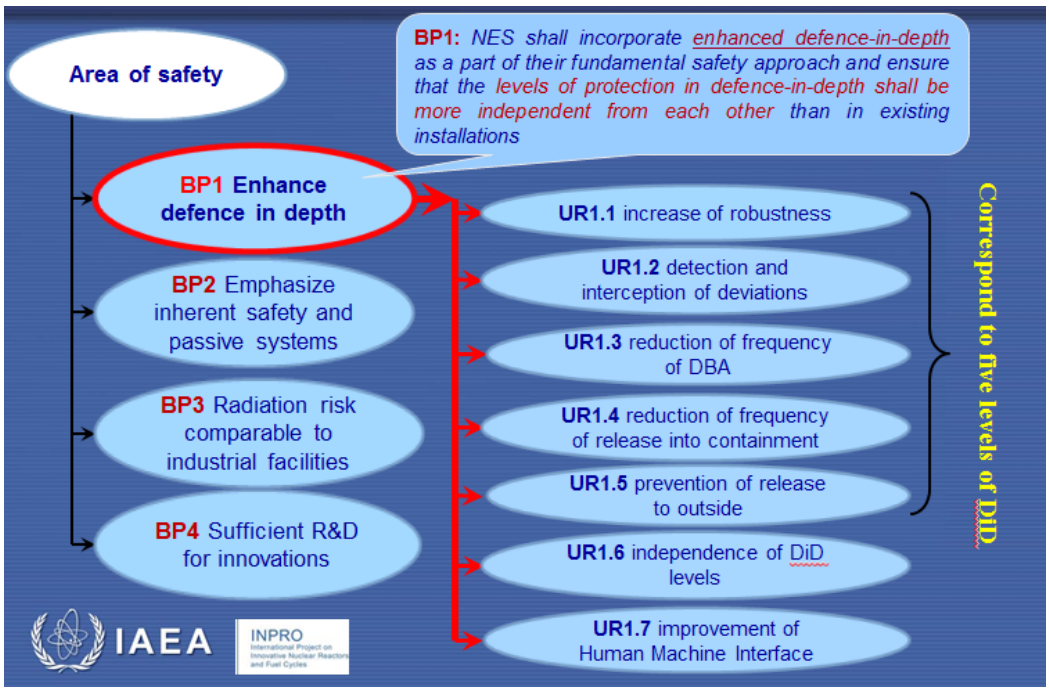


FIGURE 3.3.3. Structure of the first Basic Principles BP1.

For every level of DID, INPRO Methodology proposes certain improvements as put forward by national experts who were developers and users of nuclear technology, i.e. an international consensus. See below. For level 5, the meaning of 'avoid need for evaluation or relocation, Mr Korinny further explained that this does not mean emergency preparedness or planning is not needed, but that the design should demonstrate that something is done to achieve this goal.

IAEA INSAG-10. Objectives for DID	INPRO. Innovation directions
Level 1. Prevention of abnormal operation	Increase inherent safety, reduce necessary human actions/intervention*.
Level 2. Detection and control of abnormal operation.	Advanced control and monitoring systems.
Level 3. Control of accidents within the design basis.	Combination of active and passive features; minimize needed human intervention*.
Level 4. Control of severe plant conditions, including prevention and mitigation of the accident consequences.	Enhanced systems to control and monitor; decrease frequency of severe plant conditions.
Level 5. Mitigation of consequences of significant releases of radioactive materials.	Avoid need for evacuation or relocation.

Mr Korinny then proceeded to discuss UR1.1 'Increase of robustness', CR1.1.1 which covers margins of design, simplicity of design, quality of manufacture and construction, quality of materials and redundancy of systems. Some of the requirements may seem controversial, e.g. simplicity and redundancy of systems, however, they are already well explained in the methodology. In principle, new design should be superior to existing design.

On UR1.4, assuming large increase of NPP in the 21st century, INPRO recommends that the calculated frequency of major release is one magnitude lower than the value currently recommended by the IAEA. So this demonstrates the basic approach of INPRO Methodology to the frequency and consequences of the accidents. INPRO asks that all groups of accidents should have lower frequencies. In case of severe accidents which may result in relocation/evacuation of the population, the design needs to demonstrate that such relocation should be prevented.

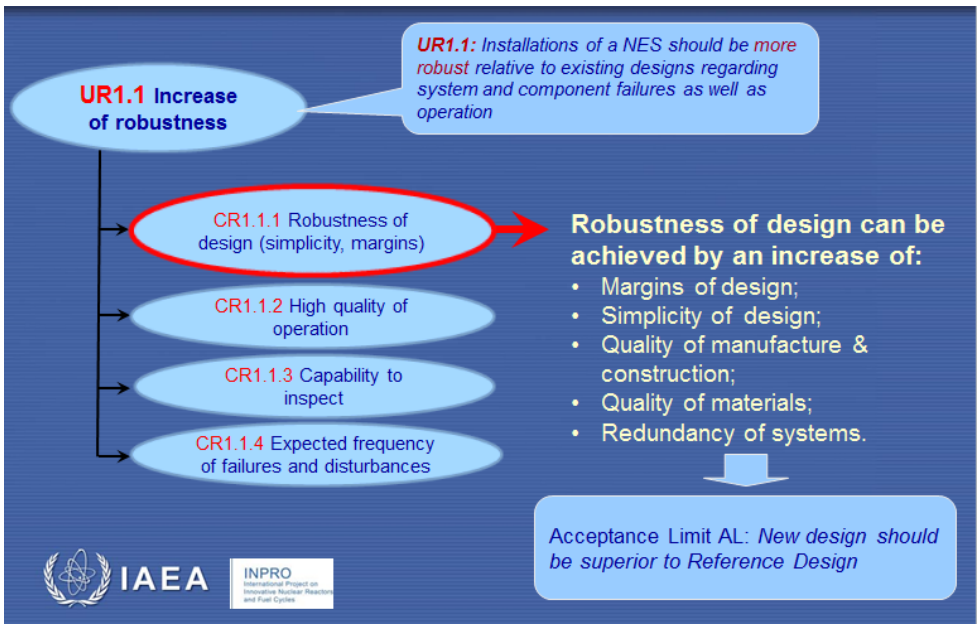


FIGURE 3.3.4. Structure of User Requirement UR1.1.

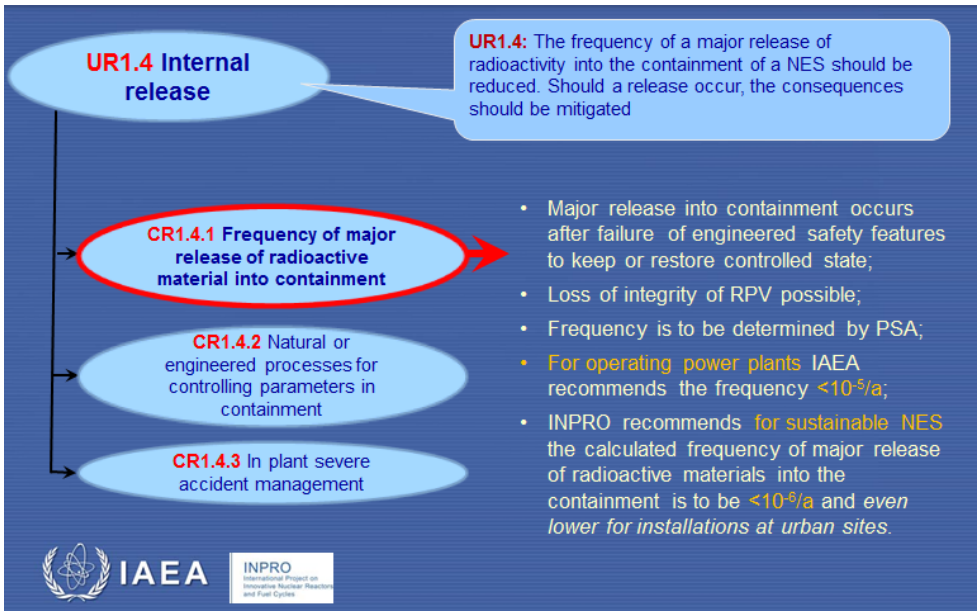


FIGURE 3.3.5. Structure of User Requirement UR1.4.

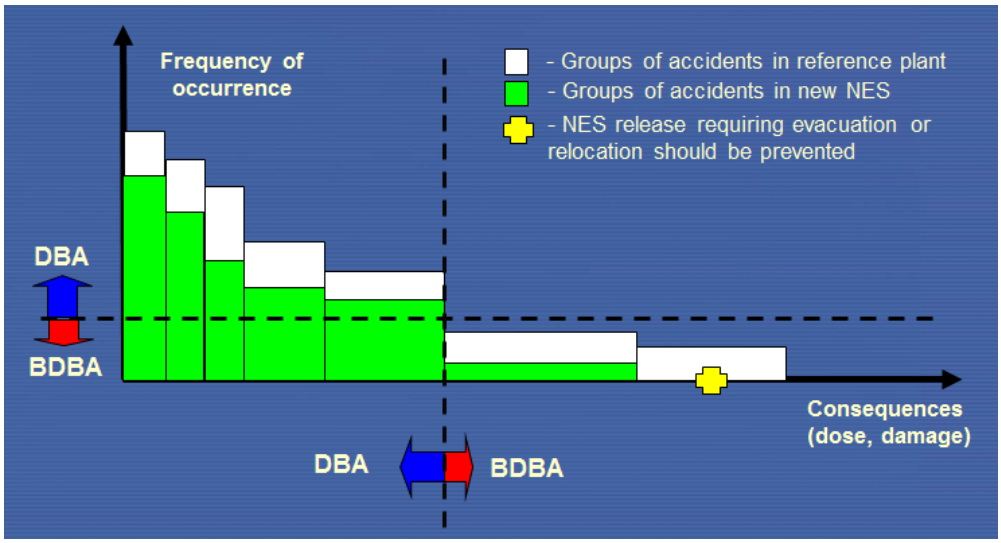


FIGURE 3.3.6. INPRO Requirements to Frequencies and Consequence of Accidents.

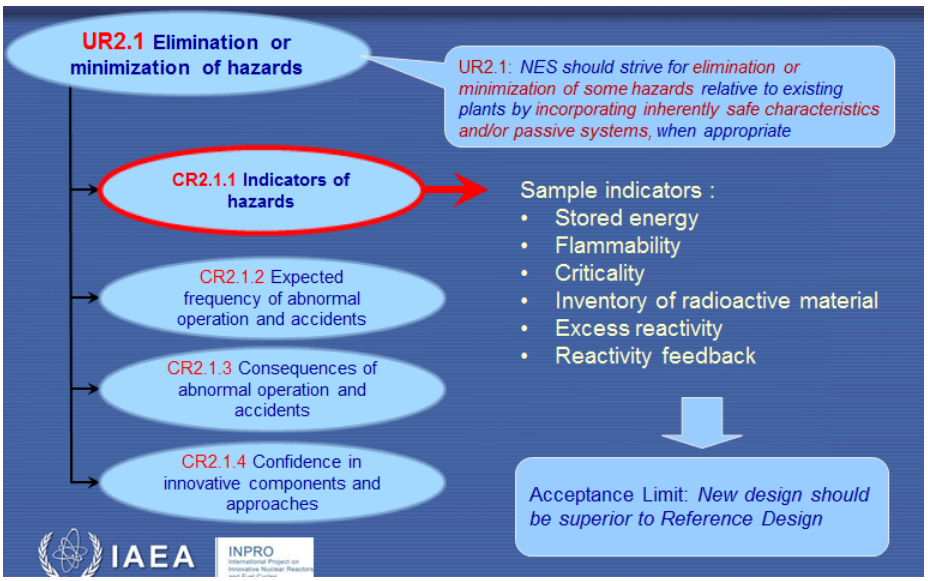


FIGURE 3.3.7. Structure of User Requirement UR2.1 in the second Basic Principle BP2.

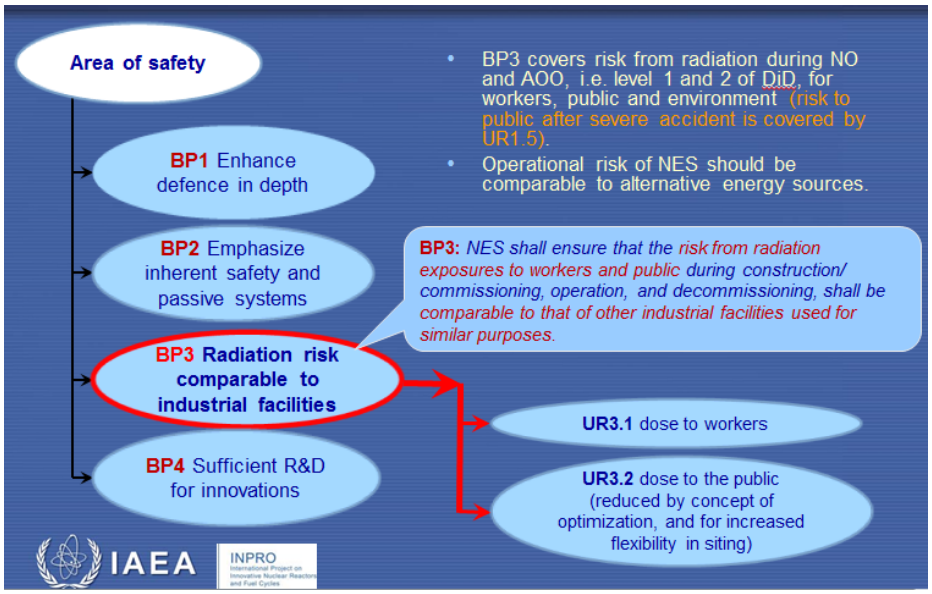


FIGURE 3.3.8. Structure of the third Basic Principle BP3.

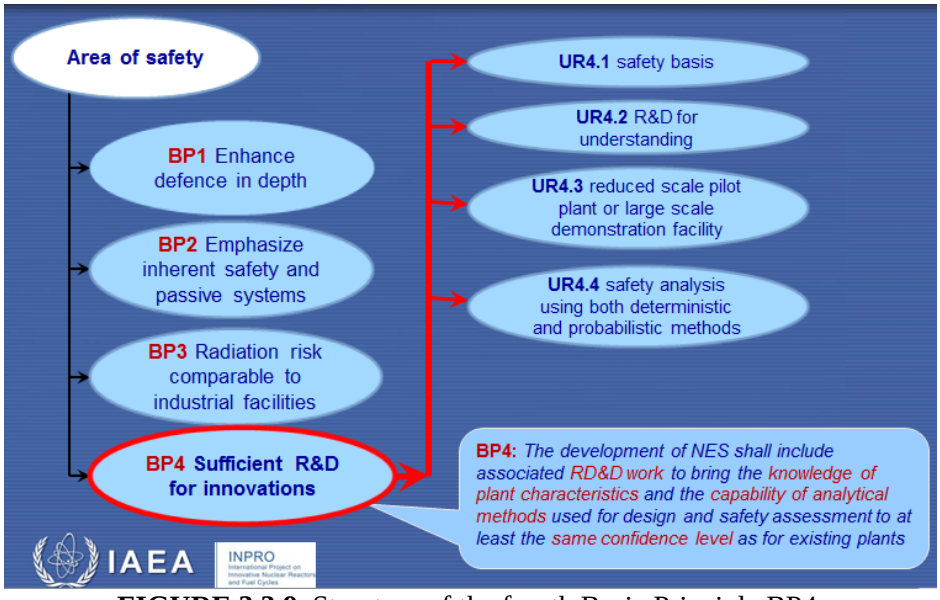


FIGURE 3.3.9. Structure of the fourth Basic Principle BP4.

BP2 asks to emphasize on inherent safety and passive system, has only one UR with 4 CR. For CR2.1.1

INPRO Methodology asks that new design should be superior to existing design. Other criteria asks to demonstrate influence of passive safety features frequencies of normal operation and accidents and the consequences of the accident, and to demonstrate in the design documentation that there is enough information to be confident in the innovative competence and the approach

BP3 discusses risk from NPP, and that risk from radiation exposure to workers and public should be comparable to that from other industrial facilities that are used for similar purposes. Thus, this is the target that the radiation from nuclear should be of the same order/level with the risk from conventional fossil power plants. BP3 has two URs on doses to workers and doses to population.

BP4 asks for demonstration of sufficient R&D for innovations. It has 4 URs which ask to demonstrate safety basis and safety analysis, sufficient R&D for understanding of novel features in the design, and also discussion / justification for necessity or avoidance of construction of large scale demonstration facility of power plant.

In his summary, Mr Korinny reiterated several points to ensure better understanding of lectures to be presented by vendors. Safety of nuclear reactors is only one of the 8 major INPRO areas. INPRO Methodology is a holistic/comprehensive system approach, which covers not only safety of reactor but also 7 other areas. INPRO method of assessment in safety is based not on the comparison of different technologies but on the evolution of progress within the given technology. To make a judgement on sustainability of reactor system, the assessor should perform a full scope assessment comprising all the 38 CR. INPRO Methodology assessment in the safety area only covers the design related safety issues, whereas managerial, regulatory and safety cultures are discussed in the INPRO area of Infrastructure. Other issue like security is covered in the INPRO area of Physical Protection. In other words, one should check other INPRO areas for completeness of the requirements. To increase the safety level of new reactors, INPRO asks the designer to:

- Enhance the concept of defence in depth in new reactor design;
- Include inherent safety characteristics and passive safety systems into the design to minimize hazards;
- Lower the health risk of new reactors to a level comparable with non-nuclear facilities;
- Perform sufficient R&D for all innovative design features of new reactors.

INPRO acknowledges the high safety standard of modern operating NPPs, and that to maintain these standards with a large increase in the number of NPPs will require an increased safety level in the new reactors. INPRO requirements in the area of safety are mainly recommendations for NES developers.

The discussion session is summarized below:

Questions	Answers
Mr Grudev (Bulgaria) asked for the value of the radioactive releases, whether this is treated as different level of radioactive release, and also in the previous methodology. The value is for accident conditions like primary and secondary leak is very small in case of BDBA release that bypass the primary containment. In some NPP like Westinghouse technology allows such release, or others in which the release does not require evacuation. It depends on countries, some of which may already have values while others do not.	**Mr Korinny (IAEA)** responded that for value of radioactive release from primary and secondary coolant, INPRO has a separate area, i.e. Environment which deals with releases into environment for normal operations and anticipated occurrences. INPRO Methodology in the area of safety only deals release from Level 3, DBA and severe accidents. The INPRO requirement specifies restriction on doses, not releases, for population. Theoretically they can be converted into releases. In the case of release, INPRO requires that doses should be so low that relocation is not needed. This dose when relocation becomes necessary is defined in the national regulatory documents, which means INPRO requirements refer to your national documents.
Mr Grudev (Bulgaria) continued asking whether this means it does not allow any release / any type of release, e.g. bypass containment - small leakage from containment during accident.	**Mr Korinny (IAEA)** indicated that this question is related to the INPRO area of environment and not to the area of Safety. Any release must comply to the national requirements. If the country allows certain amount of release, then INPRO requirements refers to this and assumes that release is possible up to that level.

4. SUSTAINABILITY ASSESSMENT OF SAFETY OF EVOLUTIONARY REACTORS

A total of seven evolutionary nuclear reactor technologies were presented in the Dialogue Forum with sustainability assessments of safety aspects for selected criteria using the INPRO Methodology:

1. APWR
2. AES-2006
3. ABWR
4. APR1400
5. Enhanced Candu-6
6. ESBWR
7. EPR

The following table provides the summary of their selected INPRO assessment criteria.

TABLE 4.1. Selection of INPRO Criteria in the Area of Safety of Nuclear Reactors

CRITERIA IN SAFETY	**APWR Mitsubishi**	**AES-2006 Russian Fed**	**ABWR GE Hitachi NE Intl.**	**APR1400 Korea**	**EBSWR GE Hitachi NE Intl.**	**EC6 Candu Energy Inc.**	**EPR AREVA**
CR 1. 1. 1 Robustness (5 EPs)	**EP1.1.1.1 EP1.1.1.2 EP1.1.1.3**	**EP1.1.1.1 EP1.1.1.2 EP1.1.1.3**	**EP1.1.1.3**		**EP1.1.1.1 EP1.1.1.2 EP1.1.1.5**	**EP1.1.1.1 EP1.1.1.5**	**EP1.1.1.1-5**
CR 1. 1. 2 Operation (8 EPs)	**EP1.1.2.2 EP1.1.2.3**		**EP1.1.2.8**		**EP1.1.2.2**	**EP1.1.2.7**	
CR 1. 1. 3 Inspection							
CR 1. 1. 4 Failures and disturbances	√						
CR 1. 2. 1 I&C and Inherent characteristic (2 EPs)	**EP1.2.1.1 EP1.2.1.2**	√		√			
CR 1. 2. 2 Grace Period		√	√		√		
CR 1. 2. 3 Inertia		√	√		√		
CR 1. 3. 1 Frequency of DBA	√		√		√		
CR 1. 3. 2 Grace period		√	√		√		
CR 1. 3. 3 Safety features	√	√	√	√	√	√	
CR 1. 3. 4 Barriers		√				√	
CR 1. 3. 5 Controlled state		√				√	
CR 1. 3. 6 Sub criticality	√	√					
CR 1. 4. 1 Frequency of release into containment	√	√	√		√	√	√
CR 1. 4. 2 Processes	√	√	√		√		√
CR 1. 4. 3 Accident management		√		√	√		√
CR 1. 5. 1 Frequency of release to environment	√				√	√	
CR 1. 5. 2 Consequences							
CR 1. 5. 3 Risk							
CR 1. 6. 1 Independence of DID levels		√	√		√	√	
CR 1. 7. 1 Human factors	√						
CR 1. 7. 2 Human response model							
CR 2. 1. 1 Hazards (6 EPs)							
CR 2. 1. 2 Frequency of AOO & DBA							
CR 2. 1. 3 Consequences							
CR 2. 1. 4 Confidence in innovation			√		√		
CR 3. 1. 1 Occupational dose			√				√
CR 3. 2. 1 Public dose							
CR 4. 1. 1 Safety concept							
CR 4. 1. 2 Safety issues							
CR 4. 2. 1 RD&D				√			
CR 4. 2. 2 Computer codes				√			
CR 4. 2. 3 Scaling				√			
CR 4. 3. 1 Novelty							
CR 4. 3. 2 Pilot facility							
CR 4. 4. 1 Risk informed approach					√		
CR 4. 4. 2 Uncertainties							

4.1. Advanced Pressurized Water Reactor (APWR)

Mr Sumio Fujii and Mr Tetsushi Yuasa from the Mitsubishi Heavy Industries (MHI) (Japan) presented INPRO assessment in the area of safety of the Advanced Pressurized Water Reactor (APWR). They selected

the US-APWR design with the consideration of availability of references in the US NRC website for potential NESA assessors. A special report was provided to the forum containing technical aspects to supplement the assessment. Mr Fujii made a short remark that the data and information provided in the meeting are not permanent and may be changed in the process of nuclear power plant construction for individual site.

4.1.1. Overview of APWR Design

Mr Fujii discussed the following aspects of APWR design:

- Development History of Mitsubishi PWR

  The development of APWR started in the late 1980s driven by economy production cost because the safety of PWR was sufficiently confirmed and production cost was the major concern by the big electric companies. Mitsubishi has designed and constructed 2-Loop (300-600 MWe), 3-Loop (900-1000 MWe) and 4-Loop (1200-1500 MWe) PWRs. In Japan, there have been a total of 24 units, all with good operational history. For the latest conventional PWR, the reactor power was scaled up by increasing the number of SGs. Since it was hard to install more SGs the new system design was required.

  The development of APWR was started in late 1980s as a joint cooperative development project by five Japanese PWR owner utilities and MHI, financially supported by the Ministry of International Trade and Industry (Currently the Ministry of Economy, Trade, and Industry) as a part of the Phase III Improvement and Standardization Program of Japanese PWR. The design of the APWR was based on MHI's conventional 4-loop plant technologies, on which MHI has accumulated significant operating experiences, and was scaled up to achieve higher electrical output. In addition to adopting those proven technologies after the first step development, further modifications were also made on the prototype APWR design to improve economy, safety, reliability, operability, and maintainability by incorporating advanced technologies.

  The first APWR (Japanese APWR or J-APWR) is 1,538 MWe Tsuruga-3, which was under safety review for construction permission then suspended due to the Fukushima Daiichi accident, and the license is stopped now. The development of Tsuruga-3 was completed around 2000.

  By referring the J-APWR, MHI has designed the US-APWR which meets regulatory requirements in United States of America as well as Utilities Requirements Document (URD). The turbine-generator system and all electric equipment of the US-APWR are designed for 60 Hz electric grid. Basic designs of the US-APWR except for the fuel length are same as those of the J-APWR whose design has completed. The US-APWR has been developed as a larger-output version of the J-APWR, aiming at higher electrical outputs and improved economics, by modifying some design features mainly in the secondary side without increasing core thermal output. The fuel length was changed to 4.2 m from 3.66 m, and the electric output was increased to about 1700 MWt.

  The EU-APWR is a sister plant of the US-APWR and is aiming to satisfy European Utilities Requirements (EUR). Nevertheless designs for the turbine-generator system and all of electric equipment are changed to 50 Hz. The designs of core and other major equipment on the reactor coolant system are same as those of the US-APWR except the reactor coolant pumps. So, the difference is just the grid frequency, in addition stricter requirements on severe core damage to be expected in the European countries. Hence, improvements are added to the design.

- Advanced PWR for Global Deployment

  Mr Fujii pointed out that the discussion would focus on US-APWR. The major objectives are to achieve the best performance in safety, reliability, operability, and maintainability by incorporating advanced technologies; and to support large electric grid utilities aiming at economical production costs. For that purpose, some modifications have been introduced to satisfy American and European Utilities Requirements. For the safety and reliability, there are four major features:

  - Top mounted ICIS (In Core Instrumentation System)

    In the conventional PWR, the detector of ICIS was inserted from the bottom, while in the US&EU-APWR the ICIS is inserted from the top hence eliminating penetrations at the bottom of reactor vessel. This design contributes to reduction of potential leakage from the reactor coolant pressure boundary.

- Full 4-train safety system

  Both mechanical and electrical components of the safety system have whole-time redundancy, which not only makes the reliability high but also enables to perform on-line maintenance, i.e. maintenance of safety system during power operation. Also the safety system has a feature of best-mix of passive and active systems.

- Full digital I&C technology enabling one-man operation

  APWR is equipped with full digital I&C system. The full digital I&C realized the ideal advanced operation console in the control room. The monitoring and operation of reactor parameters are put there, enabling one-man operation from start-up to full power.

- Manufacture and Engineering

The US-APWR as the world's largest reactor has an output of 1,700 MWe (4,466 MWt) to provide the economy advantage. The output is realized through high quality steam from the high performance SG and high performance LP-turbine having the last stage blades of 74 inches in length, which yields a thermal efficiency of more than 36%. The enhanced SG heat transfer performance is achieved from the enlarged heat transfer area with triangular lattice arrangement of SG tubes. In addition, the 14-ft fuels create additional thermal margin and flexible core operation, and also allow 24-month operation without deterioration in fuel economy.

Mr Fujii also presented a comparison of major specifications among the conventional 4-loop PWR, J-APWR and US & EU-PWR. He then noted the followings briefly:

- Comparison of the reactor vessels: The US-APWR does not have nozzles at the bottom of reactor vessel for the ICIS. In addition, the neutron reflector is made of SS thus more neutron economy. The 257 fuel assemblies make the geometry of core long and higher neutron economy.
- Major thermal-hydraulic parameters: The linear power density is less than 90 kW/l, and the specific power is only 32 kW/kg U, i.e. 10% smaller than that of the conventional PWR. So US-APWR has more operational margin from the point of view of fuel rod cooling.

TABLE 4.1.1. Comparison of Major Specifications among Conventional PWR, J-APWR and US&EU-PWR

Design Parameter	Conventional 4-loop PWR	Japanese APWR	US & EU-APWR
Electric Output	1,180 MWe class	1,538 MWe	1,700 MWe class
Core Thermal Output	3,411 MWt	4,451 MWt	4,451 MWt
Reactor Thermal Output	3,423 MWt	4,466 MWt	4,466 MWt
Core	3.66 m active height 193 Assemblies	3.66 m active height 257 Assemblies	4.2 m active height 257 Assemblies
SG Heat Transfer Area	4,900 m^2/SG	6,500 m^2/SG	8,500 m^2/SG
Thermal Design Flow Rate per Loop	20,100 m^3/h	25,800 m^3/h	25,400 m^3/h
Length of Turbine Last Stage Blades	44 inches	54 inches	74 inches
Containment Vessel	PCCV	PCCV	PCCV
Safety Systems	Electrical : 2 trains Mechanical : 2 trains	Electrical : 2 trains Mechanical : 4 trains	Electrical : 4 trains Mechanical : 4 trains
	HHSI × 2 Accumulator × 4 LHSI×2	HHSI × 4 Advanced Accumulator × 4 (No LHSI)	HHSI × 4 Advanced Accumulator × 4 (No LHSI)
I & C System	Safety System : Analog Solid State Control System : Digital	Full Digital	Full Digital

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- Safety of Mitsubishi Advanced PWR

  Mr Fujii discussed the structure of ECCS and highlighted some features and strengths:

- Four independent safety trains: four-train mechanical safety system, four-train electrical safety system, and no inter-connecting piping between trains. In comparison, there are only 2 subsystems in the conventional PWR.
- Direct Injection (DVI) which enables efficient injection of cooling water directly into the reactor vessel.
- Refueling water storage tank (RWST) as the water source for ECCS located inside the containment, which eliminates the switchover operation of water source.
- Advanced accumulator (ACC) which is connected to each cold leg to refill the reactor vessel lower plenum and down-comer immediately after a LOCA. The ACC can then supply reduced flow rate to provide a longer supplying duration that can cover the time expected to the low pressure injection system
- Elimination of the low head safety injection system, as a result of the use of ACC, and this leads to the reduction of active components.

- State-of-Art Technology for Digital Instrumentation & Control System
  - Full Digital I&C System
    - Fully digitalized technology is applied not only to control system but also safety protection system. In the past, according to the reliability of computer system and seismic acceleration problem, the application of control system is limited to control system. The safety system was accomplished by solid state protection system. However, after Tomari-3, fully I&C is used for both control and safety system.
    - Conventional operating and monitoring devices such as switches and indicators have been eliminated, except devices for diverse actuation system (DAS).
    - The reactor protection system and actuation system of engineered safeguard features (ESF) have 4-time redundancy in the I&C system. Software has been checked in very details for verification and validation. It is in the opinion that common-cause failure will not occur, however, since in reality there is a very low possibility in which all functions of computer system may be lost. DAS was designed for beyond design basis CCF incidents. DAS hardware provides back-up actuations to operate all safety systems, so that reactor can be shut down and decay heat be removed. The DAS initiates safety functions independent from the output of the digital safety system. Manual actuation is provided for all functions, but automatic actuation is also provided for functions if the time for manual operator action is inadequate.

System	Features
Main control board	・Fully computerized ・Visual Display Units (VDU) for safety and non-safety channels ・Limited number of conventional switches and indicators
Safety I&C	・Fully digitalized with Mitsubishi digital controllers ・Four-train redundancy in reactor trip system, ESF actuation system and safety logic system for component control
Non-safety I&C	・Fully digitalized with Mitsubishi digital controllers ・Duplex digital architecture for each control and process monitoring sub-system
Data communication	・Fully multiplexed, including class 1E signals ・Multi-drop data bus and serial data link ・Fiber optics communication networks for noise immunity and required isolation

  - Human Engineering and Control Room Design
    - Monitoring of plant parameters and equipment status is done on computer screen. The operation is also carried out on the same computer screen with touch screen operation. The operation of the plant condition is summed up and the operator could easily grasp it. Thus, the operator's work load is reduced and the reliability of operation is increased.
    - A large display panel is installed in the backside of the control room to show major parameters for normal and abnormal situations. The current status of the entire plant can be understood by all crews (supervisors, operators, technical advisers) and their communication is improved.

Thus, the digital system provides significant benefits to the safety of nuclear power through reduction in operation works and maintenance work loads, which reduces the potential for human error.

- *Safety Enhancements for Beyond Design Basis Accident*

All potential incidents have been considered in the design by taking into account the probability, for both internal events (e.g. malfunction of components, break of pipe, rupture of structure, human error) and external events i.e. natural hazards (e.g. seismic, tsunami, hurricane) and non-natural hazards (e.g. airplane crash, terrorist attack, gas explosion, transmission interruption). Some events have very low probability and are beyond DBA, but nevertheless MHI prepares countermeasures against BDBA and Design Extension Conditions.

Mr Fujii provided examples:

- Countermeasures against airplane crash taking into account the potential effects of the impact of a large commercial airplane for the building strengths and plant layout by means of: (i) physical/functional separation and segmentation in the layout design for safety related structures, systems and components (SSCs), and (ii) reinforcement of structural design of the buildings, which contain safety related SSCs. It was confirmed by assessment that the inherent robustness would be maintained for the followings: (i) reactor core remains cooled or the containment remains intact, and (ii) spent fuel pit integrity is maintained.
- Countermeasures against severe accidents

  Preventive measures:

  - Reduction of potential precursor of design basis accidents and severe accidents
    For example reduction of latent LOCA precursor by (i) adopting top-mounted ICIS to eliminate penetrations at the reactor vessel bottom, (ii) reduction of reactor vessel welding seams and improvement of welding methodologies, and (iii) improvement of piping bypassing the containment to a higher rating to reduce probability of interface LOCA
  - Enhancement of reliability in safety functions.
    - Installation of four-train structure for safety systems
    - Installation of refueling water storage pit (RWSP) in the containment
    - Installation of ACC to reduce the number of active components
    - Countermeasures against Anticipated Transient Without Scram (ATWS). For example, installation of diverse actuation system to cope with common cause failure in the reactor protection system.
    - Countermeasures against station blackout (SBO). For examples, installation of four-train emergency gas turbine generators (GTGs) and additional alternate power source by two-train GTGs.
    - Countermeasures against fire. For example, clear physical separation between the redundant trains of safety systems.
    - Countermeasures against intersystem LOCA. Prevention of over-pressurization of the residual heat removal system.

  As a result, the frequency of severe accidents (PRA result for core damage frequency (CDF) for US-APWR meets the NRC goal and the Utility Requirements Document (URD) goal.

Requirement	NRC	URD	US-APWR
CDF	1×10^{-4}/RY	1×10^{-5}/RY	3.0×10^{-6}/RY

Mitigation measures to maintain the integrity of the containment vessel:
- For debris dispersion, reinforcement of depressurization function of the primary system and an improvement of RV cavity geometry are considered.
- For quasi-static over pressurization, a usual containment vessel air recirculation system and an alternative containment spray supplied from the fire service water system can be used to reduce the pressure if the containment spray system is not available.

- A hydrogen control system (igniters) is installed to control the hydrogen concentration in order to avoid containment vessel damage due to hydrogen combustion.

- *Construction Management**

  Total Engineering Capability with "Single Responsibility" in the full areas:
  - Conceptual/Basic/Detailed Engineering
  - Manufacturing of components
  - Erections and installations of structures, systems and components
  - Support of human resource development for operation & maintenance
  - Technical support after start of commercial operation

  To keep quality and schedule, MHI' Data management system integrates (i) Design, (ii) Procurement, (iii) Manufacturing & Inspection, and (iv) Construction. Major Components (Reactor Vessel, Steam Generator, Reactor Coolant Pump, Reactor Internals, Control Rod Drive Mechanism, Pressurizer, Turbine, etc.) are manufactured by MHI. Works and machines have been updated/enlarged and are prepared for the global deployment.

  Reduction of on-site work volume and construction period is done through:

  - Rational designs, e.g. internal structures using steel plate reinforced concrete (SC), and large modular (prefabricated) block construction
  - Tools for efficient construction like Super-large-capacity cranes which enables on-site welding and formation of containment vessel.
  - Ability for comprehensive coordination of civil & installation work

  From the 1st Concrete to Fuel Loading, the time required is: 34.5 months (2 loop), 37.5 months (3 loop) and 40.0 months (4 loop).

- *Operation and Maintenance Management**

  - Reactor vessel inspection unit A-UT machine
    Inspections can be done for 4 days by utilizing two units simultaneously (for 4 loops). It is an application of world's first submerged automated guided vehicle with robot, using a manipulator with 7 shafts.
  - RV head nozzle inspection unit, with ECT probe for 3-type welds surface and phased array type probe for 3-groove welds
  - SG tubes inspection unit
    One unit (more than 3,000 tubes) can be inspected for ~4 days.
  - UT/ECT for BMI nozzle inspection unit
  - Piping inspection unit, with MHI point focus probe (high accuracy sizing with point focus capability) and automated UT unit for piping (lightweight, compact, point focus phased array enabling high accuracy flaw sizing).

In conclusion, Mr Fujii made the following summary:

- MHI developed Advanced PWRs based on accumulation of half-century experiences in construction and operation of conventional PWR power plants.
- Proven and advanced technologies have been introduced in APWR to improve economy, safety, reliability, operability and maintainability.
- Advanced accumulator was employed to reduce the number of active components, and low pressure injection system was eliminated.
- US- and EU-APWR were developed to meet regulator's and user's requirements including flexible operation. The fuel length was changed to 4.2 m.
- ECCS and supporting systems have 4 divisions in redundancy.
- Due to enhancements in safety systems, the total of the CDF is estimated 3.0×10^{-6}/RY in a PRA calculation.
- MHI's manufacturing and construction management is integrated to one data base system to accomplish them with high quality, high reliability and strict keeping of schedule*.

* *The materials were covered in the slides but not presented due to time constraints*

4.1.2. INPRO assessment in the area of safety of APWR

Mitsubishi selected 6 design areas corresponding to the INPRO URs that are considered very important to the safety and reliability and represent the strengths of superiority of the US-APWR design to the conventional PWR design from the viewpoint of the sustainability of nuclear energy/ INPRO objective. Since each INPRO's user requirement is related to various design areas, the INPRO assessments are summarized into 6 presentations by design areas as follow.

- Operational safety margins in design of APWR
- Reactor vessel manufacturing and welding of APWR
- Simplicity and Reliability in Safety system design
- Reliability data of equipment and CDF
- Detection of Anomalous condition and safety analysis
- Integrated full digital I&C system and advanced operator console

4.1.2.1. Operational safety margins in design of APWR

4.1.2.1.1. EP1.1.1.1 Margins of Design

Mr Fujii (Mitsubishi) discussed the following margins of design:

a. Reactor power and coolant temperature

The operation pressure of reactor coolant is always kept at 15.41 MPa (gauge), hence the subcooling is kept at more than 18 °C anywhere in the primary coolant system for both designs. However, the coolant temperature changes with power reactor, i.e. the coolant temperature is controlled to follow the programmed target value by Reactor Control System. The coolant temperature between the conventional PWR and US-APWR is not much of a difference; the US-APWR has a slightly lower value, i.e. a slightly larger margin.

b. Reactivity coefficient

- Doppler power coefficients
  According to neutronic theory, the value of Doppler power coefficient is dominated by neutron energy spectrum, which is mainly dominated by the water-fuel ratio and composition of fissile in the core. The water-fuel ratio is fixed by the geometry of fuel assembly, meaning that all PWRs have similar Doppler power coefficients as long as they use the same fissile composition in the core. So, Doppler coefficients are similar for conventional PWR and US-APWR.

- Moderator density coefficient.
  The moderator density coefficient in the subcooling condition is mainly dominated by boron concentration in the reactor coolant and also slightly by temperature and pressure of reactor coolant. The bounding values of moderator density coefficient used in safety analysis are 0.0 (Δk/k)/(g/cc) and 0.51(Δk/k)/(g/cc), which are the same for both conventional PWR and US-APWR. This is because the operational length of boron and temperature and pressure of reactor coolant are similar in both designs. The moderator density coefficients don't become negative during hot condition for both designs. The maximum design value of initial core is 0.32(Δk/k)/(g/cc) for the US-APWR. So, there is not large difference in the value of moderator density coefficient between conventional PWR and US-APWR.

c. Reactivity shutdown margin

The required shutdown margin is 1.6% Δk/k for the conventional PWR and US-APWR. The shutdown margins of the US-APWR are 2.94 and 2.15% Δk/k at BOC (beginning of cycle) and EOC (end of cycle) of the initial core respectively. On the other hand, the conventional PWR has a shutdown margin of 1.99% Δk/k at EOC. So, the US-APWR has a larger shutdown margin than the conventional PWR.

d. Forced reactor coolant flow

The thermal design flow rate of the reactor coolant for the US-APWR is 76,300 ton/h. Nearly 9% of the reactor coolant flows outside the core to cool reactor vessel and other structures. In consequence, 69, 400 ton/h of reactor coolant cools the fuel assemblies and structures generating 4,451 MWt in the core. In the conventional PWR, the thermal design flow rate of the reactor coolant is 60,100 ton/h. Around 6.5% of the reactor coolant flows outside the core to cool reactor vessel and other structures. In consequence, 56,000 ton/h of reactor coolant cools the fuel assemblies and structures generating 3,411 MWt in the core. The coolant flow rate per reactor power in the design of the US-APWR is 17.14 ton/h/MW, which

is slightly smaller than 17.62 ton/h/MW of the conventional PWR. However, the difference is considered very small.

e. Natural circulation of reactor coolant

When the reactor coolant pumps trip, the reactor coolant will flow due to density distribution and gravity. The height of reactor vessel of the US-APWR is 0.7 m longer than that of the conventional PWR. As a result, steam generators (SG) in the US-APWR are located higher, and the elevation difference gives a larger driving force to the reactor coolant for natural circulation caused by density distribution. The head of the reactor coolant pump (RCP) of the US-APWR is about 93.5 m at the thermal design flow of 25,400 m^3/h, whereas the pump head of the conventional PWR is about 80.0 m at the thermal design flow of 20,100 m^3/h. This gives a friction loss coefficient of $93.5/25400^2=0.00145$ for the US-APWR, and $80.0/20100^2=0.00198$ for the conventional PWR. Hence, the US-APWR has less friction coefficient against the natural circulation. So the US-APWR has larger natural circulation characteristic than the conventional PWR. It was noted that a natural circulation flow rate of the reactor coolant was measured in an existing PWR in Japan. When the heat generation of core was about 2% of nominal reactor power, the flow rate was about 4%.

f. Linear heat generation rate

The average linear heat generation rates of the US-APWR and conventional PWR are 15.3 and 17.9 kW/m respectively by taking into account the direct gamma heating to coolant of 2.6 %. Therefore, the US-APWR has a larger operational margin than the conventional PWR because the fuel rod is about 15% longer than that of the conventional PWR. The limiting values of hot channel factors are decided using the same concept for the US-APWR and conventional PWR, e.g. the total heat flux hot channel factor FQ of the US-PWR is 2.60, which was also used for some conventional PWRs. Therefore, the US-APWR has more operational margin at hot spots too.

g. Minimum DNB ratio

The limiting values of minimum DNBR are determined by calculating the bounding values covering 95% probability at 95% confidence level. The limiting value for the US-APWR was calculated by using WRB-2 correlation, while the limiting value for the conventional PWR was calculated by using MIRC-1 correlation. Therefore the correlations are not the same and it is not possible to compare the values directly. However, the design methodology using 95% probability at 95% confidence level is reasonable, thus both designs have the same safety margins when linear heat generation rates get to the limiting values.

h. Operable range permitted by the reactor protection system

Allowed operating space at 15.41 MPa of over power and over temperature for the US-APWR and the conventional PWR shows that the operational range allowed by reactor coolant system is similar for both designs.

i. Fuel temperature and stored energy

The principle design strategy is that the maximum fuel centerline temperature should be less than the fuel melting point so that the fuel cladding will not be mechanically damaged. There is at least a 95% probability at a 95% confidence level that the fuel rod with the most limiting linear heat generation rate does not cause the fuel pellet to melt during normal operation and AOOs. The fuel length of the US-APWR is 15% longer than that of the conventional PWR, i.e. a lower linear heat generation rate. This means the US-APWR has smaller fuel temperature and less stored energy.

The evidences indicating increased margins of the US-APWR design in comparison to existing design are available to the INPRO assessor and have been demonstrated. The superiority of the US-APWR design to the existing design (conventional PWR) in terms of robustness with regard to the Acceptance Limit AL1.1.1 is shown. The compliance composes partially satisfaction of UR1.1.

4.1.2.2. Reactor vessel manufacturing and welding of APWR

Mr Fujii (Mitsubishi) discussed as follows.

4.1.2.2.1. EP1.1.1.3 Quality of manufacture and construction

- *Process of reactor vessel manufacturing*

In the conventional PWR, the upper shell and lower shell were made from 3 pieces of steel plates by bending and welding. The products hence had a long total length of weld seams, which were supposed to be subjected to periodical in-service inspection (ISI) during the plant life. There were also weld seams at beltline where high energy neutron irradiation was predicted. For the US-APWR, each of the upper shell and lower shell is manufactured from one forging. This results in a total length reduction of welding lines, which leads to reduction of the work for quality control during the manufacture and to reduction of the work for the ISI during the plant life. There is no weld seam at beltline where high energy neutron irradiation is predicted.

- *Structure and material of reactor vessel (RV)*

  When manufacturing RV, severe control of raw materials is employed. The raw material for the RV shell is low alloy steel equivalent to SA-508 Gr3 of the ASME II. The requirement of specifications concerning impurities is that when purchasing raw materials, the MHI uses severer requirements in order to produce a RV with higher quality so as to reduce degradation in ductility due to high energy neutron irradiations by considering the reliability for the plant life. Additional requirements like JIS are imposed by MHI to achieve the reliability of the RV for the plant life of 60 years.

Impurities	SA-508 Gr3 of ASME II	Additional requirements
Cu	Less than 0.20 %	Less than 0.10 %
P	Less than 0.025 %	Less than 0.015 %
S	Less than 0.025 %	Less than 0.018 %

- *Welding between different materials*

  The reactor coolant system uses several kinds of alloy steels, which imposes dissimilar welds. Welding processes at the site such as pre-welding forms, welding materials and welding machines have been improved to attain more reliable products. Special treatment is used, e.g. for welding between the cold leg piping made of stainless steel and RV inlet nozzle made of low alloy steel. Welding area and heat become smaller, and welding material is also changed from alloy 600 to alloy 690 to obtain very strong welding seam.

Facts from the assessment are as follows: (i) Reduction of total length of weld seams, (ii) No weld seam at beltline; (iii) Severe quality control of raw materials, and (iv) Improvement in welding process. Evidences indicating higher quality of manufacture and reduction of potential break frequency of the US-APWR design in comparison to existing design are available to the INPRO assessor and have been demonstrated. The superiority of the US-APWR design to existing design in regard to the Acceptance Limit AL1.1.1 is shown.

4.1.2.2.2. CR 1.3.1. Calculated frequency of occurrences of DBAs

In theory the frequency of potential breaks has been improved, e.g. probability of LOCA becomes smaller given the improvement of the process of manufacturing, welding and construction. That is, for probability risk assessment MHI uses this kind of initiating event in the table with reference to NUREG etc. However, MHI still uses the same value for probability risk assessment because it is impossible to show evidence with high reliability without experience / operational history. A period of 25 years is needed to obtain real reliability of welded material. In other words, theoretically it is improved but evidence is necessary to show how much improvement is done. Therefore, the frequency of occurrence of accidents must be small according to UR1.3 but it is impossible to show reduction of the value.

TABLE 4.1.2. Frequency of design basis accidents for US-APWR

	Name	Event Description	Frequency	Reference
1	LLOCA	Large Pipe Break LOCA	1.2E-6	NUREG/CR-6928
2	MLOCA	Medium Pipe Break LOCA	5.0E-4	NUREG/CR-6928
3	SLOCA	Small Pipe Break LOCA	3.6E-3	NUREG/CR-6928
4	VSLOCA	Very Small Pipe Break LOCA	1.5E-3	NUREG/CR-6928
5	SGTR	Steam Generator Tube Rupture	4.0E-3	NUREG/CR-6928
6	RVR	Reactor Vessel Rupture	1.0E-7	WASH-1400
7	SLBO	Steam Line Break/Leak in downstream of Isolation Valve	1.0E-2	NUREG/CR-5750
8	SLBI	Steam Line Break/Leak in upstream of Isolation Valve	1.0E-3	NUREG/CR-5750
9	FWLB	Feed-water Line Break	3.4E-3	NUREG/CR-5750
10	TRANS	General Transient	8.0E-1	NUREG/CR-6928
11	LOFF	Loss of Feed-water Flow	1.9E-1	NUREG/CR-6928
12	LOCCW	Loss of Component Cooling Water	2.4E-5	**Fault Tree Analysis**
13	PLOCW	Partial Loss of Component Cooling Water	3.2E-3	NUREG/CR-6928
14	LOOP	Loss of Offsite Power	4.0E-2	NUREG/CR-6928
15	LOAC	Loss of Vital AC Bus	9.0E-3	NUREG/CR-6928
16	LODC	Loss of Vital DC Bus	1.2E-3	NUREG/CR-6928

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Evidences indicating reduction of potential break frequency given the higher quality of manufacture of the US-APWR design in comparison to existing design are available to the INPRO assessor and have been demonstrated. The superiority of the US-APWR design to existing design in regard to the Acceptance Limit AL1.3.1 is shown. The compliance composes partially satisfaction of UR1.3.

4.1.2.3. Simplicity and Reliability in Safety System Design

4.1.2.3.1. EP1.1.1.2. Simplicity of Design

Mr Yuasa presented several simplicity found in the APWR design as follows:

- *Separation between subsytems in ECCS*

  In the US-APWR, four trains for the mechanical and electrical systems are clearly separated from each other (not tie lines). Meanwhile in the conventional PWR, two trains for the mechanical systems have interconnections, e.g. common lines from RWSP and containment recirculation sump. Tie lines are also required for high head and low head injection lines to supply injection water to every RCS loop.

- *Elimination of Low Pressure Injection System*

  For the conventional PWR, the low head safety injection (LHSI) system is installed and shares the emergency core cooling function with accumulator system and the high head safety injection (HHSI) system. For large break LOCA, the LHSI system also supplies high flow rate of cooling water from early stage, as rapid decreasing of the RCS pressure subsequently after injection from accumulator. On the other hand, in the US-APWR, the advanced accumulator system has flow-changing characteristics by passive flow damper, which enables a longer supplying duration that can cover the LHSI function and eliminate LHSI system.

- *Cooling water source (Refueling Water Storage Tank)*

  In the conventional PWR, the HHSI system has two water sources, i.e. refueling water storage tank located outside containment vessel and the other is recirculation sump at the bottom of the containment vessel. This configuration requires each suction line piping and switching operation during accidents. During LOCA, the first injection starts from the RWST and the tank gets to low water level. Hence, the operation of changing the suction from the RWST to the containment recirculation sumps is required at RWST low water level. Train separation is also required in recirculation mode post LOCA, which is achieved by closing the tie lines valves. These affect the PRA results. The system is also required for the LHSI. Meanwhile, in the US-APWR the refueling water storage pit (RWSP) is installed inside / at the bottom of the containment as a single source of water and collecting injection water during recirculation. This results in the design that eliminates the changing operation. Adoption of 4-train system can also eliminate tie-lines among trains, which contributes to the enhancement of system reliability and also PRA results. Comparison of piping layout shows that the US-APWR has separated and dispersed layout of 4-train system with each suction line connected nearby the recirculation pump, leading to the reduction of pipe length of both the safety system injection and suction lines (safety injection system, containment spray system).

Facts from the assessment are as follows: (i) Clear separation between the trains (elimination of tie-lines), (ii) Elimination of low pressure injection, and (iii) No need to change the cooling water source. Evidences indicating increased simplicity of the US-APWR design in comparison to existing design are available to the INPRO assessor and have been demonstrated. The superiority of the US-APWR design to existing design in regard to the Acceptance Limit AL1.1.1 is shown. The compliance composes partially satisfaction of UR1.1.

4.1.2.4. Reliability Data of Equipment and CDF of APWR

Mr Yuasa (Mitsubishi) discussed five related CR for the subject matter.

4.1.2.4.1. CR1.3.1. Calculated frequency of occurrence of DBAs

Many improvements to reduce the occurrence frequency of DBAs such as LOCA have been introduced in the US-APWR. For examples:

- improvements of reactor vessel material & manufacturing process
- improvement of welding process for the reactor coolant piping
- elimination of the penetrations at the RV bottom by employing the top-mounted in-core instrumentation system

In theory, the improved designs and well-organized quality assurance program can reduce the failure rates of components. Consequently, frequency of occurrence of accidents must become small. In reality, however, it is hard to numerically evaluate the quality of improvement and to estimate improved failure rate before accumulation of operational experiences. For evidence, major initiating events in the US-APWR are shown in the table below. The calculated frequency of accident occurrences used in PRA are still cited from NUREG and WASH-1400, with minor exceptions when the number of latent weaknesses in the SSC can be clearly analysed and failure rate can be evaluated based on the improvement against the latent weaknesses. In other words, the frequency does not reflect the improvement of manufacturing and welding processes. For example, the frequency of reactor vessel rupture is assessed at 1.0×10^{-7}/RY (cited from WASH-1400 in 1974) even though the lower-shell of the RV for the US-APWR is forged without welding seam line from one large ingot fabricated in electric smelter in a vacuum room to minimize impurities. The frequency of RV rupture remains in CDF since there is no way to avoid if it occurs. Note the exception for the frequency of loss of component cooling water which is calculated by fault tree analysis, reflecting the system configuration of the US-APWR.

TABLE 4.1.3. Major initiating events in US-APWR

	Name	Event Description	Frequency	Reference
1	LLOCA	Large Pipe Break LOCA	1.2E-6	NUREG/CR-6928
2	MLOCA	Medium Pipe Break LOCA	5.0E-4	NUREG/CR-6928
3	SLOCA	Small Pipe Break LOCA	3.6E-3	NUREG/CR-6928
4	VSLOCA	Very Small Pipe Break LOCA	1.5E-3	NUREG/CR-6928
5	SGTR	Steam Generator Tube Rupture	4.0E-3	NUREG/CR-6928
6	RVR	Reactor Vessel Rupture	1.0E-7	WASH-1400
7	SLBO	Steam Line Break/Leak in downstream of Isolation Valve	1.0E-2	NUREG/CR-5750
8	SLBI	Steam Line Break/Leak in upstream of Isolation Valve	1.0E-3	NUREG/CR-5750
9	FWLB	Feed-water Line Break	3.4E-3	NUREG/CR-5750
10	TRANS	General Transient	8.0E-1	NUREG/CR-6928
11	LOFF	Loss of Feed-water Flow	1.9E-1	NUREG/CR-6928
12	LOCCW	Loss of Component Cooling Water	2.4E-5	**Fault Tree Analysis**
13	PLOCW	Partial Loss of Component Cooling Water	3.2E-3	NUREG/CR-6928
14	LOOP	Loss of Offsite Power	4.0E-2	NUREG/CR-6928
15	LOAC	Loss of Vital AC Bus	9.0E-3	NUREG/CR-6928
16	LODC	Loss of Vital DC Bus	1.2E-3	NUREG/CR-6928

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4.1.2.4.2. CR1.3.3. Reliability of engineered safety features

Reliability of a system is assessed by using a fault tree with reliability data of composing components. Such set of reliability data is elaborated based on accumulated operational experiences in each country because the quality control management methods for manufacture of components in a factory and preventive

maintenance in a power plant vary in country by country. This indicates a fact that a high pressure injection system with the same design has different malfunction probabilities in the USA and Japan. Therefore, it is impossible to compare the reliabilities of engineered safety features between the US-APWR and the conventional PWR. The calculated failure probabilities of the front line systems and support systems for the US-APWR were provided by MHI. See the examples below.

TABLE 4.1.4. Calculated failure probability of front line system

Fault Tree Name	Fault Tree Description	Fault Tree Probability
HHI System and Pressurizes Pressure Control System		
HPI-FAB	Failure of Bleed and Feed operation	4.1E-03
HPI-FAB-AC	Failure of Bleed and Feed operation (LOAC)	4.1E-03
HPI-FAB-DC	Failure of Bleed and Feed operation (LODC)	4.1E-03
HPI-FAB-LP1	Failure of Bleed and Feed operation (LOOP)	4.2E-03
HPI-FAB-LP2	Failure of Bleed and Feed operation (LOOP)	4.1E-03
Pressurizes Pressure Control System		
PZR-FAB	Failure of Bleed operation (Other Initiating Events)	2.7E-03
PZR-SGT	Failure of Decompress RCS (SGTR)	6.3E-03
Alternate Containment Cooling		
NCC	Failure of Alternate Containment Cooling (Other Initiating Events)	2.9E-02
NCC-AC	Failure of Alternate Containment Cooling (LOAC)	2.9E-02
NCC-DC	Failure of Alternate Containment Cooling (LODC)	3.2E-02
NCC-LL	Failure of Alternate Containment Cooling (LLOCA)	2.9E-02
NCC-LP1	Failure of Alternate Containment Cooling (LOOP)	5.2E-02
NCC-LP2	Failure of Alternate Containment Cooling (LOOP: No breakdown)	2.9E-02
NCC-PC	Failure of Alternate Containment Cooling (PLOCW)	2.9E-02
Secondary Side Cooling (Main Steam Control System and Emergency Feed Water System)		
MSP-LO-LP1	Failure of Secondary Side Cooling (LOOP : Emergency Power Success)	1.6E-02
MSP-LO-LP2	Failure of Secondary Side Cooling (LOOP : Emergency Power Success)	6.8E-03
MSP-SG	Failure of Secondary Side Cooling (SGTR)	2.6E-03
MSP-SL	Failure of Secondary Side Cooling (Other Initiating Events)	7.0E-03
MSP-SL-PC	Failure of Secondary Side Cooling (PLOCW)	5.0E-02
Reactor Trip		
RTP-LO	Failure of Reactor Trip (LOOP)	1.0E-07
RTP-MF	Failure of Reactor Trip (Other Initiating Events)	1.5E-07

Fault Tree Name	Fault Tree Description	Fault Tree Probability
Main Steam Isolation		
MSR-O-00	Failure of Main Steam Isolation (SLBO)	6.3E-04
MSR-I-00	Failure of Main Steam Isolation (SLBI, FWLB)	4.4E-08
Isolate Rupture SG		
MSP-OS	Failure of Isolate Rupture SG (SGTR)	2.1E-05
Class 1E GTG		
OPS	Failure of All Class 1E GTG (LOOP)	3.9E-05
Alternative GTG Power		
SDG	Failure of Alternative GTG Power (LOOP)	2.5E-02
Alternate CCW		
ACW	Failure of Alternate CCW	6.7E-03
ACW-LP1	Failure of Alternate CCW (LOOP)	8.2E-03
ACW-LP2	Failure of Alternate CCW (LOOP: Alternative GTG)	8.2E-03
SEC-PLOCW	Failure of Alternative GTG Power (LOCWS)	4.6E-03
CCW Re-Start		
CWS-R2	Failure of CCW Re-Start (Alternative GTG)	2.7E-02
CWS-R4-LP1	Failure of CCW Re-Start (LOOP)	2.2E-04
CWS-R4-LP2	Failure of CCW Re-Start (LOOP : No breakdown)	1.7E-04
Other Headings		
HIT	Failure of Injection Control (SGTR)	2.7E-02
MFW	Failure of Main Water System Recovery (TRANS ,LOAC, LODC)	1.0E-01
RCP-SEAL	RCP Seal LOCA Occurs	1.0E+00
POV	Safety Relief Valve LOCA Occurs	2.8E-04
PRB	Failure of Power Recovery (1 hour) (LOOP)	5.3E-01
PRC	Failure of Power Recovery (3 hours) (LOOP)	4.1E-01
IE-CCW-SWS	LOCWS IE (LOCWS)	2.4E-05
TTP	Failure of Turbine Trip	1.0E-02
MTC	Moderator Temperature Coefficient	5.0E-02
PZS	Failure of Pressurizer Safety Valve Operation	1.0E-02
EBI	Failure of Boric Acid Transfer	2.6E-02

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TABLE 4.1.5. Calculated failure probability of support line system

Fault Tree Name	Fault Tree Description	Fault Tree Probability
RWSP		
RWS	RWSP	1.1E-05
Heating Ventilation and Conditioning System		
HVA-EFW-A	EFW area HVAC B Train	4.4E-02
HVA-EFW-B	EFW area HVAC C Train	8.9E-03
Component Cooling Water System		
CWS-00A	A Train	7.4E-04
CWS-00B	B Train	3.1E-02
CWS-00C	C Train	9.6E-04
CWS-00D	D Train	3.1E-02
CWS-VS-00A1	Charging Pump Cooling A Train	6.5E-05
CWS-VS-00C1	Charging Pump Cooling C Train	7.2E-05
Essential Service Water System		
SWS-01A	A Train	6.0E-04
SWS-01B	B Train	1.7E-02
SWS-01C	C Train	8.2E-04
SWS-01D	D Train	1.7E-02
S SIGNAL		
SGN-SA	A Train	1.1E-03
SGN-SB	B Train	1.1E-03
SGN-SC	C Train	1.1E-03
SGN-SD	D Train	1.1E-03
S SIGNAL (Manual DAS Failure)		
SGN-SA-DAS	A Train	7.2E-05
SGN-SB-DAS	B Train	7.2E-05
SGN-SC-DAS	C Train	7.2E-05
SGN-SD-DAS	D Train	7.2E-05
P SIGNAL		
SGN-PA	A Train	1.2E-03
SGN-PB	B Train	1.2E-03
SGN-PC	C Train	1.2E-03
SGN-PD	D Train	1.2E-03

Fault Tree Name	Fault Tree Description	Fault Tree Probability
A (B-D) Class 1E 6.9kV Bus		
EPS-69KA	A Train	2.5E-05
EPS-69KB	B Train	2.4E-04
EPS-69KC	C Train	2.4E-04
EPS-69KD	D Train	2.5E-05
Non-Class 1E 6.9kV Bus		
EPS-69KA-P1	A Train (Power Source only AAC)	6.5E-02
EPS-69KD-P2	D Train (Power Source only AAC)	6.5E-02
A (B-D) Class 1E 480V Load Center Bus		
EPS-480A	A Train	4.5E-05
EPS-480B	B Train	2.6E-04
EPS-480C	C Train	2.6E-04
EPS-480D	D Train	4.5E-05
Non-Class 1E 480V Load Center Bus		
EPS-480A-P1	A Train (Power Source only AAC)	6.5E-02
EPS-480D-P2	D Train (Power Source only AAC)	6.5E-02
A1 (D1) Class 1E 480V Load Center Bus		
EPS-48A1	A1 Train	5.8E-05
EPS-48D1	D1 Train	5.8E-05
A (B-D) Class 1E 480V Motor Control Center Bus		
EPS-MCA	A Train	5.5E-05
EPS-MCD	D Train	2.7E-04
EPS-MCC	C Train	2.7E-04
EPS-MCD	D Train	5.5E-05
Non Class 1E 480V Motor Control Center Bus		
EPS-MCA-P1	A Train (Power Source only AAC)	6.5E-02
EPS-MCD-P2	D Train (Power Source only AAC)	6.5E-02
A1 (D1) Class 1E 480V Motor Control Center Bus		
EPS-MCA1	A1 Train	6.7E-05
EPS-MCD1	D1 Train	6.7E-05

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4.1.2.4.3. CR 1.4.1. Frequency of release into the containment

- Calculated core damage frequency (CDF)

  Reliability of engineered safety features (ESFs) cannot be compared directly with the existing PRA results. However, the enhancement of the ESF design can be achieved by the adoption of four train concept (front line / support system / Emergency Power Sources (EPS) (e.g. Gas Turbine Generators (GTGs)), in-containment RWSP makes good effect in CDF calculation in comparison to the conventional PWR, against DBA.

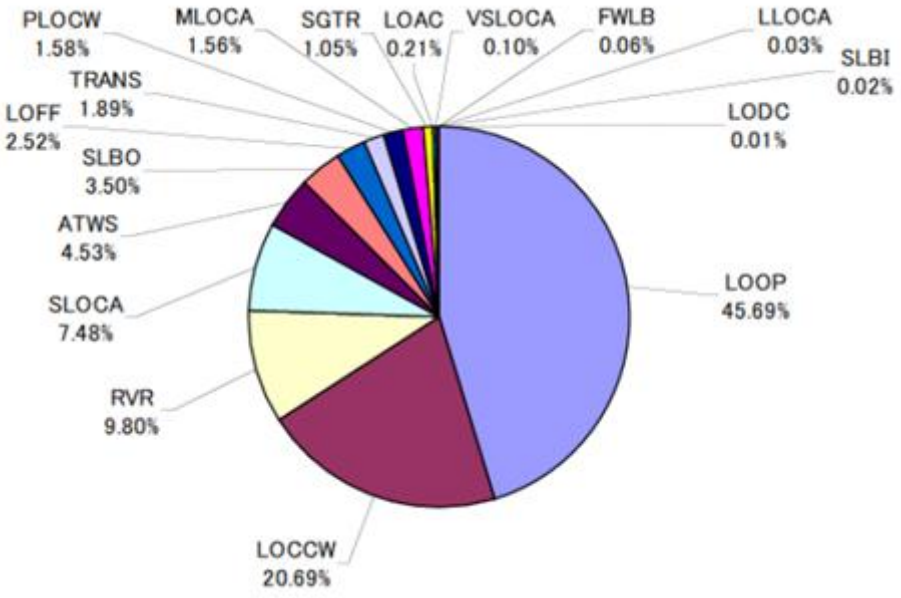
- Prevention of severe accidents

The countermeasures against 'beyond DBA' are implemented. To cope with CCF or multiple failures, diversity or independence must be considered and these countermeasures contribute greatly to the reduction of CDF. Examples of countermeasures:

- Anticipated Transient Without Scram (ATWS): Installation of diverse actuation system to cope with CCF
- Station blackout (SBO): Installation of additional alternate power source by two-train GTGs, besides four-train emergency gas turbine generators (GTGs)
- Fire: Clear physical separation between the redundant trains of safety systems
- Intersystem LOCA: Prevention of over-pressurization of the residual heat removal system

The CDF of the US-APWR for internal initiating events are as follows:

- CDF due to events except fire and flood at power operation: 1.0×10^{-6}/RY
- CDF due to fire at power operation: 8.6×10^{-7}/RY
- CDF due to flood at power operation: 8.9×10^{-7}/RY
- CDF due to events except fire and flood at low-power and shutdown: 1.8×10^{-7}/RY
- CDF due to fire at low-power and shutdown: 1.8×10^{-8}/RY
- CDF due to flood at low-power and shutdown: 9.5×10^{-8}/RY



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FIGURE 4.1.1. CDF Contribution at power operation (internal events)

The total frequencies including for shutdown mode amount to 3.0×10^{-6}/RY which satisfies the requirements by American authority (1.0×10^{-4}/RY) and URD by EPRI (1.0×10^{-5}/RY) respectively. Meanwhile, the CDFs for external initiating events have not been calculated for the US-APWR.

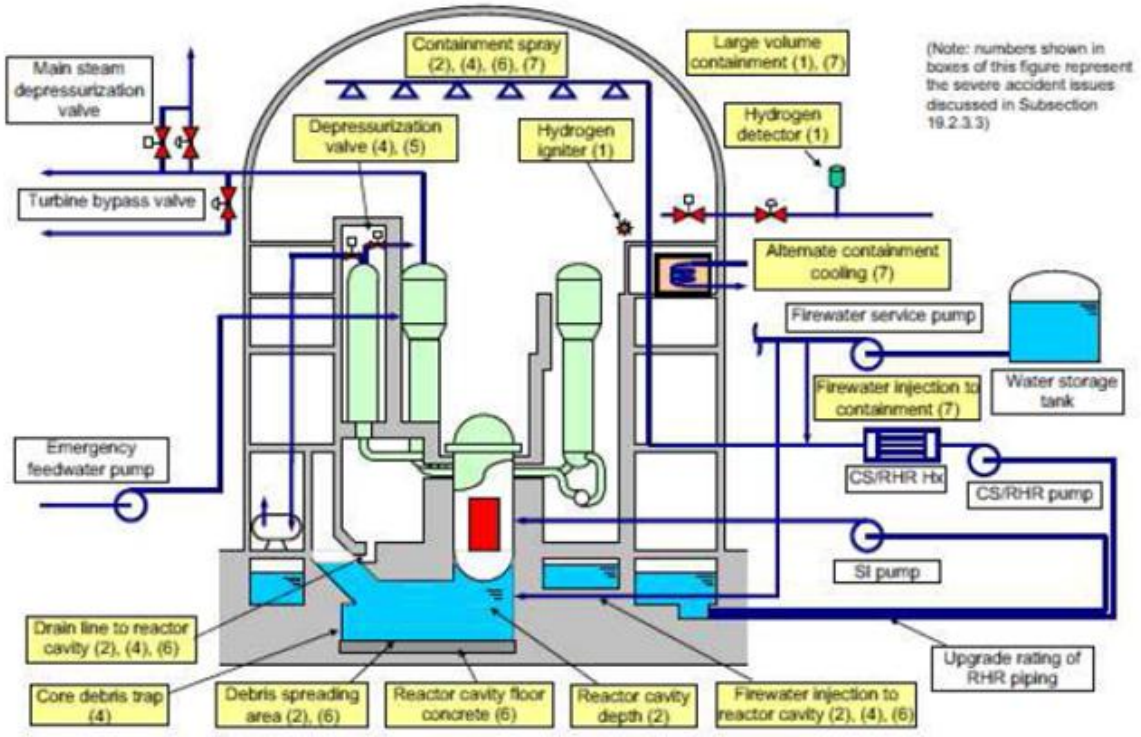
In view that the AL 1.4.1 requires comparison with the existing design, Mr Yuasa noted different reliability database was used for PRA calculation for the US-APWR and the conventional PWR. The CDF due to events except fire and flood at power operation for the conventional PWR is 1.7×10^{-7}/RY compared to

1.0×10^{-6}/RY for the US-PWR. PRA calculations in Japan used reliability databases developed based on Japanese maintenance management and operational experiences for the conventional PWR. This influences the frequencies of initiating events and reliabilities of safety functions for the PRA. Therefore, he concluded that it is not appropriate to compare the results of PRA directly between the US-APWR and the conventional PWR.

4.1.2.4.4. CR 1.4.2. Processes for controlling relevant system parameters and activity levels in containment

The containment for the US-APWR is large volume type pre-stressed concrete containment (PCCV), designed as an essentially leak-tight barrier. As to severe accident mitigation, analyses were performed for accident progression and necessary countermeasures are installed, such as for:

- Hydrogen generation and control – achieved by igniter with continuous monitoring of hydrogen concentration
- Core debris coolability – achieved by reactor cavity flooding system
- Steam explosion (in-vessel and ex-vessel) – confirmed by analysis
- High-pressure melt ejection
- Temperature-induced SG tube rupture (containment bypass) – prevented by depressurization barrier dedicated for severe accident at pressurizer
- Molten core concrete interaction (MCCI) – a phenomenon that may lead to potential basemat melt-through which is also prevented by reactor cavity flooding system
- Long-term containment overpressure – prevented by diverse air cooling units cooled by containment cooling unit (CCU) system.



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FIGURE 4.1.2. US-APWR Severe Accident Mitigation Features

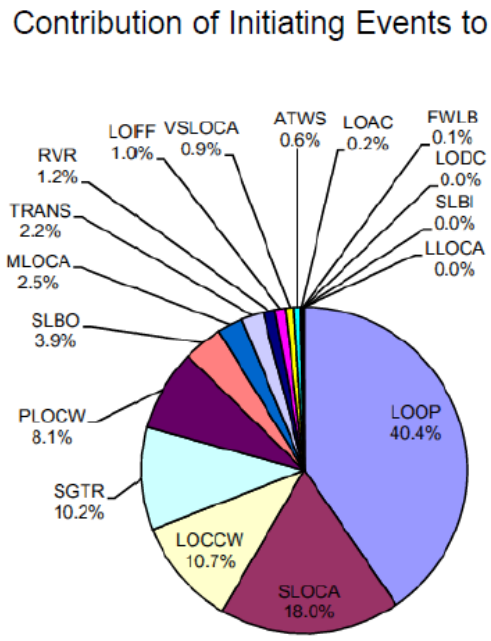
For details, please refer to the supplement document prepared by MHI for the Dialogue Forum, which is also available in Design Control Document Chapter 19. It was confirmed that the frequency of loss of containment function is very small.

4.1.2.4.5. CR1.5.1. Frequency of release to the environment

The probabilities of large releases (LRF - Large Release Frequency) of radioactivity to the environment due to the loss of containment function were calculated for internal initiating events.

- LRF due to events except fire and flood at power operation: 1.1×10^{-7}/RY
- LRF due to fire at power operation: 1.9×10^{-7}/RY
- LRF due to flood at power operation: 1.6×10^{-7}/RY
- LRF due to events except fire and flood at low-power & shutdown: 1.8×10^{-7}/RY
- LRF due to fire at low-power and shutdown: 1.8×10^{-8}/RY
- LRF due to flood at low-power and shutdown: 9.5×10^{-8}/RY

The total value is ~6.7×10^{-7}/RY which complies with the requirement by AL 1.5.1 which is 1.0×10^{-6}/RY.



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FIGURE 4.1.3. Contribution of initiating events to large release frequency (LRF)

Evidences concerning severe core damaged frequency of the US-APWR design in comparison to existing design are available to the INPRO assessor and have been demonstrated. The superiority of the US-APWR design to existing design in regard to the AL1.3.1, AL1.3.3, AL1.4.1, AL1.4.2 and AL1.5.1 is shown. Some ALs cannot be confirmed quantitatively but the facts compose partial satisfaction of the respective UR1.3, 1.4 and 1.5.

4.1.2.5. Detection of Anomalous condition and safety analysis

4.1.2.5.1. EP1.2.1.1 Continuous monitoring of the plant health

Condition monitoring systems in the US-APWR to prevent AOO (anticipated occupational occurrences) from escalating accidents:

- Leak monitoring systems.

  The system provides the means of identifying the source and monitoring leaks from the reactor coolant pressure boundary (RCPB). 'RCPB leak before break' concept is applied to the main coolant piping and large branch lines. Early detection of leakage leads to reduction of components design load such for core internals, other components and supports and also leads to elimination of pipe whip restraint.
  - Unidentified leakage from RCPB is detected by in-containment radioactivity monitors (air particles / airborne gaseous), containment air cooler condensate monitoring and CV sump water level in less than one hour at 0.5 gpm at flow rate or equivalent.
  - Identified leakage, e.g. intersystem leakage through valves connected to RCS, such as safety injection line, RHR suction line, is detected by resistance temperature detectors (RTDs) mounted on piping surface.
  - Identified leakage through SG tubes is detected by radiation monitors in the secondary side (MSS, SGBDS, condenser air exhaust pump lines, etc.)

  Other leak monitoring systems

- Leakage from radiation control area is detected by leak monitoring (water level) equipment in each floor.

- Loose parts monitoring: The loose parts monitoring system provides early detection of loose metallic parts in the RCS, using multiple redundant field mounted sensors (piezoelectric accelerometers), fixed on the RCS components.
- Core internal vibration monitoring system: The ex-core neutron detectors are used for continuous monitoring of core vibration. Correlation between the detected signals and the vibration characteristics will be confirmed during a preoperational test.
- Diagnostic of rotating machinery: The vibration monitoring system is installed for key pumps to plant safety or power operation, such as RCPs and safety related pumps.
- Chemical monitoring: The sampling systems are installed on both primary and secondary side to collect and monitor various samples, which are essential for plant safety and operation. Each sample is analysed to the conformance with the chemistry specification (boron concentration, pH, radioactivity, conductivity and dissolved gases, etc.)
- Seismic monitoring system: The seismic instrumentation is designed with a solid-state multi-channel digital instrumentation with computerized recording and playback capability that allows the seismic response of the plant features important to safety can be evaluated promptly after an earthquake (within 4 hours). The results are annunciated in the MCR, to be used for shutdown determination. In Japan (an earthquake country), the plants will be shutdown automatically in the event of low-level earthquake. This depends on site condition and regulation in the country.
- Computerized aids to operators. Computer-based procedure system supports operators to take appropriate actions to cope with anomaly by providing information in a real-time manner, e.g., potential causes of the anomaly, navigations to acknowledge the anomalous influence/location, possible countermeasures and consequences of operator's actions.

4.1.2.5.2. EP1.2.1.2 Dynamic plant analysis

On assessment in a real situation, it is considered appropriate in an actual assessment to check all evidences and to confirm if key system variables relevant to safety do not exceed limits acceptable for continued operation. The evidences are available on the NRC website; http://www.nrc.gov/reactors/new-reactors/design-cert/apwr.html where a package of design control documents is available: Chapter 7 "Instrumentation and Controls", Chapter 15 "Transient and Accident Analysis" and Chapter 19 "Probabilistic Risk Assessment and Severe Accident Evaluation" describe how the key system variables relevant to safety do not exceed limits acceptable for continued operation. During the Dialogue Forum, it was considered impossible to explain behaviours of all key system variables relevant to safety in a short time. However, it's obvious that assessors can access and confirm the evidence indicating satisfaction of the Acceptability of EP1.2.1.2 by available documents.

Evidences indicating provisions for detecting anomalous conditions was shown and it was confirmed that the US-APWR has inherent characteristics, continuous monitoring capability of the plant health and an instrumentation and control system to detect and intercept and/or compensate deviations from normal operational states. Deterministic and probabilistic plant analysis are also included in the evidences. The US-APWR design fulfils the Acceptance Limits of EP1.2.1.1 and EP1.2.1.2, which composes partially satisfaction of UR1.2.

4.1.2.6. Integrated Full Digital I&C System and Advanced Operator Console

Mr Fujii (Mitsubishi) presented the materials for 3 selected CRs/EPs

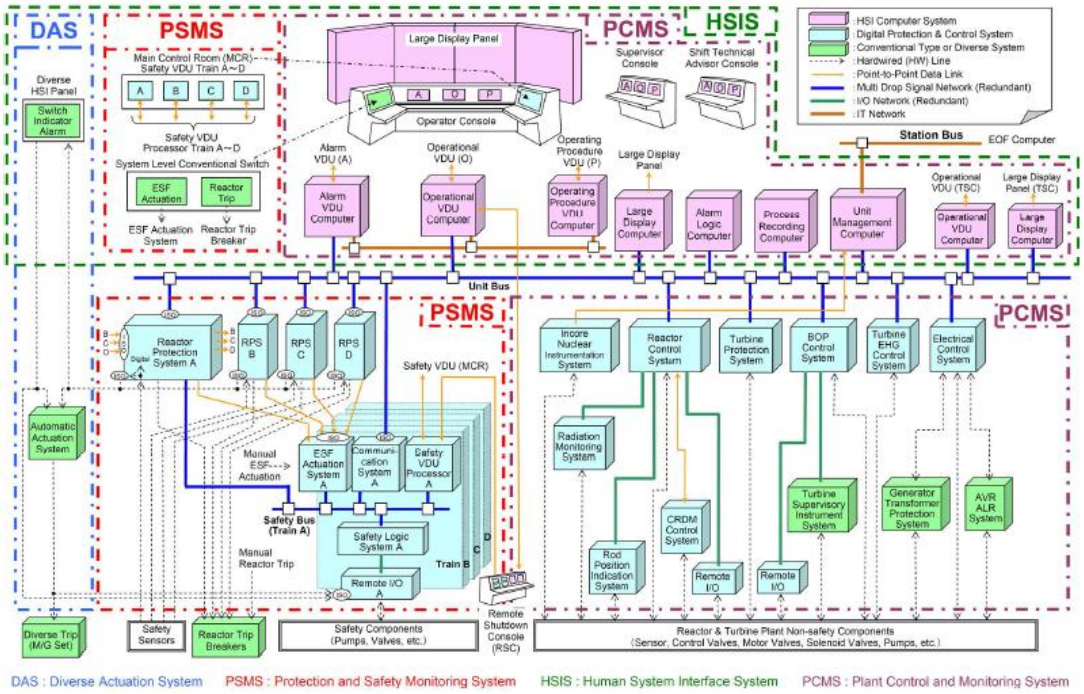
- EP1.1.2.2 Reliability of control systems
- EP1.1.2.3 Reduced impact of incorrect human intervention
- CR1.7.1. Human factors

Design Features and Design Basis of Integrated I&C System

Fully digitalized technology is applied to the I&C system for both safety and non-safety functions. The reactor protection system and actuation system of engineered safeguard features (ESF) have 4-time

redundancy in the I&C system. Conventional operating and monitoring devices such as switches and indicators have been eliminated, except devices for diverse actuation system (DAS).

System	Features
Main control board	• Fully computerized • Visual Display Units (VDU) for safety and non-safety channels • Limited number of conventional switches and indicators
Safety I&C	• Fully digitalized with Mitsubishi digital controllers • Four-train redundancy in reactor trip system, ESF actuation system and safety logic system for component control
Non-safety I&C	• Fully digitalized with Mitsubishi digital controllers • Duplex digital architecture for each control and process monitoring sub-system
Data communication	• Fully multiplexed, including class 1E signals • Multi-drop data bus and serial data link • Fiber optics communication networks for noise immunity and required isolation



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FIGURE 4.1.4. I&C System Overview of the APWR

Applying digital based system brings more attractive advantages which enhance the safety performance and reliability of overall I&C system, for examples:

- Lower potential for human error and reduce workload on operators by less human interaction and improved user interface for remaining manual tests, calibrations and repair
- Higher availability and reliability based on continuous self-testing and additional redundancy
- Higher accuracy by non-drift nature and self-compensation function
- Better performance based on advanced algorithms and to increased automation and improved user interface
- Reduction of maintenance workload and resources by standardized design

The I&C system of the US-APWR satisfies requirements of (i) Official regulations such as Regulatory Guide, Standard Review Plan and relevant industrial codes & standards including single failure criterion, redundancy, independence, testability, control and indication of bypass operation, countermeasure for common mode failure, life cycle management for digital component, electromagnetic compatibility, and (ii) Utility's requests such as the maintenance during the operation.

Diverse actuation system (DAS)

A diverse actuation system (DAS) is installed to provide back-up actuations for safety and non-safety components as countermeasures against common cause failure (CCF) in the software of the digital safety

I&C systems. The DAS consists of diverse devices from the digital safety system so that a CCF in the digital system doesn't impair the DAS functions. The DAS is designed for beyond design basis CCF incidents. The design ensures the DAS can sustain one random component failure without its spurious actuation to avoid adverse effects on the plant availability.

4.1.2.6.1. EP1.1.2.2 Reliability of I&C System

Reliability of the integrated digital I&C system is secured by:

- Use of digital equipment which has sufficient experiences in power plants before being employed for the US-APWR. Thus, before being used for NPP, the hardware and software were used for coal and oil-fired power plants, then the hardware and software are structured and enlarged so the reliability is confirmed in advance to the usage in the US-APWR.
- 4-time redundancy for reactor protection system and actuation system of engineered safety features (ESF)
- Confirmation of soundness and perfection of software by verification and validation (V&V) program
- Installation of the DAS to avoid adverse effects of the CCF
- Less human interaction and improved user interface which provide lower potential for human error and reduce workload on operators
- Having continuous self-testing function
- Having non-drift nature and self-compensation function which lead to higher accuracy

All these strengths can only be obtained from an integrated I&C system, and not from conventional solid state type.

Countermeasures against incorrect human intervention

In the process of design stage of US-APWR, countermeasures against incorrect human intervention has been systematically considered in a human factor engineering (HFE) program and its study results have been incorporated into designs for hardware and software. Details of the HFE, Human-Machine Interface (HMI) Design and operator supporting functions are as follows. The HFE program of the US-APWR addresses:

- Main control room (MCR)
- Remote shutdown room (RSR)
- Technical support center (TSC)
- Local control stations (LCSs)
- Emergency operations facilities (EOFs)

Usually EOFs are prepared by the electric power company outside/near the NPP, and also MHI does HF program for EOFs for electric power company.

Mr Fujii used the main control room (MCR) as an example where important equipment for monitoring and operation is an object of HMI Design:

- Operator Console
- Large Display Panel
- Diverse HIS Panel
- Supervisor Console
- Shift Technical Advisor (STA) Console

Details of Human-Machine Interface Design in RSR, TSC, LCS and EOF are available in the supplement prepared by Mitsubishi for the Dialogue Forum.

Operator console and Large display panel

Plant parameters and associated operating switches are displayed on same screens, and touch screen operations are applied. Thus, the operator's work load is reduced and the reliability of operation is increased. A large display panel is installed to display major parameters for normal and abnormal situations. This allows the current status of the entire plant to be understood by all crews and their communication is improved. Therefore the digital system provides significant benefits to the safety of nuclear power through reduction in operation works and maintenance work loads, which reduces the potential for human error.

Operation Supporting Functions

By using digital computer system, working conditions of operators are dramatically improved.

- Alarm Control Function using Alarm Logic.

  When a complicate incident occurs, there appear many alarms in the MCR, which make operators confused. The digital I&C system is equipped with alarm control function which executes alarm filtering logic, classifies these alarms according to their priority and their acknowledgement status.

- Electronic Operating Procedures

  Operators shall follow instructions of operating procedures composed of a huge volume. The operating procedures are electronically filed so as to be able to easily be referred. A processor for the exclusive use of the operating procedures is equipped, which manages the displays for the operating procedures on the operator, STA and supervisor consoles.

- Remote Shutdown Console (RSC)

  Operators shall control the reactor in safe condition in any occasions. The US-APWR is equipped with a RSC which has a function to achieve and keep the reactor in a safe shutdown condition without the MCR. The RSC is also considered in accordance with human performance engineering.

Verification & Validation of Human-Machine Interface Design

- V&V for Human Factors Engineering (HFE). The goal of the HFE program was to achieve that an adequate HFE program was developed and implemented. To meet the HFE goals, a V&V program was designed and begun with the objective of supplying HFE guidance to the design team. This includes computer-based operating procedures, alarm prioritization, touch-operation, large display panel with variable display and consoles for operators, shift supervisor and shift technical advisor.
- V&V for Human-Machine Interface (HMI) (Human-System Interface, HSI). The goal of the V&V testing was to evaluate the digital HSI design and the performance of crews when using the HSI design.
- V&V Methodology was based on the V&V testing to support the HSI design as generally described in NUREG-0711. Testing employed:
  - Experienced plant crews as test participants
  - Realistic normal and emergency scenarios
  - Collection of objective data of operator performance as well as subjective operator feedback collected via questionnaires and verbal debrief sessions.

A V&V facility with a plant simulator has been built for the principal purpose of supporting the V&V activities for the US-APWR HSI. When the design was verified to the US standards NUREG-0700, it was found to have excellent compliance.

4.1.2.6.2. EP1.1.2.3: Reduced impact of incorrect human intervention

Reduction of impacts by incorrect human intervention is achieved in comparison to the conventional PWR by:

- Systematical consideration in a human factor engineering (HFE) program and incorporation of its study results into designs for hardware and software.
- The digital system providing the reduction in operation works and maintenance work loads, which reduces the potential for human error.
- The operation supporting functions such as; alarm logic, electronic operating procedures and remote shutdown console.
- The V&V testing to evaluate the digital HSI design and the performance of crews.

4.1.2.6.3. CR1.7.1: Systematical consideration of human factors

Human factor has been systematically considered for the US-APWR by:

- Incorporating the HFE program into designs of MCR, RSR, TSC, LCSs and EOFs
- Incorporating the HMI design into designs of the MCR, RSR, TSC, LCS and EOF
- Testing V&V of the HMI design by using a V&V facility with a plant simulator

Evidences indicating improvements in the reliability of control system, impact of incorrect human intervention and human machine interface of US-APWR design in comparison to existing design are

available to the INPRO assessor and have been demonstrated. The superiority of the US-APWR design to existing design in regard to the acceptability of AL1.1.2 and AL1.7.1 is shown, which composes partially satisfaction of UR1.1 and UR1.7 respectively.

4.1.3. Availability of References

The design control document (DCD) of the US-APWR is available on the NRC website. It consists of about 9,000 pages that can be used for INPRO assessment except for economics and construction cost because they are specific to site location, country, and environment condition.

4.1.4. Discussion

Questions	Answers
Mr Popov (Belarus) asked how the introduction of the new safety system influences the economics of APWR.	**Mr Fujii (Mitsubishi)** replied that reduction of production cost has been done by using scale up of reactor power and electric output. If the capacity increases twice, then the cost of construction increases only by 1.4 – 1.5 times. On the contrary, for safety the design eliminates only safety injection system but adding 4 train system which is not a reduction in cost.
Mr Popov (Belarus) asked the main regime of operation of APWR whether it is based on load-following.	**Mr Fujii (Mitsubishi)** explained that load-following operation can be done in few hours, e.g. from 100% to 50% in 1 hour and stay for about 6 hours before increasing the reactor power back to 100% in 1 hour. This can be done on a daily basis for a total reactor lifetime of 60 years.
Mr Grudev (Bulgaria) asked the failure rate of reactor vessel.	**Mr Fujii (Mitsubishi)** said it was to be discussed later on.
Ms Yang (China) wanted to confirm whether the assessment was for J-APWR or US-APWR. She added that assessment would need lots of indicators/parameters/ coefficients of US-APWR.	**Mr Fujii (Mitsubishi)** replied the J-APWR and US-APWR are slightly different, and for the purpose of INPRO assessment the document for US-APWR is open, unlike that of J-APWR. Mitsubishi is doing design and construction for the US utility company, i.e. Mitsubishi owns the document.
Mr Alshammari (IAEA) asked whether the coolant temperature is in supercritical region.	**Mr Fujii (Mitsubishi)** clarified that the pressure ~15 MPa (gauge) so the maximum coolant temperature is 325 °C, meaning subcooled by more than 18 °C.
Ms Yang (China) asked how MHI used the INPRO criteria so the objective meets the INPRO criteria. She asked whether the criteria are only for reactor vendors, or a holistic approach.	**Mr Fujii (Mitsubishi)** explained that the target of INPRO Methodology is not for vendors but to customers / utility. Vendors could provide information to utility, and they must perform the assessment by themselves. Vendors do not do the assessment. MHI was requested by the IAEA to make the INPRO assessment **Mr Kupitz (Chairman)** added the assessment has been made based on the INPRO Methodology. To perform the assessment, one requires specific information about the design. Vendors are in the position to do the assessment. In the meeting, vendors demonstrated how to do the assessment and what kind of data was used.
Mr Zidi (Algeria) asked for the MHI statement in the conclusion 'partially satisfaction of User Requirement'.	**Mr Fujii (Mitsubishi)** explained that MHI only selected several criteria for demonstrating the assessment of INPRO Methodology for the Dialogue Forum. Full assessment would require time and money for MHI.
Mr Tjahjono (Indonesia) asked the redundancy in case of failure of water source which is said to serve as a single source in the US-APWR.	**Mr Yuasa (Mitsubishi)** explained that the water source RWSP is located circumferentially in the lower floor / bottom of the containment so leakage would not occur. The lower portion of containment is covered by concrete and the water source is installed in the concrete. The injection water will drop again to the water source inside the containment, so recirculation is continued during accident.

Ms Yang (China) asked for clarification of the meaning 'Clear separation between the trains (elimination of tie lines)' given the 2 trains in the conventional PWR and 4 trains in the US-APWR.	**Mr Yuasa (Mitsubishi)** explained that no tie-lines are required for the US-APWR 4-train system because 50% is enough for core cooling capacity, so if one line fails as single failure and one line is on maintenance, then two lines are enough. The hot line injection is to prevent boron extract at core region, and changing operation is required during accident (this serves as another function).
Mr Grudev (Bulgaria) said that in the presentation it was mentioned hydrogen concentration is controlled by igniters. He wanted to know if there is enough oxygen to the end.	**Mr Yuasa (Mitsubishi)** explained that in PWR containment in PWR is usually covered by air atmosphere in which oxygen exists. The concerned of explosion when oxygen combined with hydrogen is confirmed by analysis for the integrity of structure inside the containment and the containment itself.
Mr Reyes (Chile) asked the PWR basic design that the US-APWR was being compared to.	**Mr Fujii (Mitsubishi)** indicated that it was Ohi Unit 4. (*Note that Ohi 4 generates 1180 MWe*)
Mr Zhou (China) asked about the continuous monitoring system of radioactivity. It does not include the fuel damage detecting system and the core damage assessment system. He further asked whether the two systems in the conventional PWR and US-APWR have small difference.	**Mr Yuasa (Mitsubishi)** explained that Mitsubishi has CVCS (Chemical and Volume Control System), and usually sampling of extracted reactor coolant daily to monitor radioactive fission products. He added that both systems in the conventional PWR and US-APWR have almost similar function, only the capacity is slightly changed.
Mr Susyadi (IAEA) asked the followings. Firstly, supposedly the reactor has the same situation like in Fukushima with a long station blackout and no diesel generation, then how long the battery system can maintain a safe shutdown also including for lighting, monitoring etc. Secondly, whether there is any modification or additional equipment put into the reactor after the Fukushima.	**Mr Fujii (Mitsubishi)** answered that there are safety grade and usual grade battery with a lifetime or capacity of 8 hours. In addition, there is emergency gas turbine system. In case of Fukushima, the emergency diesel generator producing electricity for the safety systems needed cooling water; and in tsunami the diesel generation could not be in operation. For the US-APWR the emergency electric power is supplied by gas turbine generator which does need cooling water and it can be located on the hill/building not near the water, and thus it can escape from tsunami or flooding. So in the case of tsunami, emergency power can be expected from the gas turbine generator. This is big difference between diesel generator and gas turbine generator. He earlier noted and restated later that after Fukushima accident, no enhancement has been introduced to the APWR design.
Mr Grudev (Bulgaria) indicated that in the explanation of quality of human factor, but no mention of education of the operators. What is the preference: bachelor, master degree or without degree but qualified?	**Mr Fujii (Mitsubishi)** explained that it would depend on the country's culture and social system. For example, in Japan most chief supervisors have bachelor degree. In France it is not the case and they need reactor engineers as technical advisor in addition to chief supervisor. In Japan, in the shift team, there are no such technical advisor / reactor engineers. So crew must be considered by electrical power company, not by vendors who supply sufficient technology.
Mr Harper (IAEA) asked about the electronic operating procedures, whether the suggestions pop up at the appropriate time or whether the command signals make things happen or suggestions for the operators. He went on asking how to prevent incorrect human intervention, whether there is an active system that prevents this or an active system that makes things happen without the	**Mr Fujii (Mitsubishi)** explained that in the automatic instruction display system, thee computer system displays applicable operation procedure on the screen. More than 20 years ago, MHI tried to introduce automatic operation, in which operator could accept yes and no. The design was considered but risky. So it was decided that only applicable procedures are displayed on screen in the US-APWR. Operator can refer. In the past operator spent much time checking the documents, whereas now it can be done faster. That represents the first objective of the operator supporting system. On incorrect human

operator's actions – sometimes referred to as the HAL 9000 system, i.e. who is in charge of operating the plant.	intervention, the system in NPP is not like that in the airplane A-380 operating system where operator's action and computer operation may have conflict. In NPP, operator's action has a priority in any occasion, so there is no conflict between operator's action and computer system.

4.2. AES-2006

Mr Mikhail Bykov and Mr Denis Kolchinsky from Rosatom (Russian Federation) presented INPRO assessment in the area of safety of the Russian AES-2006.

4.2.1. Overview of AES-2006 Design

Mr Kolchinsky (Rosatom) presented the overview of the AES-2006 (VVER-1200, V-491) design which is an evolution of the reference plant AES-91 (VVER-1000, an evolution of V-320 design). The VVER translates as Water-Water Power Reactor and is a pressurised water reactor. Main principles and approaches for designing of the AES-2006 are maximizing the use of well-proven solutions and technologies, minimizing costs and timeframes by using experience obtained during construction and operation of reference NPP, and ensuring the required safety level including for design extension condition (DEC) The required safety level includes optimal safety systems configuration based on active (generally for DBA) and passive (generally for BDBA) elements, diversity and redundancy of safety functions, BDBA management, and reduction of human factor influence.

The improvements of AES-2006 in comparison with V-320 are increased power output capacity to 3,200 MWth or 1,197 MWe, increased a gross efficiency coefficient to 37%, improved steam parameters at the exit of steam generator to 7.0 MPa (nominal), and improved safety characteristics, i.e. FCD < 10^{-6}. The reactor has a capacity factor of 92% over its 60 year lifetime. The core fuel duration is of 4 years with refuelling period of 12 months. In terms of plant layout, all buildings of the nuclear island are concentrated around the reactor, which minimizes radioactive influence on the staff and reduces the communication lengths of pipes and cabling between buildings. There is physical division of the buildings which contain safety components into safety trains by the fire protected walls. Turbine building arrangement excludes the possibility of reactor building damage by the missiles from turbine. Provision is made for capabilities of access control to the nuclear island buildings. The optimized systems layout allows for better workflow and reduction of capital and operating costs. The area of a power plant site for two units, including cooling towers and other hydro-technical constructions, is of 90 hectares.

The main safety measures of AES-2006 in comparison with V-320 are:
- Defence in depth and diversity (for all safety functions)
- Redundancy and physical separation, i.e. with 4 trains versus 3 trains
- Safety systems for DBA and measures for BDBA management, e.g. core catcher, passive heat removal systems (PHRS)
- I&C for safety functions, e.g. independence of DID levels, diversity, common cause failure
- Low volume of radioactive emission
- External impacts protection, e.g. aircraft crash, DBE of 0.25 g

Both the AES-91 and AES-2006 have 4 safety trains, double containment and engineering measures for BDBA management. The buildings have an enhanced seismic stability. However, the AES-2006 has containment PHRS (C-PHRS) and PHRS via steam generators (PHRS-SG) for BDBA. There is an independent electricity supply lasting 72 hours. The reactor containment building holds an emergency water storage tank. The heat removal system comprises of a combination of passive and active controls. The plant withstood stress tests external impacts such as flooding, tsunami, tornado and seismic threshold determining tests for the containment.

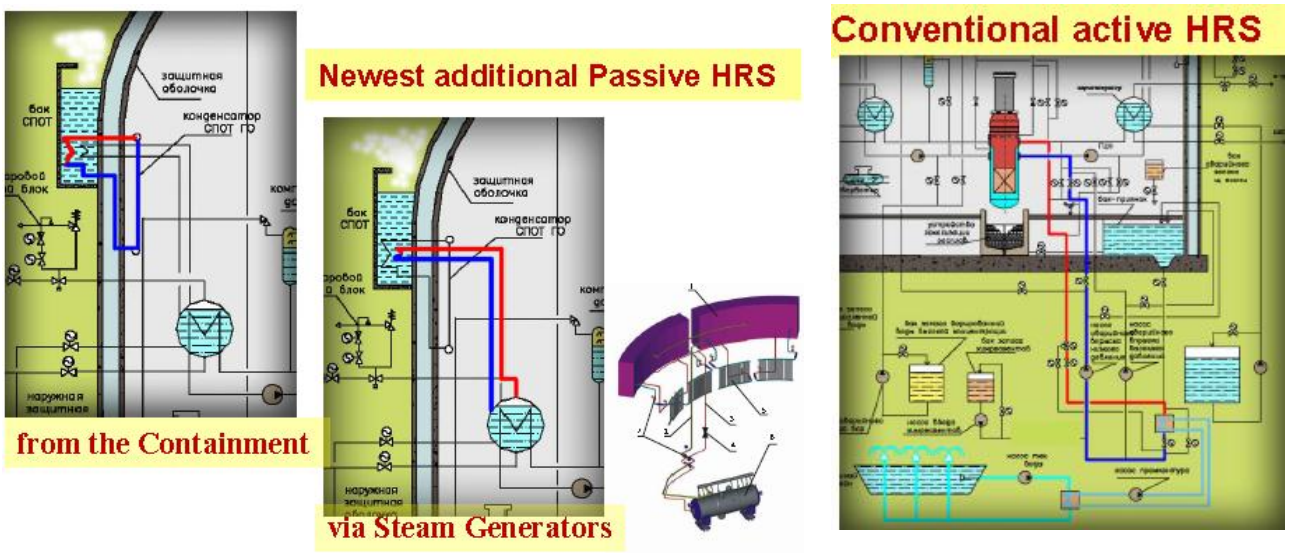


FIGURE 4.2.1. Passive heat removal system (PHRS) (additional to conventional active heat removal system (HRS))

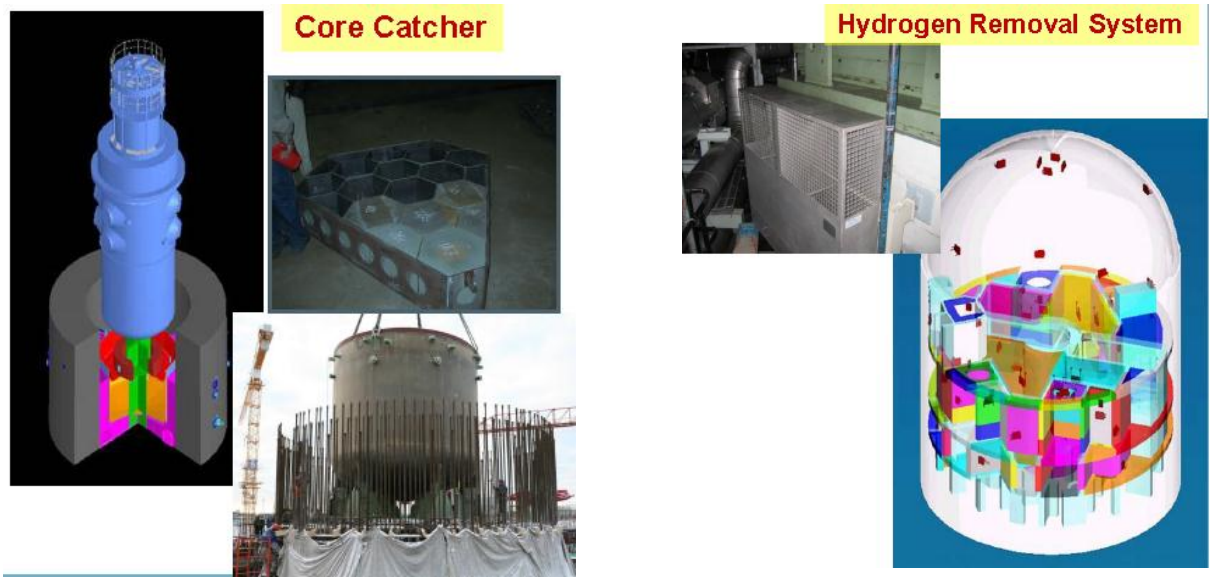


FIGURE 4.2.2. Core catcher and hydrogen removal system as measures for severe accident management

Mr Kolchinsky explained that in the previous designs there were only three categories of design basis conditions (DBC). In the deterministic safety analysis of the AES-2006, however, as per the level of possible negative consequences and an occurrence probability, the list of design conditions is divided into the several categories: DBC1-4, DEC (Design Extension Conditions) and SA (severe accidents. For each category of design conditions, the acceptance criteria are stated and safety analysis is prepared for justification.

The design adopts radiation safety for population for DBA and severe accidents as follows.

- For DBA, the dose incurred by population will not exceed the operational dose limit established for normal NPP operations. The radius of emergency protection area does not exceed 0.8 km around reactor, i.e. limited by site boundary.
- For severe accidents, evacuation of people closely living near the NPP is not required. The radius of area where protection measures for population are planned does not exceed 3 km.

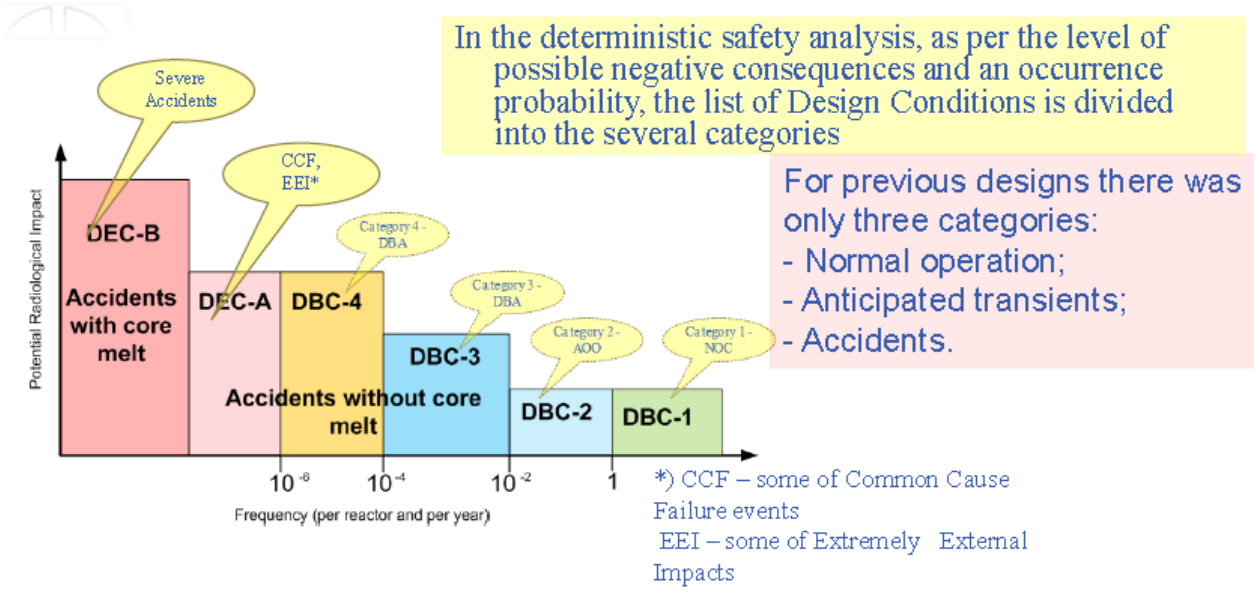


FIGURE 4.2.3. Design basis condition (DBC) and design extension conditions (DEC)

From the Fukushima accident point of view, the safety plant assurance has been improved in that the full list of external impacts (including flooding, tsunami, tornado, etc) was considered. Within the framework of 'stress tests', expert engineering evaluation of seismic strength of inner containment was performed to determine threshold seismic impact value. The analyses takes into account three initiating events, i.e. loss of all electrical power supply sources including SBO, (ii) loss of the ultimate heat sink, and (iii) combination of both.

Evaluation of seismicity resistance analysis was performed by means of the following methods: (i) direct strength analysis of containment as per linear and spectra theory of seismic stability at step-by-step increasing of acceleration level, and (ii) evaluation based on design experience and seismic fragility test of the type of construction as per recommendations of PERI-N-6041. Taking the minimum value of acceleration obtained from the two methods, threshold value of maximum acceleration on the ground level is 0.51 g.

Mr Kolchinsky informed the forum that there are 8 units of AES-2006 under construction in Leningrad, Baltic and Belorussia. The final report of an INPRO assessment of the AES-2006 performed by Belarusian experts, has already been published by the IAEA. The publication is considered as clear evidence that it is necessary for designers and/or technology holders to support or assist potential future NPP owners to prevent the use of unreliable sources and potentially incorrect evaluation. Some positive effects from cooperation were mentioned, among other, to provide independent estimate, to provide more exposure of nuclear technology to the public, and to increase owner's competence level and knowledge.

4.2.2. INPRO assessment in the area of safety of AES-2006

4.2.2.1. EP 1.1.1.1 Margins of design

Mr Bykov (Rosatom) delivered the first assessment on robustness. The robustness criterion is traditionally achieved in consideration of the enhanced seismic resistance of safe shutdown earthquake (SSE) and increased plant lifetime to 60 years. The AES-2006 design is designed with an SSE level of 8 points on the MSK-64 scale (0.24 g at the free surface of the plate) as compared to 7 points in the previous design. (Note the operating basis earthquake (OBE) stays at 6 points). The main equipment maintains the necessary safety margins, i.e. safety margins are not reduced, and additional fixing is used for some pipelines to increase seismic resistance. The availability target of the AES-2006 is increased to 0.92 (from 0.80), thermal power to 3,200 MW (from 3,000 MW) and the secondary design pressure to 8.1 MPa (from 7.84 MPa). Mr Bykov indicated that thermal power and secondary design pressure are related directly to the economic aspects of the NPP. However, the economic aspects contradict the safety of NPP because completely safe NPP means the plant lies in the state of the shutdown. This is the reason why he did not show only the simplification of

the reactor design. There are apparently sets of design which allow to simplify manufacturing, operational procedures and technical assistants of the NPP.

In terms of RPV structure, the inner diameter is increased by 100 mm to decrease neutron flux to the reactor vessel. The length of the vessel is increased by 300 mm and barrel length by 200 mm therefore a reduced power per unit length is obtained. Modifications of the structure lead to the increase in margins of the design, i.e. increased diameter contributes to capability of the protection of the reactor wall against neutron flux and increased length means increased amount of water in upper part of the reactor to provide additional water source in the case of LOCA. Seismic resistance is increased from previous design by reducing the number of welds and optimizing/changing the location of welds across the core. The material quality of the reactor vessel is also improved by using advanced steel to limit nickel content in welds and harmful impurities in base metal and welds. The modification allows a decrease in the critical temperature of the nozzle area shells close to -35°C. There is a new surveillance program to improve reliability of monitoring the fluence to the reactor vessel, i.e. arranging of surveillance specimens to be irradiated immediately on RPV wall. All these modifications result in increased life time to 60 years and increased thermal capacity to 3,200 MWth.

The differences in the design of V-491 (AES-2006) versus V-320 (which AES-91 was based upon):

- Reactor vessel and upper head
  These include the use of 18 control rods, 18 I&C connections located at the periphery, advanced steel material, arrangement of surveillance specimens to be irradiated immediately on RPV wall, the lengthening of vessel by 300 m and reduction of the number of welds and optimization of their locations in the core region.
- Core barrel
  - Application of removable compensation plates to adjust the gap in the spacer collar
  - The height-adjustable supports of fuel assemblies
  - Emergency support in the central part of the bottom to allow passage of coolant in the core barrel breakage accident
  - Application of pressing devices of core barrel based on expanded graphite
- Protection tube unit (PTU)
  - PTU and core baffle are fixed for the core barrel break accident
  - Application for core monitoring only in-core instrumentation detectors channels the same length
- Core baffle
  - Six stops are set to hold the PTU against lateral movement relatively core barrel and allow the SCRAM in case of core barrel break accident
  - Changed location and diameters of the lengthwise holes for better cooling
  - Increased height of the baffle

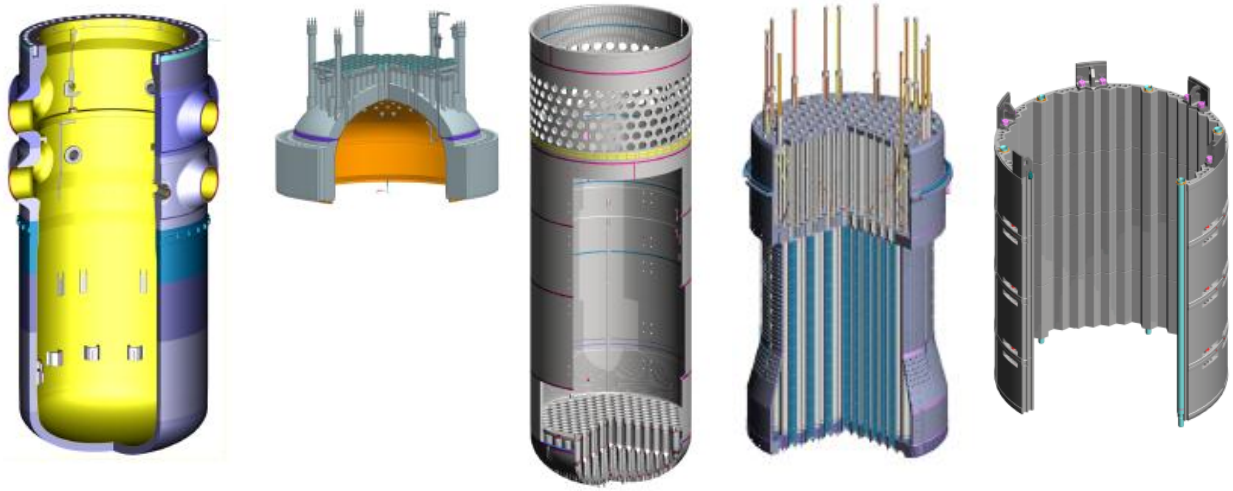


FIGURE 4.2.4. Modifications to the reactor pressure vessel (RPV)

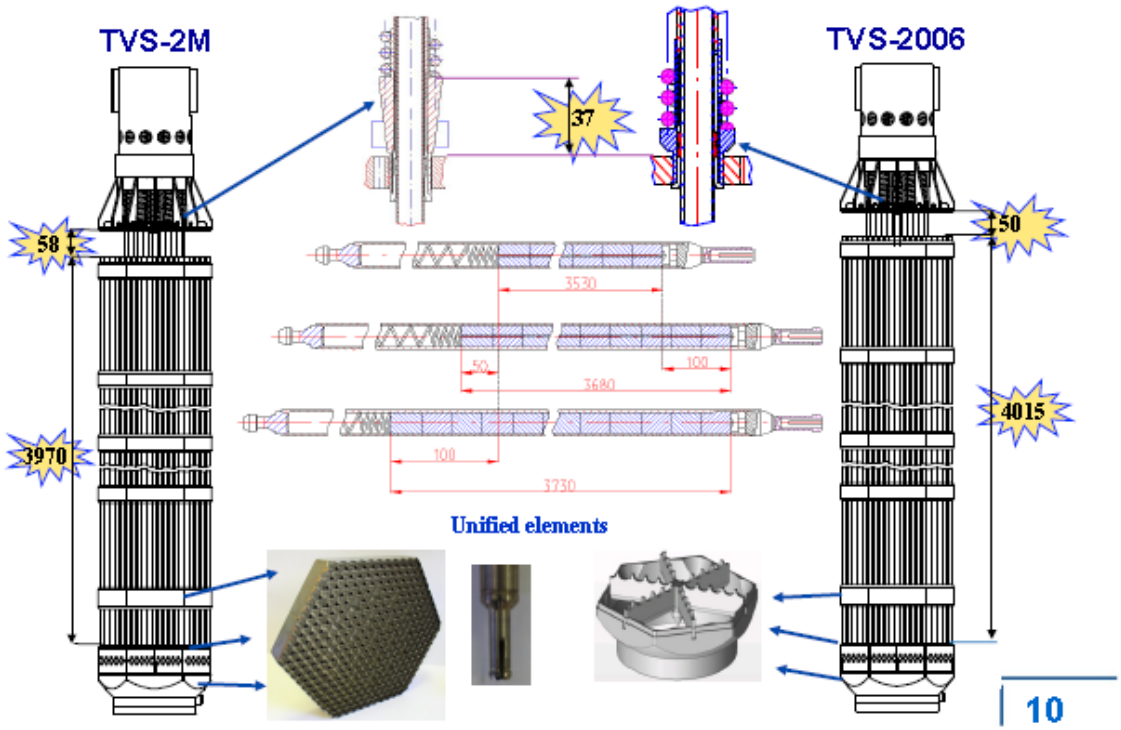


FIGURE 4.2.5. Comparison of TVS-2M and TVS-2006

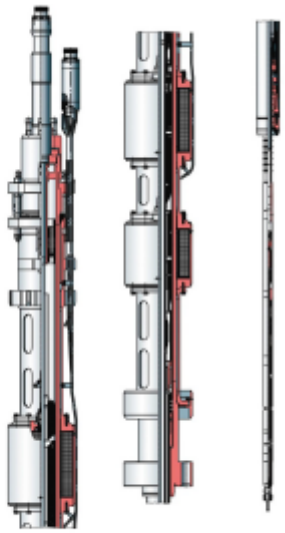
On reactor core, several measures were noted: (i) increased amount of fuel, (ii) reduced power per unit length, (iii) reduced temperature of re-criticality, (iv) increased representation of core measurements by in-core instrumentation detectors, (v) increased number of CPS control rods, and (vi) increased number of places of possible installation of control rods to provide the ability to optimize the location of control rods in a control group. Mr Bykov explained the reasoning behind the decreased maximum linear power with the increased thermal power due to the composition of the core. The re-criticality temperature is also decreased by increasing the number of control rods by up to 121. On core coolability, it was shown the increase made to the reactor outlet pressure, fuel height in the core in the hot state, inlet and outlet coolant temperatures, core heat flux distribution coefficient, maximum fuel cladding temperature, minimum DNBR, and maximum fuel pellet temperature. The fraction of core heat that can be removed by natural circulation is at least 10% N_{nom} (thermal power) in both VVER-1200 and VVER-1000 projects, which is constant, and in addition by the SG PHRS 4×50 (2 MW max) in VVER-1200 (V-491) project.

TABLE 4.2.1. Comparison of VVER-1200 versus VVER-1000

Parameters	VVER-1000	VVER-1200
Nominal thermal power of the reactor (MW)	3000	3200
Core heat flux distribution coefficient	2.60	2.40
Maximum linear power (W/cm)	448	420
Maximum fuel pellet temperature (°C)	1805	1852
Minimum DNBR	1.30	1.39
Fuel height in the core in the cold state (mm)	3530	3730
Number of FAs with CPS CR	61	up to 121
The re-criticality temperature (°C)	220	< 100
Fuel Cycles	3×350; 3×1.5; 4×1; 5×1	3×1.5; 4×1; 5×1
Maximum fuel burn (MWday/kgU)	68	70
Load following modes	-	100-75-100 Nnom
CPS CR	-	The increased length of the section with $Dy_2O_3TiO_2$

TABLE 4.2.2. CPS drive ShEM-3

Parameter	ShEM-2	ShEM -3
1. Operation period, years: - Mechanical units; - Electromagnets; - Position indicator.	 20 10 5	 40 30 30
2. Maximal operated motion of control rod bar, mm	3600	3800
3. Interval of the control rod position discrete check by position indicator, mm	350	20
4. The error of control rod position check by position indicator, mm	⊠30	⊠10
5. Total amount of sensor check control rod positions	11	191
6. Maximal pulling force, kgs	34	70
7. Allowable electromagnet temperature, ^{0}C	170	260
8. Cooling air rate, m3/h	400^{+50}	250
9. Allowable cooling air feed break, h	0,5	2
10. Sealing units	2	1



The steam generator has a larger inner diameter (4.2 m), vaporization area and heat exchange area. There is an increased SG water supply in the secondary circuit from 52 to 63 ton for safety improvement, and enlarged space under heat exchange pipes for easier access for space inspection and sludge removal and overall improved stressed state of coolant collector. The steam generator also has increased design pressure to attain satisfactory power usage coefficient, and increased steaming power.

The pressurizer comprises extended graphite seals in detachable joints and has a 4-stage controllable injection system. Control is modernized to minimize regulators operation and to use natural changes of the primary circuit volume. The individual injection line into the pressurizer comes from the Boron Emergency Injection System. All pipelines including connections are made of stainless steel, which result in the increased life time and enhanced transient process stability, among other, improved values of set points for injection operation. The main coolant pump is also improved with a 5% increased water supply, increased fire safety by using engine lubrication system with uninflammable liquid and leak tightness of seals during 72 hours without cooling. Overall, most of equipment is modernised so as to increase power, seismic resistance, lifetime and the availability factor of the plant.

In conclusion, the assessment indicated preservation or increase of safety margins and reliability of the V-491 design, representing an evolution from the V-320 design, on the following aspects:

- Fuel thermal design and thermal hydraulic design of the V-491, i.e. thermal power, core heat flux distribution coefficient, maximum linear power, maximum fuel pellet temperature, and minimum DNBR
- Fraction of core heat that can be removed by natural circulation, i.e. with the addition SG PHRS in V-491 project
- Parameters characterizing possible overstressing and fatigue of the reactor system components: (i) reduction of RPV temperature of embrittlement of the nozzle area shells -35 °C, (ii) application of removable compensation plates to adjust position of core barrel reduces vibration amplitude of the core barrel, and (iii) changes of location and diameters of the holes of the core baffle for improving cooling reduce the temperature stresses.

Therefore, EP 1.1.1.1 ‘Margins of design’ is met with the AES-2006 design (V-491) design.

4.2.2.2. EP1.1.1.2 Simplicity of design

Mr Bykov (Rosatom) listed the following simplification of reactor unit elements in the AES-2006 design and operation:

- Safety system algorithms owing to the decreased re-criticality temperature
- Reduction of number of control rods detachable joints, i.e. simplification in mounting and maintenance

- Loose requirements to operation due to justification of brittle fracture resistance for ECCS water temperature of 20 °C, i.e. no ECCS water heating is necessary
- Loose requirements for equipment and tubing fastening based on the leak before break (LBB) concept), i.e. simplification in mounting and maintenance
- Simplification of operation due to elements service life increase (Control and Protection System (CPS) control rod drive))
- Reduction of composite welded joints amount in ECCS, Pressurizer System, i.e. simplification in mounting technology simplification
- Reduction of the reactor vessel welds number, resulting in shortening of manufacture time and reduction of operational control scope
- Renunciation of double-chamber equalizing vessels in favour of single chamber vessels
- Corridor heat exchanging pipes bundle in steam generator decreases sludge accumulation, i.e. simplifying sludge removal and check-up of heat exchange area
- Increase of maintenance period from 4 to 8 years
- Gate-type equipment application allowing steam generator maintenance concurrently with fuel reload.

Mr Bykov noted, however, that EP1.1.1.2 is considered not suitable for evolutionary projects on the basis of active systems because the main principle of improving the reliability and protection against common failure is redundancy and diversity.

4.2.2.3. EP1.1.1.3 Quality of manufacture and construction

Mr Bykov (Rosatom) indicated that to achieve quality of manufacture and construction, Rosatom is making use of referent technical solutions and the quality assurance program for designing, manufacturing of reactor plant (RP) equipment and pipes and materials admitted by the regulatory body for equipment manufacture. Besides, Rosatom collaborates with companies that have certificates of the regulatory body for NPP equipment production.

Mr Bykov mentioned there are some changes in principle that Rosatom is currently working on with the international regulatory to certify projects and quality programs.
- Ductile-to-brittle transition temperature decrease, decrease of impurities in the reactor vessel metal
- Application of detachable joints with expanded graphite
- Automated welding of reactor circulation pipes
- Application of stainless steel pipes for connecting line and ECCS pipes – reduction of number of composite welded joints
- Fewer number of welded joints for reactor vessel
- Provision of the conditions for “clean mounting” of the equipment at manufacturing factories and during NPP mounting.

Mr Bykov indicated that the implementation of EP1.1.1.3 is fully represented in the Belarus NESA report, and the assessment shows that EP 1.1.1.3 is met. (*The report also shows that EP1.1.1.4 and EP1.1.1.5 are met, i.e. CR1.1.1 has been fully met for the V-491 design).*

4.2.2.4. CR 1.2.1 I&C and inherent characteristics

Mr Bykov (Rosatom) pointed out that the criterion is met through the use of monitoring, control and diagnostic system (MCDS), which is designed based on the principle of combining the functionally completed systems that fully perform their functions and are joined by data exchange. The goals are to monitor the core and reactor plant parameters; provide scram and preventive protection signals by local parameters (DNBR, linear power); advise the operator about power field control; monitor vibration load for reactor plant elements, loose parts and primary circuit tightness and remaining life; present the current core and reactor plant parameters, data exchange of different systems and data archiving; monitor margins for operational limits and safety limits for parameters measured and calculated by MCDS; monitor water chemistry and give recommendations to the operator; and perform self-diagnostics of hardware and software.

The MCDS comprises of several systems: in-core instrumentation system (ICIS), system integrated diagnostics (SID), vibration-noise diagnostics system (VDS), coolant leak detection system (CLDS), loose parts detection system (LPDS), remaining life automated control system (RLACS) and system of integrated

analysis (SIA). The AES-2006 MCDS is an evolution of the V-20 ICIS All these systems within the MCDS are decentralised and used to control the plant and perform diagnostics during its operation.

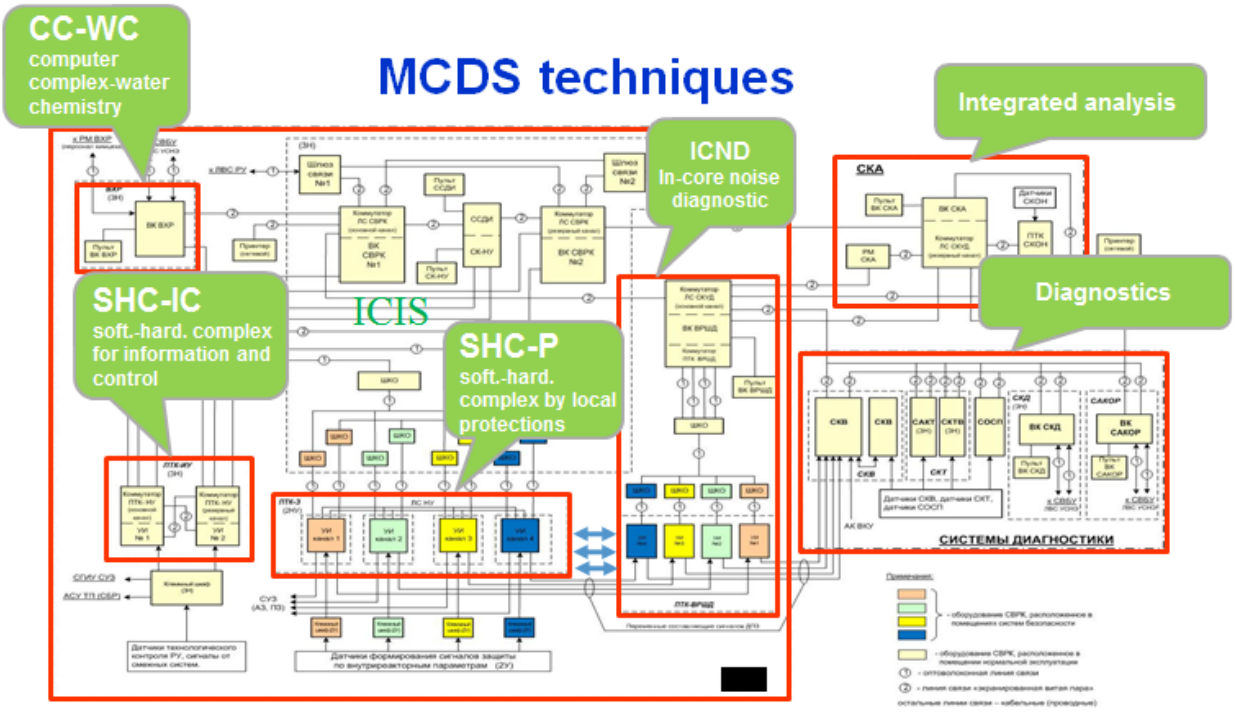


FIGURE 4.2.6. MCDS Techniques

4.2.2.5. CR 1.2.3 Inertia

Mr Bykov (Rosatom) presented the criterion on inertia for the AES-2006. The more comprehensive application of inertia, inherent safety and passive features of the design has resulted in several improvements of this design, such as an increase of 20% of secondary circuit water, the exclusion of combustible liquids inside the containment along with maintaining the required RCP coastdown period, a decrease in the re-criticality temperature, passive heat removal systems for steam generators and containment, an enlargement of the water layer between the core and reactor vessel, the use of stainless steel for the pressurising system and ECCS piping, water chemistry justification for ECCS water (20 °C), longer period for keeping the RCP shaft leak tightness without cooling (72 hours vs 24 hours), justification and application of LBB concept for nominal diameter > 200 mm.

The reliability and safety margins have been preserved or increased in this new design and there is an extended use of inherent safety features, passive principles for simplification of technical solution which enhance performances. The control and monitoring of process variables and parameters characterising the reactors state have also been developed.

4.2.2.6. CR 1.2.2 Grace period (normal operation)

Mr Bykov (Rosatom) noted that safety analyses in the V-491 design are performed by considering the grace period in compliance with international organization requirements. Two of such were pointed out, i.e. the IAEA Safety Report Series No. 23 which requires 15-30 minutes to diagnose the event and take proper corrective action, and the EUR requirement which requires 30 minutes for operator actions from MCR or 1 hour for actions outside MCR. In other words, for V-491, the operator has a 30 minute time margin in case of an accident to diagnose the event and take proper corrective action. In an assessment of DEC, all actions beyond this time margin can be taken into account.

4.2.2.7. CR 1.3.2 Grace period (design basis accident)

Mr Bykov (Rosatom) discussed the grace period for DBA. Accordingly, the AES-2006 design and reliability of safety systems as well as supporting systems allow non-interference of any controls for at least 8 hours, as proven by the fulfilled safety analyses.

It was concluded the fulfilment of this criterion is achieved by the design basis of safety systems and justified in the analysis of DBA and BDBA, i.e. 24 hr for DBA which can be increased to 72 hours with operator action. Therefore, CR 1.3.2 is met.

4.2.2.8. CR 1.3.3 Safety features

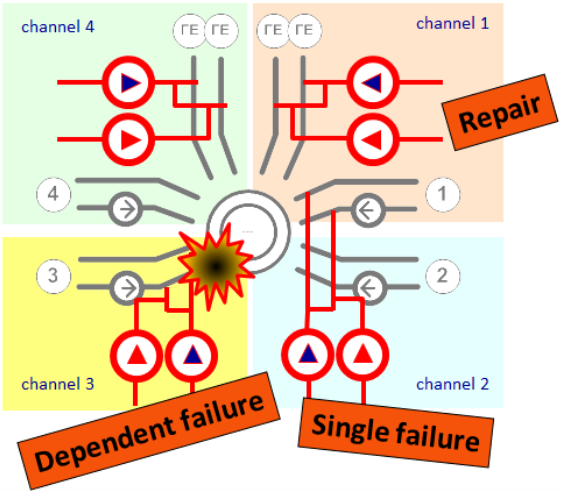
Mr Bykov (Rosatom) discussed the following AES-2006 safety features to meet CR1.3.3.

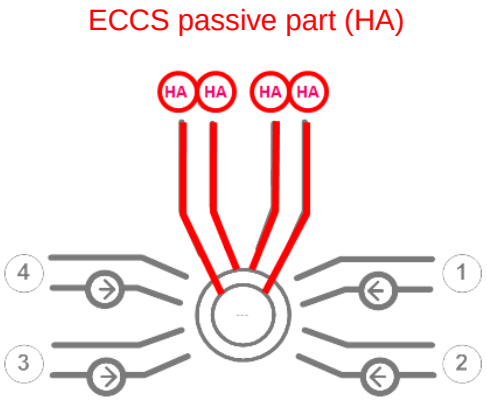
TABLE 4.2.3. Safety features of AES-2006 (V-491) versus V-320

	V-320	V-491
ECCS active part	Separate three-channel maximum- and low-head systems with channel redundancy 3 x 100 % each	Separate four-channel maximum- and low-head systems with channel redundancy 4 x 100 % each — AL complete
ECCS passive part	Four identical channels fully independent of each other	Four identical channels fully independent of each other — no needs*
Emergency boron injection system	Three-channel with channel redundancy 3 x 50 %	Four-channel with channel redundancy 4 x 50 % — no needs*
Emergency feedwater system	Three-channel with channel redundancy 3 x 50 % and tanks of emergency feedwater inventory	Four-channel with channel redundancy 4 x 50 % and tanks of emergency feedwater inventory — AL complete
Passive heat removal system (PHRS)	Not available	Passive four-channel system with channel redundancy and 18 water-cooled heat exchangers in each channel — AL complete
Passive heat removal system from the containment	Not available	Available — AL complete
Core catcher	Not available	Available — AL complete

no needs* — The system makes safety functions during short time, after that it works no needs

The ECCS active part includes high pressure system (HP ECCS) and low pressure systems (LP ECCS) with redundancy of 4×100% each, supplying boric acid solution to the reactor coolant system under LOCA within the whole spectrum of DBAs up to leak with D_{nom} (nominal diameter) of 850. The EECS passive part consists of four identical and independent channels, each comprising ECCS accumulator, valves and pipelines. It supplies boric acid solution to the reactor at the primary pressure less than 5.9 MPA under DBA with loss of coolant until the start of the LP pumps.



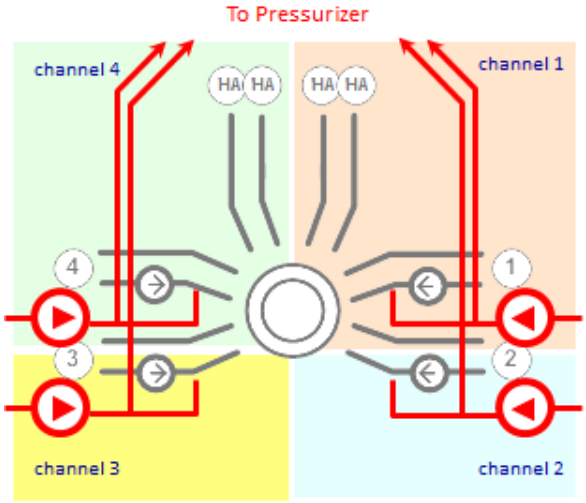


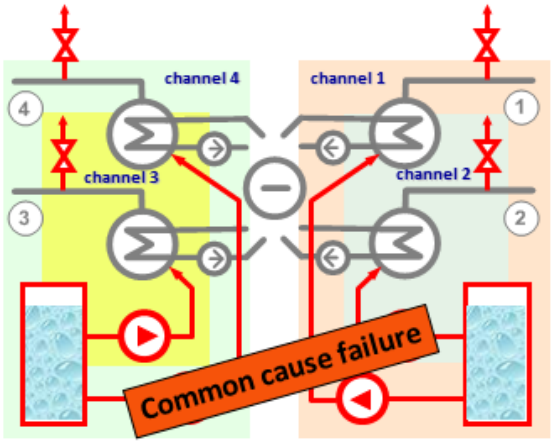
	Parameter	Value
P	Max. pressure head of HP ECCS, MPa	7,9
G	Max. flowrate of HP ECCS, m^3/h	260
P	Max. pressure head of LP ECCS, MPa	2,5
G	Max. flowrate of HP ECCS, m^3/h	900
C	Concentration of H_3BO_3 , g/kg	16

	Parameter	Value
N	Number of channel, pcs.	4
P	Nominal pressure, MPa	5,9
C	Concentration of H_3BO_3 , g/kg	16
V	Volume of tank/solution, m^3	60 / 50

FIGURE 4.2.7. DBA concept: ECCS active part and ECCS passive part

The emergency boron injection system (EBIS) consists of a four-channel system with redundancy of 4×50%. It is intended for injection of boric acid solution into the pressurizer under accidents with primary-to-secondary leaks for quick decrease in the primary pressure and for supply of boric acid solution into the primary circuit for quick bringing the reactor plant into subcritical state under the conditions with failure to scram (ATWS). The Emergency Feedwater pump (EFWP) has four channels with a redundancy of 4×50% and a tank of emergency feed water inventory. This is for feed water supply to SG in emergency conditions by the time normal feed water system and auxiliary feed water system are unavailable.





	Parameter	Value
V	Volume of tanks, m^3	80
G	Nominal flowrate, $m^3/ч$	14,5
P	Max. pressure head, MPa	24,5
C	H_3BO_3 concentration, g/kg	40

	Parameter	Value
G	Flowrate of pump, t/h	150
T	Feedwater temperature, °C	5 - 40
N	Number of tanks, pcs.	2 + 1
V	Tank volume, m^3	700
D	Possibility of setting a pressure maintenance / specific rate in cooldown mode	Available

FIGURE 4.2.8. Emergency Boron Injection System (EBIS) and Emergency Feedwater pumps (EFWP)

The passive heat removal system (PHRS) from steam generators with four-channel redundancy and 18 water-cooled heat exchangers in each channel are intended to remove the core decay heat to the ultimate heat sink through the secondary circuit under BDBAs. The passive containment heat removal system (CHRS) is used for emergency heat removal. The emergency gas removal system is for BDBA management to remove steam-gas mixture from the primary circuit, collecting in the top head of reactor, in the SG collectors and pressurizer under the conditions of coolant boiling and gassing. Finally, the core catcher is a concrete cavity used to provide support for the bottom of the reactor vessel and as a cavity protection from corium thermo-mechanical impact. The liquid and mechanical corium debris will collect here ion case of a core spillage and will be steadily cooled by the cooling water. It ensures a sub-criticality core melt in the concrete cavity and reduces radionuclides and hydrogen releases in the containment.

Several BDBAs considered in the project are listed below:

- Loss of all AC power supply sources for 8 and 24 hours
- Loss of spent fuel pool cooling for 8 and 24 hours

- Spectrum of steam line breaks inside and outside the containment up to steam line of maximum diameter with rupture of one tube in a steam generator
- Complete loss of feed water supply
- LB LOCAs with failure of low pressure ECCS active part
- SB LOCAs with failure of high pressure ECCS active part
- Long-term (up to 24 hours) loss of heat removal by the planned and emergency cooling down systems with the reactor top head removed and/or with the reactor sealed
- Primary-to-secondary coolant leak in case of multiple failures of SG tubes, or leaks over the steam generator primary side collector with equivalent diameter (D_{nom}) of 100 mm
- ATWS

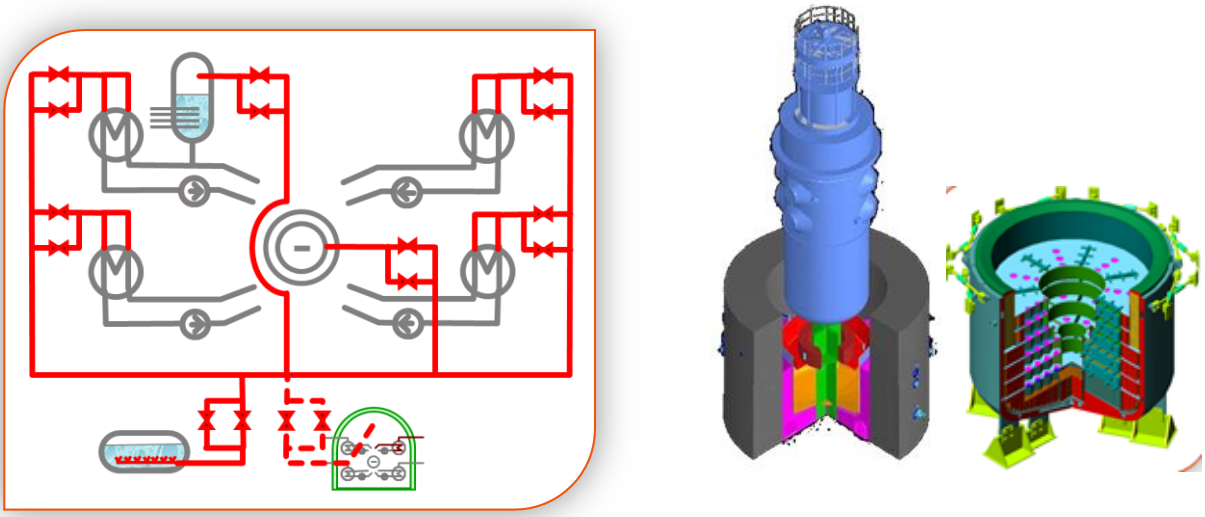


FIGURE 4.2.9. Emergency Boron Injection System (EBIS) and Emergency Gas Removal System

For numerical simulation of transient and accident conditions of VVER reactor plant at OKB "GIDROPRESS", thermal-hydraulic codes TRAP-KS and KORSAR.GP are used. These codes are applied during calculations of reactor plant safety analyses, calculations for PSA, during determination of boundary conditions for strength analysis and so on. The conditions being simulated are transients, DBA (primary-to-secondary coolant leaks, reactivity initiated accidents, loss of heat removal, change of flow rate, spurious function of systems, etc.), and BDBA (without fuel melting).

TABLE 4.2.4. Computer codes used for simulation

Simulated processes	**TRAP-KS**	**KOPSAR/GP**
Thermal-hydraulic processes in the reactor plant equipment	One-phase or two-phase flow (a drift model is used) of coolant with taking into account non-condensable gases and boric acid solution.	Two-phase mixture of water (with boric acid and non-condensable gases dissolved in it) and steam-gas phase;
Neutron-physical model	Point model or 3D model of neutron kinetics (KARTA module + library of neutron-physical constants by SAPFIR95.1 code)	Point model or 3D model of neutron kinetics (KARTA module + library of neutron-physical constants by SAPFIR95.1 code)
Coolant mixing in the reactor chambers	KAMERA – calculation of coolant mixing in the reactor pressure and collection chambers	KAMERA – calculation of coolant mixing in the reactor pressure and collection chambers
Related calculation of processes in the containment	The use of ANGAR or KUPOL codes integrated into a package	The use of KUPOL code integrated into the code
Simulation of passive safety systems	Specialized models of HA-2, QBIS, SG PHRS, SG ECBS	In the initial data file with the help of the code basic components

TABLE 4.2.5. DBA results

Initiating event	Category	Acceptance criteria							
		P_{1c} < 115 % Pdes.	P_{2c} < 115% Pdes.	K_{DNBR} > 1	Tuo_2 < 2540 °C	T_{Zr} < 1200°C	luo_2 <830 [1] J/g	< 1 % of failed fuel rods	<10 % of failed fuel rods
Increase in heat removal through the secondary side	2	17,9	8,90	1,105	1990	-	480	-	-
	3	18,1	9,05	1,12	2120	-	520	yes	-
Decrease in heat removal through the secondary side	2	19,6	9,24	1,064	1818	365	427,8	-	-
	3	18,55	8,85	1,07	1811	-	427	yes	-
Decrease in primary coolant flowrate	2	17,44	8,88	1,04	1810	-	426,1	-	-
	4	18,24	8,90	-	1810	703	426,1	-	yes
Increase in the primary coolant inventory	2	17,82	8,98	1,075	1840	358	434,4	-	-
Decrease in the primary coolant inventory	2	15,9	8,6	1,14	1811	-	427	-	-
	3	16,5	8,80	1,02	1827	352	430	yes	-
	4	16,50	8,81	-	1811/ 1834[2]	1050	427/ 470[2]	-	yes
Spurious operation of system	2	19,64	9,03	1,065	1822,8	365,2	429,8	-	-

[1] Acceptance criteria for design conditions of category 2 is 586 J/kg;
[2] Value for tvel/tveg

TABLE 4.2.6. BDBA results

Initiating event	Acceptance criteria					
	P_{1c} < 115 % Pdes.	P_{2c} < 115% Pdes.	P_{1c} < 135 % Pdes.	P_{2c} < 135% Pdes.	Tuo_2 < 2540 °C	T_{Zr} < 1200°C
Failure of all a.c. power sources for 8 and 24 hours	16,87	8,61	-	-	1430	352,5
Spectrum of steamline breaks inside and outside the containment up to maximum diameter of the steamline with rupture of one tube in steam generator	16,62	7,4	-	-	1580	351,2
Complete loss of feed water supply	18,01	8,02	-	-	1431	357
LB LOCA with failure of the ECCS active part	< beg.	7,75	-	-	1427	761
SB LOCA with failure of the ECCS active part	< beg.	7,65	-	-	< beg.	400
Primary-to-secondary coolant leak in case of multiple failures of SG tubes, or steam generator primary side collector leak with equivalent diameter Dnom 100 mm	< beg.	8,58	-	-	< beg.	< beg.
ATWS type accidents (with failure to scram)	-	-	21,09	8,82	1998	788

The computer codes are verified to:

- Local experimental facilities – MARVIKEN, EDWARDS, 7-assembly bench and DNB study bench at OKB "GIDROPRESS", Sosud (MPEI);
- Integral experimental facilities – SB-1 (OKB GP), PMK-NVH (Hungary), PACTEL (Finland), BETHSY (France), ISB-VVER, PSB-VVER (Russia), IST-EC, SOT, SPOT, GE2M-PG (Russia), LOBI
- Experiments performed at VVER-1000 Units (NV NPP, Balakovo NPP, Kozloduy NPP, Kalinin NPP, and others) and VVER-440 (LOVIISA NPP, NORD NPP, Kola NPP

- Standard tasks AER OECD/NEA, where comparison was considered with experimental data and calculation data obtained from different computer codes
- Standard tasks LBLOCA (PSB-VVER), BEMUSE, SPE-1,2&3 (PMK-NVH), ISP-27, ISP-33, ISP-50, standard tasks on reflooding at 7- and 37- rod bundles at SSC RF – IPPE.

The PSA results provide value for frequency of major release of radioactive material into the containment, defined by core melt frequency. The core melt frequency for the V-491 design is 5.94E-07 per year, which is approximately 100 times less than for a V-320 design, which is 4.88E-05 per year. It also means that the calculated frequency of major release of radioactive material into the containment will be much lower for the V-491 design.

In conclusion, the VVER reactor plants are based on technical solutions with proven reliability and efficiency for more than 40 years operation experience. The developed designs are implementing the evolution concept of VVER reactor plant improvement, oriented not only to safety assurance of units but also enhancement of their cost efficiency. A combination of active and passive systems of safety is optimized with respect to the competitive designs. The reliability of engineered safety features is improved by increasing the redundancy and application of passive heat removal systems, therefore CR 1.3.3 is met.

4.2.2.9. CR 1.3.4 Barriers

Mr Bykov (Rosatom) discussed the criterion on barriers and pointed out that in the process of safety analyses fulfilment the acceptance criteria are used to confirm physical barriers integrity during the progress of postulated initiating event (IE).

- Primary pressure and steam generator pressure will be less than 115% of the calculated values (*IE category 2,3,4*)
- Fuel pellets do not melt even locally (the temperature is less than 2,540 °C for spent fuel and less than 2,840 °C for fresh fuel (*IE category 2 and for the absolute majority of design conditions of category 3*)
- Departure from the nucleate is not reached with 95% probability for the hottest fuel element (minimum margin before DNBR in the core is >1.0) (*IE category 2*)
- The average radial enthalpy of fuel does not exceed 586 J/g (*IE category 2*) or 830 J/gUO_2 on axis of any fuel element (*IE category 3,4*)
- Special I&C should be provided to detect possible fuel loading errors during bringing to power (*IE category 3,4*)
- The number of damaged fuel elements should not exceed 1% of total number of fuel elements located in the core (*IE category 3*)
- Number of damaged fuel elements should not exceed 10% of total number of fuel elements located in the core (*IE category 4*)

Other acceptance criteria for category 3 and 4 are as follows:

- The maximum cladding temperature under accident conditions does not exceed 1,200 °C
- Local cladding oxidation depth does not exceed 17% of the original cladding thickness.
- Channels for the coolant flow inside the fuel assemblies should not be locked up to such a degree that cooling ability is broken due to swelling, destruction of claddings, as well as because of the deformation of other parts of fuel assemblies and internals
- Melting of control rods is not permitted
- Movement of control rods in the reactor should not be disrupted because of possible strains in the fuel assemblies, control rods and internals;
- Interaction between different components of fuel assembly should not lead to these components melting;
- Amount of hydrogen obtained in the fuel cladding – coolant interaction should not exceed 1% for category 3 and 10% for category 4 of that released if the entire cross section of cladding surrounding fuel pellets completely reacted with water and turned into ZrO_2 ($Zr + H_2O \rightarrow ZrO_2 + H_2$). In the analysis of the actual amount of hydrogen released, all the reactions that lead to the formation of hydrogen should be taken into account.
- Safe state of the core should be reached in such a way that conditions are created to provide maintenance of the reactor in sub-critical state, its cooldown in shutdown condition after the accident, as well as to dismantle the core and internals

Accidents with initial boundary and conditions leading to maximum possible release of mass and energy (primary coolant, secondary steam or water) into the leak are analysed for determinative design conditions. Safety analysis with the use of criteria for building structures, taking into account the change of parameters in the containment, is performed in SAR. Building structure's integrity is provided during severe accidents under conditions of core corium catcher functioning. The DBA analyses results (shown in CR1.3.3) confirm CR1.3.4 for the V-491 project.

4.2.2.10. CR 1.3.5 Controlled state

Mr Bykov (Rosatom) explained that in accordance with the Russian standards, all calculations are performed to achieve a controlled stable state. This criterion is confirmed by the results of the calculation presented in Chapter 15 of their SAR.

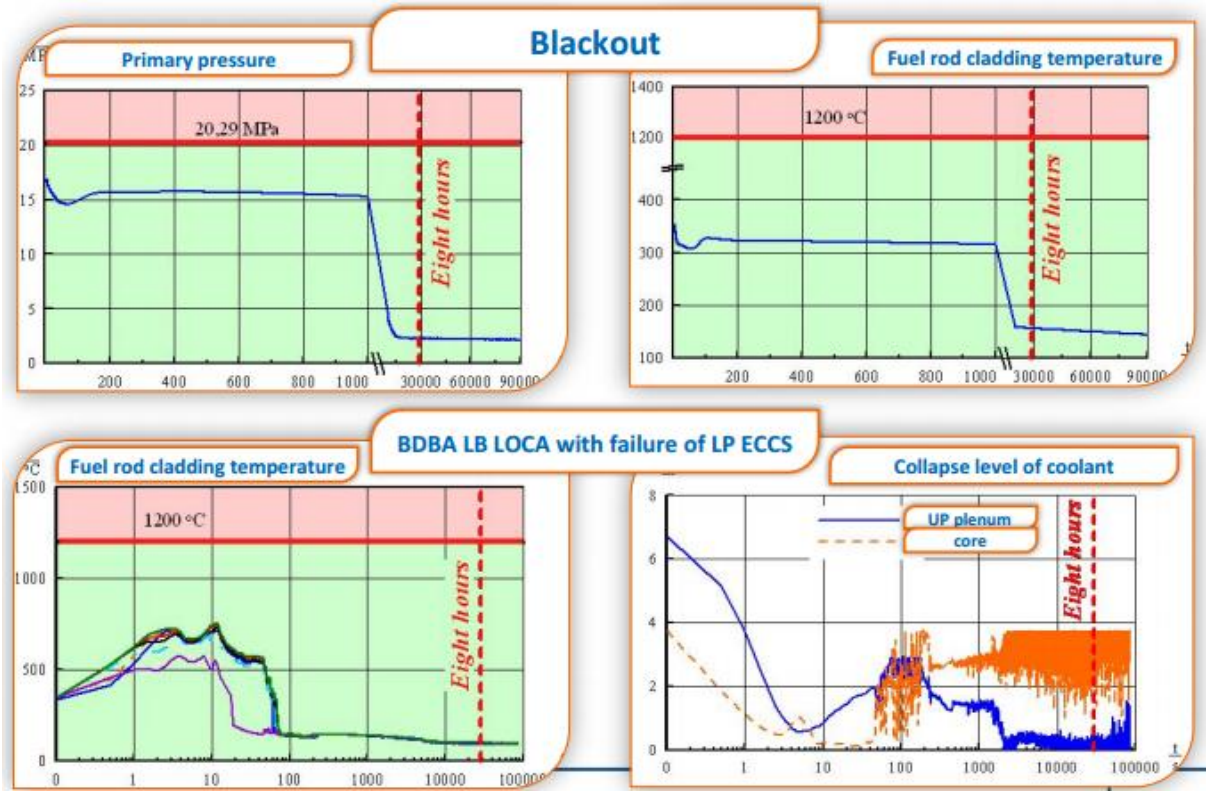


FIGURE 4.2.10. Safety analysis (Integrity of barriers)

4.2.2.11. CR 1.3.6 Sub criticality

Mr Bykov (Rosatom) explained that the AES-2006 features two systems to ensure sub-criticality: scram number of control rods - 121 and the Emergency Boron Injection System (EBIS). Compared to the V-320 design, in the V-491 design the number of control rods is increased to 121, the core re-criticality temperature decreased to 91°C and the EP efficiency without the most effective control rod is 12.5%. Safety analysis results show that xenon core poisoning after reactor shutdown leads to increased sub-criticality. Maximum of the poisoning comes in 8 hours after reactor shutdown. So CR1.3.6 is met.

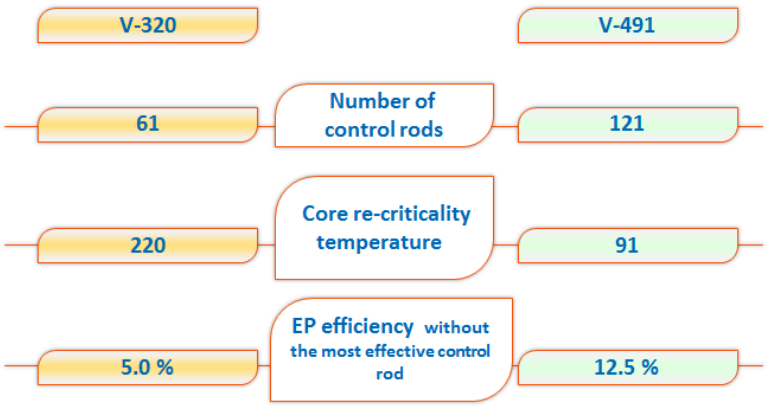


FIGURE 4.2.10. Safety analysis (Sub criticality)

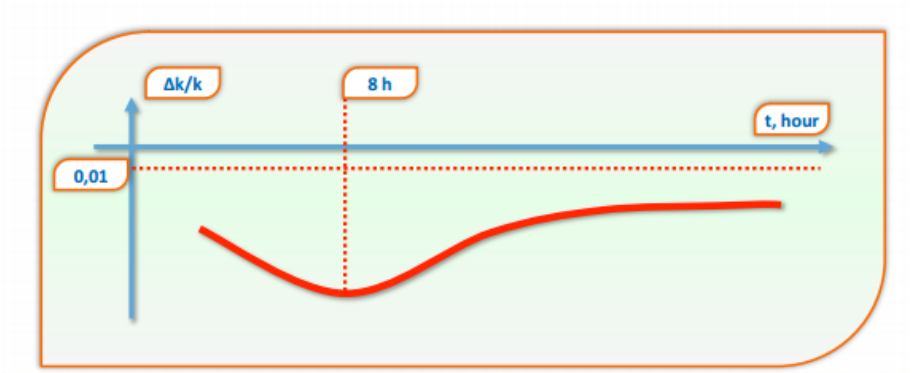


FIGURE 4.2.11. Xenon core poisoning

4.2.2.12. CR 1.4.1 Frequency of release into containment

Mr Kolchinsky (Rosatom) presented the assessment for this criterion. He pointed out that activity release into the containment under LOCAs is determined by the presence of damaged fuel cladding in the core. The following criteria are justified in the design: (i) for design condition of category 3, number of damaged fuel rods shall not exceed 1% of the total number of fuel rods in the core, and (ii) for design conditions of category 3, the number of damaged fuel rods shall not exceed 10% of the total number of fuel rods in the core. With this respect, in case of design accidents the expected annual irradiation dose of limited part of population at the protection zone boundary are effective dose below 1 mSv/event for category 3, and and below 5 mSv/event for category 4. The results of accident analyses are presented in PSAR, Chapter 15 for LAES-2 and BtAES.

LOCA accidents are managed by active safety systems to ensure reactor coolant inventory and heat removal. The safety system configuration has four-train configuration and ensures function margin as N+2 (versus N+1 in V-320 design). If the safety systems are working in the design basis mode, the acceptance criteria are implemented. Binding of volatile radionuclides in containment atmosphere is ensured by adding of chemicals (alkali) into boric water of containment spray system.

Some postulated multiple failures (including common cause failures in safety systems) have been considered in the design as DEC conditions and do not lead to severe core damage. The list of postulated DEC conditions was defined according probabilistic analysis.

The frequency of core damage is a quantitative indicator of CR1.4.1. Two main reasons which helped significantly decrease the CDF for the AES-2006 design in comparison with the V-320 design are:

- N+2 safety system
- Use of PHRS-SG for mitigation some of the safety system multiple failures consequences, such as black-out, emergency feed water failure, high pressure emergency injection failure at small LOCA, etc.

The main functions of PHRS-SG are as (i) residual heat removal and reactor shut-down cooling at full-loss-of-power conditions and at full-loss-of-feed water-supply conditions; (ii) prevention of radioactive coolant injection into atmosphere through BRU-A (steam bypass station to atmosphere) or SG safety valves during

accident with coolant leak from primary to secondary circuit; (iii) minimise radioactive coolant release due to coolant leak from primary to secondary circuit and simultaneously steam pipeline break in not cutting part out of the containment; and (iv) redundancy of safety systems in case of failure for emergency reactor cooldown conditions during accidents.

TABLE 4.2.7. Comparison of core damage frequency (CDF)

Operation Mode	CDF, 1/(r *y)		
	V-320 (BNPP)	AES-91 (TNPP)	AES-2006 (LAES-2)
Power operation	$2{,}55 \cdot 10^{-5}$	$1{,}35 \cdot 10^{-6}$	$1{,}36 \cdot 10^{-7}$
Shutdown	$2{,}33 \cdot 10^{-5}$	$3{,}50 \cdot 10^{-7}$	$4{,}58 \cdot 10^{-7}$

BNPP – Balokovskaja NPP (Ref.: PSA-1)
TNPP – Tianwan NPP in China (Ref.: PSA-1)
LAES-2 – Leningradskaja NPP-2 (Ref.: PSA-1)

In conclusion, due to new safety measures in the AES-2006 design, the CDF has been decreased to almost 10 times in comparison with the AES-91 and to 100 times in comparison with V-320 design. Therefore, CR1.4.1 is met.

4.2.2.13. CR 1.4.2 Processes

Mr Kolchinksy (Rosatom) noted that for this criterion the main aims are (i) containment integrity keeping at initial period of accident without operator actions, (ii) heat removal at a late stage of an accident, and (iii) corium cooling in the reactor vessel or core catcher. The following design basis are stated as approved in Chapter 15 of PSAR, i.e. to exclude steam explosions during interaction between water and corium, direct heating of containment, detonation of combustible mixtures and formation of non-condensing mixtures. The systems that ensure these conditions are the containment spray system, the chemical injection system for iodine binding, the hydrogen removal system and the PHRS from the containment in case spray system is not efficient.

Emergency Spray System is intended for pressure reduction in the containment in case of LOCA, removal of fission products from the containment atmosphere and control of chemical water composition in the sump-tank by adding chemicals. Passive Heat Removal System is designed for (i) reducing and maintaining pressure inside the containment within the design limits during BDBA including severe ones with core damage, (ii) removal to the ultimate heat sink the heat released into the containment during BDBA including severe ones with core damage, and (iii) redundancy of containment spray system.

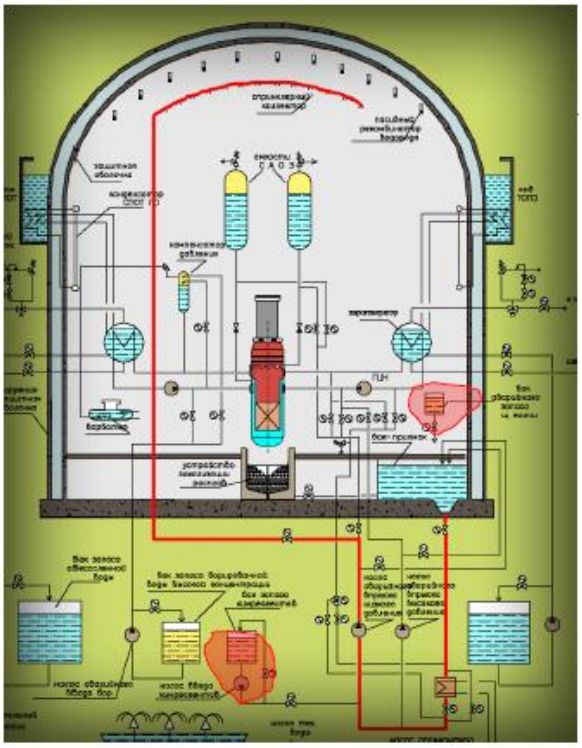


FIGURE 4.2.12. Emergency spray system

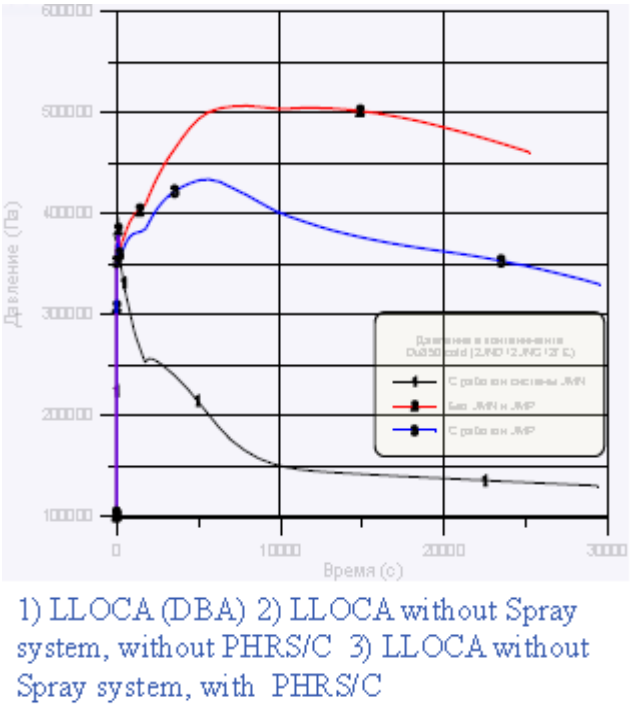


FIGURE 4.2.13. Pressure in the containment

The design basis for hydrogen safety includes to exclude detonation and deflagration in the containment in DBAs, and to exclude detonation of hydrogen in BDBA while still permitting deflagration if the localizing safety systems perform their design functions. The capacity of hydrogen removal system is designated as if 1000 kg of H_2 is generated in the containment during 5-7 hours. There are two measures provided in the design, i.e. use of passive catalytic hydrogen recombiners and employment of natural inertization of gas medium by steam.

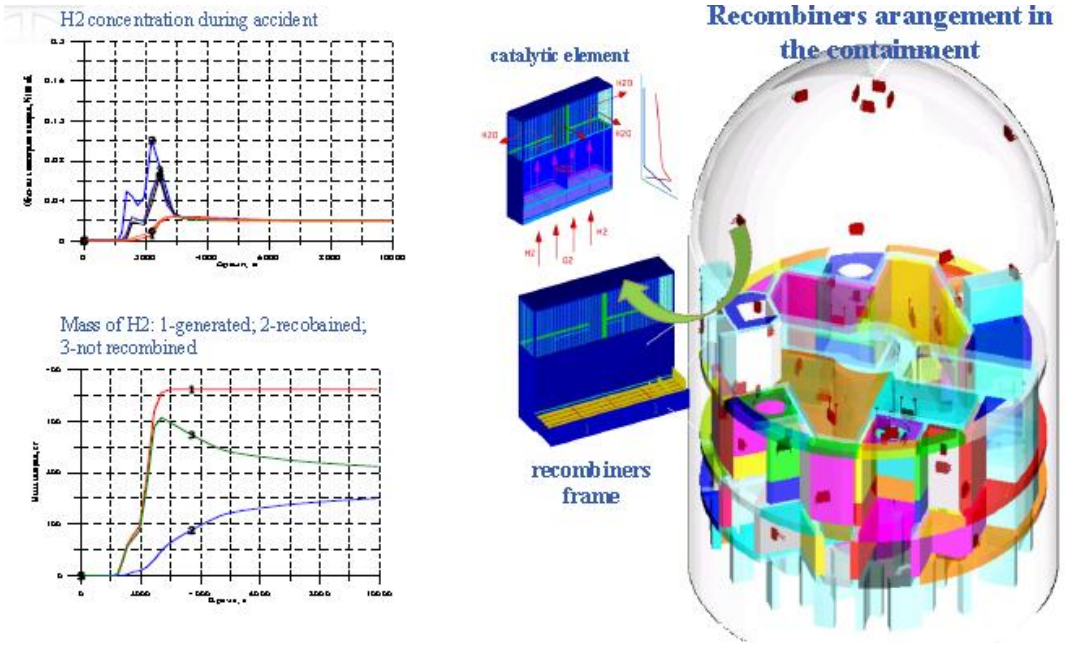


FIGURE 4.2.14. Passive catalytic hydrogen recombiners

In conclusion, the AES-2006 design has systems for controlling the system parameters and activity levels in the containment/confinement. Therefore, CR1.4.2 is met.

4.2.2.14. CR 1.4.3 Accident management

Mr Kolchinksy (Rosatom) discussed BDBA management systems implemented to satisfy this criterion. They are core catcher, hydrogen removal system (passive recombiners), system of primary circuit overpressure protection and emergency gas removal system, PHRS-SG, PHRS from containment, core catcher water supply system, emergency chemical agents supply system, special measures for fuel pool and emergency heat removal tanks make-up, special electrical train with moving diesel generator and accumulators, and special I&C equipment and control panel in the main control room (MCR) (Severe Accident Management Simulation Module (SAMS)).

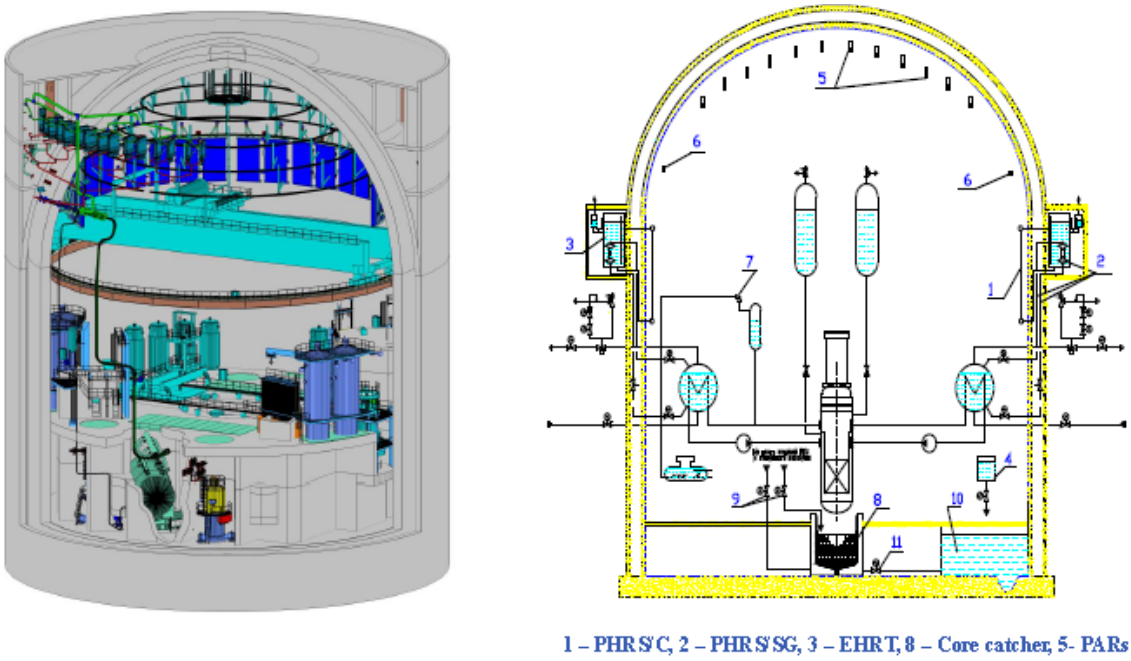
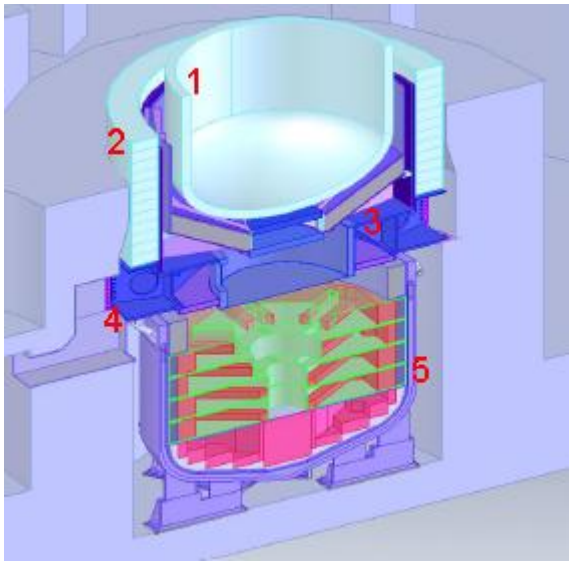
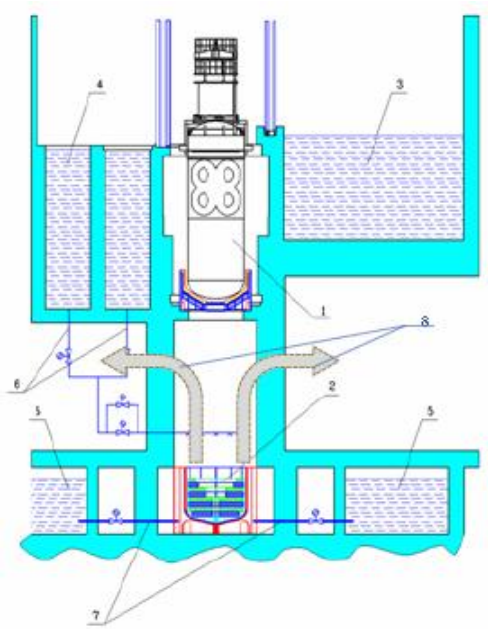


FIGURE 4.2.15. Means for BDBA management

The core catcher functions as a reactor vault protected against corium thermomechanical interaction, reception and localization of solid and liquid corium components, heat transfer from corium, to ensure core sub-criticality, and to decrease hydrogen and radionuclides release into the containment. The core catcher is dry, water-cooled and crucible type located in the sub-reactor space. It is made of a mixture of iron and aluminium oxides, steel and oxide as part of sacrificial materials. Two-layer core catcher vessel has resistance to thermal stresses, and heat shield to protect the upper part of the vessel against thermal radiation. Gadolinium oxide is used as part of the sacrificial material to ensure sub criticality of molten core. The core catcher does not require periodic service works, and the equipment and the measurement system testing should be carried out after accidents, which are accompanied by leakage in the containment.



1-reactor vessel; 2-dry reactor protection; 3-console framework; 4-service area; 5-core catcher vessel



1-reactor; 2-core catcher; 3-fuel pool; 4-reactor internals inspection vault; 5-pit-tanks; 6-core catcher flooding pipes (water supply to the corium surface), 7-core catcher heat exchanger feeding pipelines, 8-steam removal (pipes).

FIGURE 4.2.16. Core catcher: filling and cooling of core catcher

The basic codes for justification of passive systems:
- KORSAR-B1- best-estimate code a fully non-equilibrium two-fluid model of coolant flow;
- SOKRAT/V1 - best-estimate code for modelling severe accident conditions;
- KUPOL-M - containment code (lumped parameters);
- ANSYS SFX, STAR-CD, PGS-TK, LOGOS, FIRECON special CFD codes for containment calculation;
- Program unit KORSAR – KUPOL, SOKRAT-KUPOL integrate codes for joint calculations RV & containment;
- DANKO, ANSYS - strength calculations of building constructions, equipment
- FIRECON, FIRECON- Calculation of loads on hydrogen combustion, determination of possible combustion modes

Experimental validation of passive systems was performed using:
- Large-scale test rig “SPOT” for HRS/SG, CKTI, St. Petersburg
- Test rig “PHRS/C” in OKBM, Nizhniy Novgorod
- Large-scale test rig “KMS” in NITI, Sosnovyi Bor



FIGURE 4.2.17. Test rigs to validate passive systems

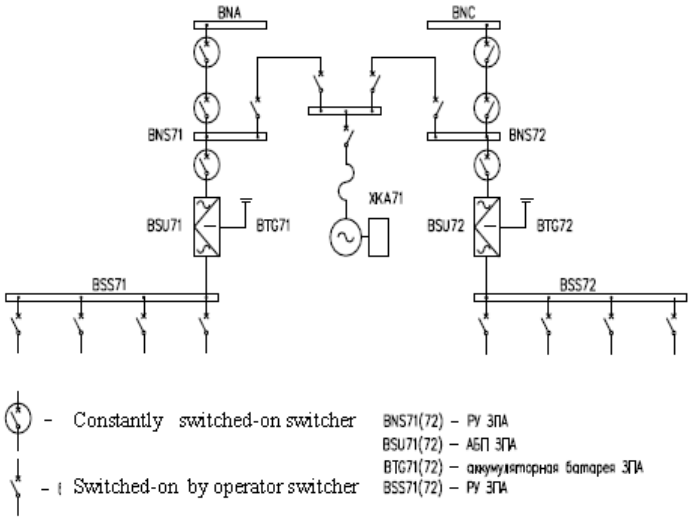


FIGURE 4.2.18. Additional power supply channel from mobile diesel generator

Regarding Information and Measurement System, the special control panels for severe accident management (SAM) are located at the Main and Emergency Control Rooms. In case of severe accident at ex-vessel stage, temperature in core catcher and cooling water level around the core catcher vessel would be monitored at the SAMS panel in the MCR. Rapid temperature increase (>25 °C/s) registered by sensors in core catcher is the criterion of reactor vessel brake down and corium dropping into the core catcher.

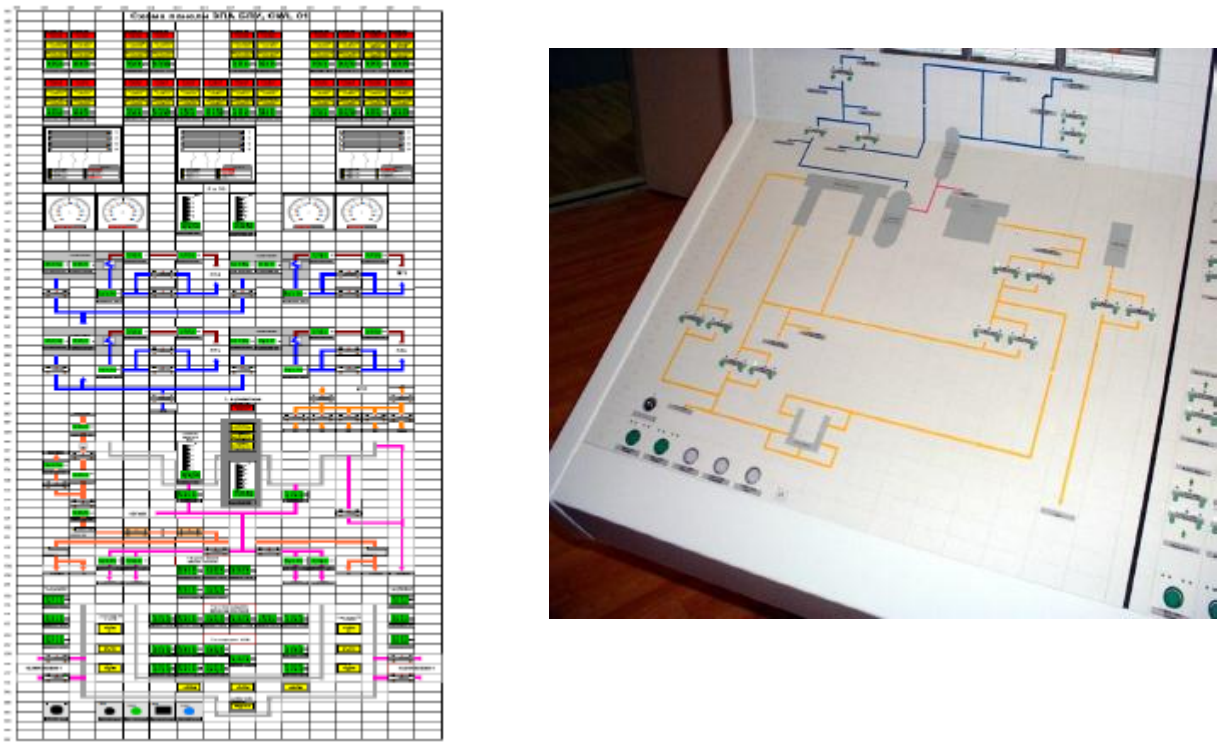


FIGURE 4.2.19. Information and measurement system

Rosatom has developed documents on BDBA management instruction to be supplied to the NPP owner:

- Emergency Operating Procedures (EOPs) as guidance for managing DBAs and BDBAs (before core damage occurs) for core damage prevention
- Severe Accident Management Guidelines (SAMG) that propose a range of possible mitigating actions and allow additional evaluation and choose alternative actions.

In conclusion, the AES-2006 design contains the special measures, equipment and procedures for BDBA (including severe accidents) management missing in the V-320 design. Therefore, CR1.4.3 is met.

4.2.2.15. CR 1.6.1 Independence of DID levels

Mr Kolchinsky (Rosatom) pointed out the criterion was not yet performed in the Belarus report. The DID concept is based on using a system of sequential physical barriers for spreading ionizing radiation and radioactive substances into the environment as well as the system of organisational and technical measures on protection of the barriers keeping their efficiency, as well as on protection of personnel, population and the environment.

The physical barriers for V-320 and AES-2006 are the same, which are fuel matrix (pellets), fuel rod cladding, reactor coolant system boundary, primary containment. For the organization and technical measures Level 1 and 2 are the same for both designs, but Level 3 (control or accidents within the design basis) and Level 4 (severe accident management) are new, with stricter safety requirements from WENRA. In the 2013 WENRA's DID requirement, Level 3 is subdivided into Level 3a which includes postulated single initiating events, and Level 3b which includes postulated multiple failure events.

The fundamental safety functions are aimed at fulfilling the tasks of physical barrier protection from radioactive substance propagation. These include reactivity monitoring and control, reliable heat removal from the core using a coolant inventory and a leak-tight primary circuit, and the ability to localise and reliably retain radioactive fission products.

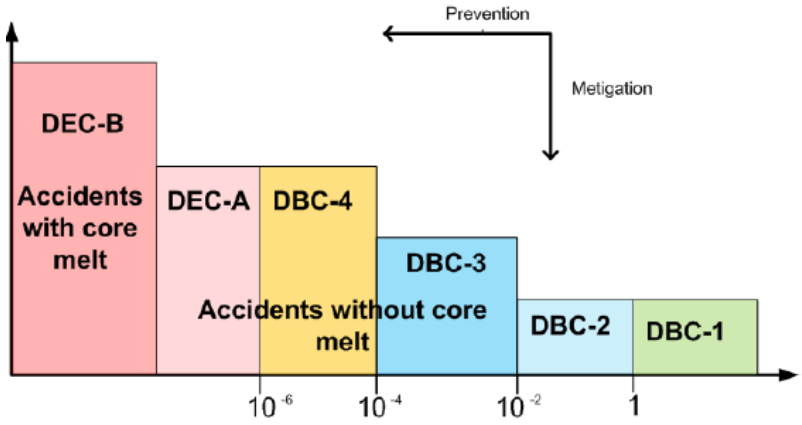


FIGURE 4.2.20. DBC and DEC definition

Generally independence justification analysis is performed using deterministic approach (including hazard analysis) and then technical solutions are verified by probabilistic method. In each DID level, the special set of SSC for implementation of all necessary safety function is provided in the design. The SSC should be independent of each other, and each set of SSC includes mechanical SSC, I&C, electrical sources, etc.

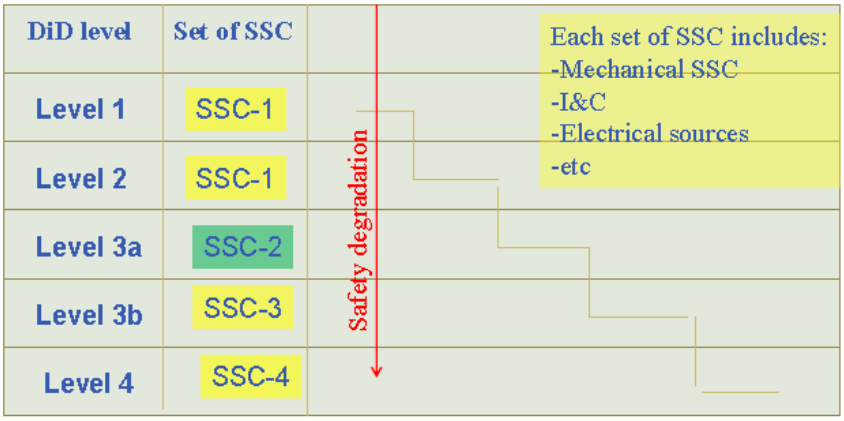


FIGURE 4.2.21. Deterministic approach

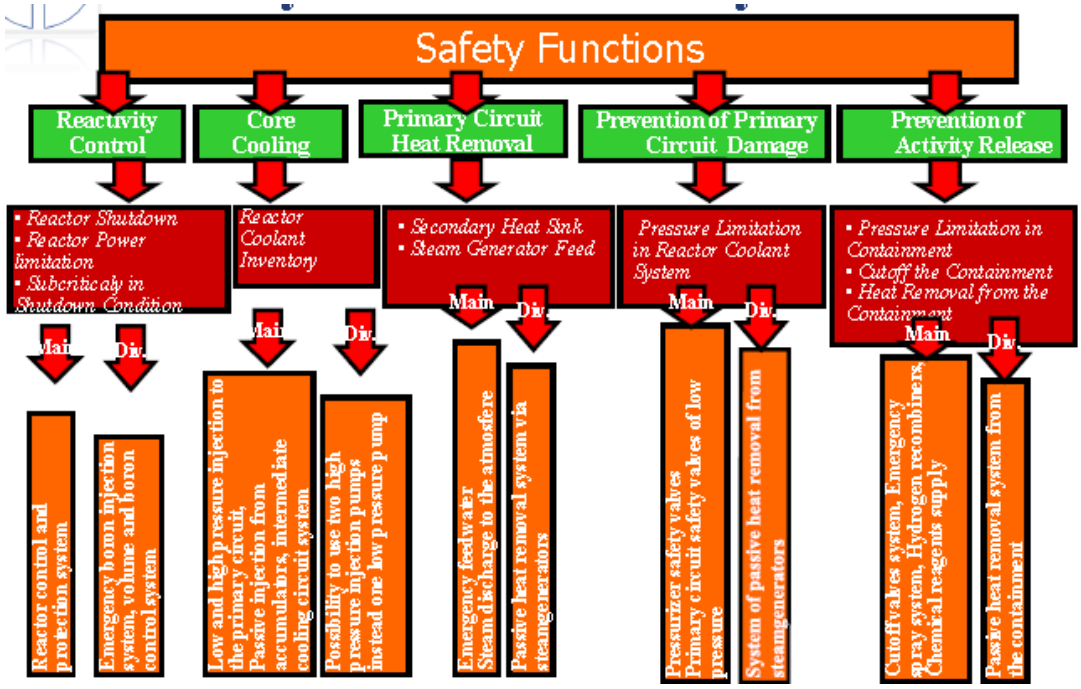


FIGURE 4.2.22. Safety function diversity

The following tables shows sample of analysis approach of SSC for DID Level 1-4, for reactivity control, heat removal from reactor core, limitation of fission products release and supporting system. The colours show the safety class of SSC. The same table is developed for each initial event. The safety function diversity in DBA and DEC with multiple failures is presented in this type of diagram. As shown, all the safety functions have two diversity ways to implement, e.g. for I&C also justified with different suppliers/manufacturers and types of equipment.

TABLE 4.2.8. Analysis approach for Level 1 and 2

	Safety levels	Level 1			Level 2		
	Level name	Control under NO and Prevention of AOO			Prevention of DBA		
	Technical Features	Technical features of normal operation			Technical features of normal operation		
	Канал электроснабжения технологической системы	5 / 6	1 / 2 / 3 / 4	Internal self-protection	5 / 6	1 / 2 / 3 / 4	Passive principle
Reactivity control	Power limitation and/or reactor shutdown	APC, GICS	PP	Feedback	APC, PLC, PP, SPP	PP	Drop of the CPS absorbing rods into the core under gravitation
	Control of self-supporting fission chain reaction	APC, GICS, KBA	N/A	N/A	APC	N/A	
	Subcriticality assurance	KBA, KBC equipment	N/A	N/A	KBA, KBC equipment	N/A	N/A
Heat removal from reactor core	Coolant reserve maintenance at primary circuit leakiness	KBA	N/A	Considerable volume of the coolant in PRZR and PCP	KBA	N/A	N/A
	Pressure maintenance in the primary circuit	TEH, controller of the injection into PRZR	N/A	Design solutions of PRZR	TEH, controller of the injection into PRZR, quick-acting injection valves	N/A	N/A
	Provision of heat removal by the primary circuit systems	RCPS-SG-Reactor	Under cooldown: ECCS LP, heat exchangers of ECCS	N/A	RCPS-SG-Reactor (train 5/6) Under cooldown: ECCS LP, heat exchangers of ECCS (train 1/2/3/4)	Under cooldown: ECCS LP, heat exchangers of ECCS	Continuous inertial rundown of RCPS under the action of special rotating masses
	Pressure maintenance in the secondary circuit	TG CV	N/A	Design solutions of SG	BRU-K	N/A	N/A
	Provision of heat removal by the secondary circuit systems	TG, BRU-K,-SN,-D SEFP, EFP, CP, RP	N/A	Considerable volume of SG boiler water	BRU-K,-SN,-D SEFP, EFP, CP, RP	N/A	N/A

Limitation of fission products release	Limitation of pressure inside the containment	NO ventilation	N/A	Design solutions, safety margins	NO ventilation	N/A	N/A
	Localization inside the containment	NO ventilation	N/A	Design solutions, safety margins	NO ventilation	N/A	N/A
	Localization in the annular space	HVAC (Ventilation in the annular space) (NO)	N/A	Design solutions, safety margins	HVAC (Ventilation in the annular space) (NO)	N/A	N/A
	Localization in SG	N/A	N/A	Design solutions, safety margins	N/A	N/A	N/A
	Localization in auxiliary systems	N/A	N/A	Design solutions, safety margins	N/A	N/A	N/A
Supporting systems	I&C system	NO I&C	NO I&C	N/A	NO I&C	NO I&C	N/A
	Power supply	NO power supply	NO power supply	N/A	NO power supply, RGM, ASB, unit SB	Emergency DG	N/A
	HWAC	HWAC for NO	HWAC for NO	N/A	HWAC for NO	HWAC for NO	N/A
	Cooling water	PGB	KAA, PEB	N/A	PGB	KAA, PEB	N/A

Safety class of technical features (according Russian standards)	2
	3
	4
Conservative and passive means	

TABLE. Analysis approach for Level 3a, 3b and 4

	Level 3a	Level 3b (DEC-A)								Level 4 (DEC-B)	
	Control of DBA and prevention BDBA	BDBA management without core melting								Severe Accidents	
	Safety systems	I&C Common Cause Failure				Mechanical Features Common Cause Failure					
	1/2/3/4	5/6	1/2/3/4	7/8	Passive principle	5/6	1/2/3/4	7/8	Passive principle	7/8	Passive principle
Reactivity control	RPS	N/A	JDH(ATWS)	N/A	Drop of the CPS absorbing rods into the core under gravitation	GICS	JDH(ATWS)	N/A	Drop of the CPS absorbing rods into the core under gravitation	N/A	Core melt is subcritical
	ECCS HP,ECCS LP, heat exchangers of ECCS	N/A	ECCS HP	N/A			JDH(ATWS)				
	Elimination of boric delution (cut-off valves)	N/A	Elimination of boric delution (cut-off valves)	N/A	AT	KBA and KBC equipment	JDH(ATWS)	N/A	AT	N/A	Properties of the core catcher sacrificial material
Heat removal from reactor core	ECCS HP, AT, ECCS LP	N/A	ECCS HP	N/A	AT	KBA	JDH	N/A	AT	N/A	N/A
	PPOSV	N/A	N/A	N/A	PPOSV use of spring	Controller of the injection into PRZR, quick-acting injection valves	N/A	N/A	PPOSV use of spring	N/A	N/A
	ECCS LP, heat exchangers of ECCS	N/A	N/A	N/A	Continuous inertial rundown of RCPS under the action of special rotating masses	N/A	N/A	N/A (if required Feed and bleed through additional control line of PPOSV)	Continuous inertial rundown of RCPS under the action of special rotating masses	SG PHRS, (provision <1MIIa)	Core catcher cooling with water from inspection shafts, fuse valve
	BRU-A, SG POSV	N/A	N/A	N/A	SG POSV (passive, use of spring)	BRU-K	N/A	N/A	SG POSV (passive, use of spring)	N/A	SG POSV (passive, use of spring)
	EEFP, BRU-A	N/A	N/A	SG PHRS, makeup pump of PHRS and fuel pool	SG POSV (passive, use of spring)	BRU-K,-SN,-D SEFP, EFP, CP, RP, Technical Condencer	N/A	SG PHRS, makeup pump of PHRS and fuel pool	SG POSV (passive, use of spring)	SG PHRS	N/A

Limitation of fission products release	JMN (spray system)	N/A	N/A	Make up pump of PHRS and fuel pool	Containment vessel PHRS, passive autocatalytic hydrogen recombiner	N/A	N/A	Make up pump of PHRS and fuel pool	Containment vessel PHRS, passive autocatalytic hydrogen recombiner	Make up pump of PHRS and fuel pool	Containment vessel PHRS, passive autocatalytic hydrogen recombiner
	Localization of SV JMN (spray system-chemicals supply into SV)	N/A	N/A	N/A	Containment	N/A	N/A	N/A	Containment	N/A	Containment, core catcher
	KLC (Ventilation in the annular space) KLG (isolation of ventilation in the annular space from ventilation systems)	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
	MSIV, localizing valves of SG blowdown, valves of feedwater, emergency feedwater, isolation gate valve upstream BRU-A) JDH on injection in PRZR - during primary-to-secondary leakage	N/A	MSIV, localizing valves of SG blowdown, valves of feedwater, emergency feedwater, isolation gate valve upstream BRU-A) JDH on injection in PRZR - during primary-to-secondary leakage	N/A	N/A	N/A	Motor-operated gate valve in steam line	N/A	N/A	N/A	N/A
	Localizing valves on auxiliary systems	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Supporting systems	ESFAS	N/A	Hard Wired Deversity I&C		N/A	NOI&C	ESFAS	I&C for BDBA	N/A	I&C for BDBA	N/A
	Emergency DG, emergency SB	N/A	Emergency DG, emergency SB	SB for BDBA, movable DG	N/A	Unit SB	emergency SB	SB for BDBA, movable DG	N/A	SB for BDBA, movable DG	N/A
	Ventilation in Safety building	N/A	Ventilation in Safety building	N/A	N/A	HWAC for NO	N/A	HWAC for BDBA	N/A	N/A	N/A
	KAA, PEB	N/A	KAA, PEB	N/A	N/A	PGB	N/A	N/A	N/A	N/A	N/A

Safety class of technical features (according Russian standards)	2 3 4
Conservative and passive means	

The probabilistic assessment is shown below to choose the events and measures on each DID level. The CDF for AES-2006 is 5.8E-7/RY, which is less than the IAEA requirement of 1.0E-5/RY. Similarly the LRF is 3.7E-9/RY for the AES-2006 design in comparison with 1.0E-7/RY in the IAEA requirement.

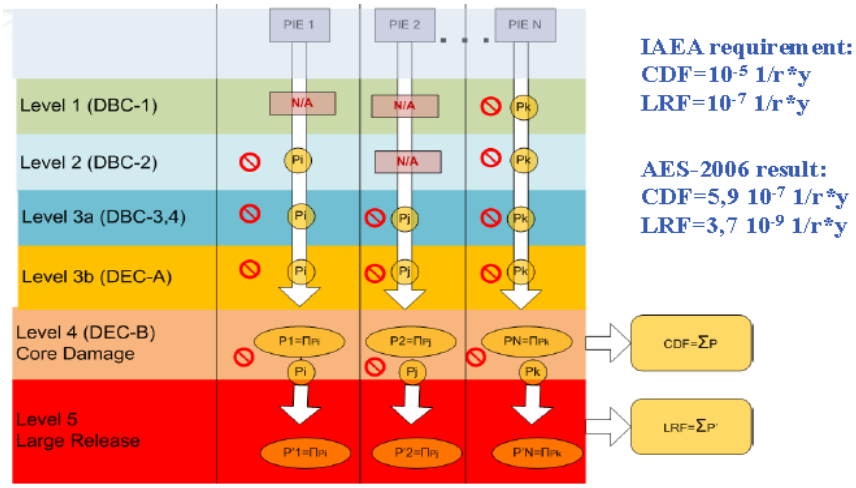


FIGURE 4.2.23. Probabilistic assessment

According to the probabilistic assessment, there is a list of external impacts and common cause failures.

TABLE 4.2.9. Examples of common cause failure events postulated in the AES-2006 design

Detonation	Postulated Initiating Event	Loss of a safety system
ATWS	Anticipated transient	Fast shutdown
Station blackout	Loss of off-site power	Emergency power supply
Total failure of all computer I&C systems	Loss of normal operation I&C	ESF Actuation System (ESFAS)
Total loss of feed water	Loss of main feed water	Emergency feed water supply
CCF at LOCA	Small LOCA	High pressure emergency injection system
“Heavy” commercial aircraft crush	Extremely external impact	Dismantling of safety building and loss of all safety systems in it

In conclusion, the assessment in accordance with the INPRO methodology has not been implemented for the AES-2006 design given the much time and resources required but all the necessary information is contained in the design document and analyses implemented by other methods. So CR1.6.1 is met.

4.2.3. Availability of References

There is limited design information available in the open sources, e.g. the V-491 design in the IAEA ARIS system. All design information can only be provided to assessor / company through direct contact requiring a special procedure. Rosatom had the opinion that technology holder / designer should be involved in a NESA.

The following references to the particular assessment provided in the presentation were given:
- PSA and PSAR for Leningradskaja NPP-2 (LAES-2)
- Baltic NPP (BtAES) in Kaliningrat Region
- Belorusskaja NPP in Belarus
- Balakov NPP (V-320 design) (*Open information*)

4.2.4. Discussion

Questions	Answers
Mr Gevorgyan (Armenia) indicated that many countries need SMR, and VVER-640 is completed and will be constructed in Leningrad. He asked why St Petersburg Institute ATOMENERGOPROJECT (SPbAEP) was not suggested for user countries.	**Mr Kolchinsky (Rosatom)** reasoned that no customer was interested in the design so it was put in archive. After 10 years in archive, the requirements have changed and the design needs to be modernized. All solutions of the reactor are modern and actual, i.e. all research work and experience used in the new designs. He mentioned a new activity has already started to recover the design, working together with general constructor of NPP and Kurchatov Institutes, though at not very fast pace. **Mr Bykov (Rosatom)** added that Armenia can send a note to Rosatom to provide medium sized PWR because there are several preliminary designs/projects on going.
Mr Koshy (IAEA) asked to elaborate how criticality is avoided in the core catcher. In the design, it is expected to be dry and cooled externally. In a severe accident while cooling down the core, how it can be kept dry and at the same time avoid criticality. He later asked in case assuming feeding is stopped so that moderator does not come anymore, whether there is experimental to validate it.	**Mr Kolchinsky (Rosatom)** explained there are two ways to exclude criticality in the core catcher. Firstly, using gadolinium fraction in the core catcher materials, and secondly the property of the materials in the core catcher dilutes / fractionates core catcher debris and it is impossible to gather critical mass. There was special experiment in the research institute but on a small scale.
Mr Goncalves Filho (Brazil) asked about the material of the core catcher and maximum temperature.	**Mr Kolchinsky (Rosatom)** responded that the vessel is made of carbon steel. The maximum local temperature of the corium can be above the melting temperature of the core, i.e. above ~2,500-2,800 °C, but the mixture is colder and the temperature close to wall is about 1000 °C.

4.3. ABWR (Advanced Boiling Water Reactor)

Mr James A Beard and Mr Kenichi Yasuda (GE Hitachi Nuclear Energy International) presented INPRO assessment in the area of safety of the Advanced Boiling Water Reactor (ABWR) design.

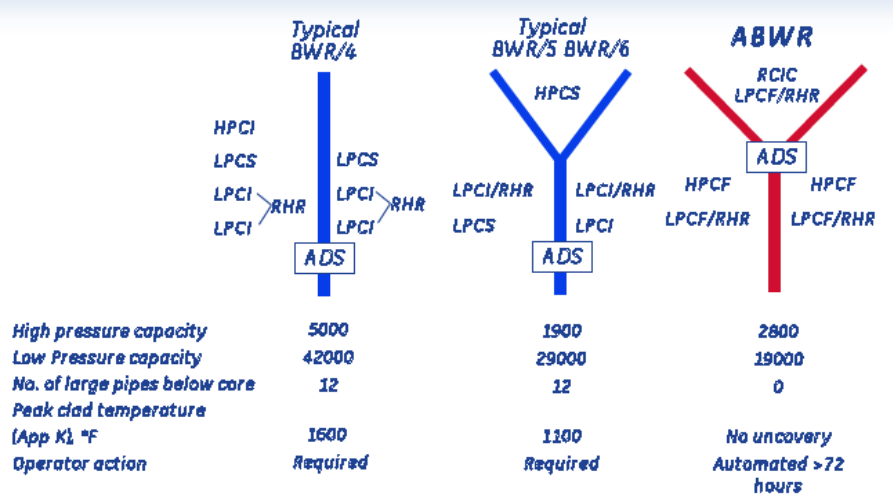
4.3.1. Overview of ABWR Design

Mr Beard (GE Hitachi NE Intl.) presented the overview of the ABWR which is a Gen III reactor with the lowest core damage frequency of its class. This reactor is a successor of the Dresden 2 reactor (Mark III) and

has a thermal power output of 3,926 MW and an electric output of 1,360 MW. The company seeks to simplify the design in order to enhance safety. One significant improvement in the ABWR is the adoption of internal recirculation pumps for reactor recirculation, i.e. eliminating external recirculation loop with lots of large piping in the previous design that could results in possibility of a LOCA in which water will be drained water out of the reactor pressure vessel. This improves the reactor response to a LOCA. The active safety systems offer a 72 hour fully automated capability without operator action to maintain the reactor under safe shutdown state. The operating cycle length without refuelling is of 12 to 24 months. The design also features power maneuverability or load-following capability, i.e. ±10% within 10 seconds at a rate of 5% per minute afterward within the range of 50-100% power, which can be done 90% of the operational cycle.

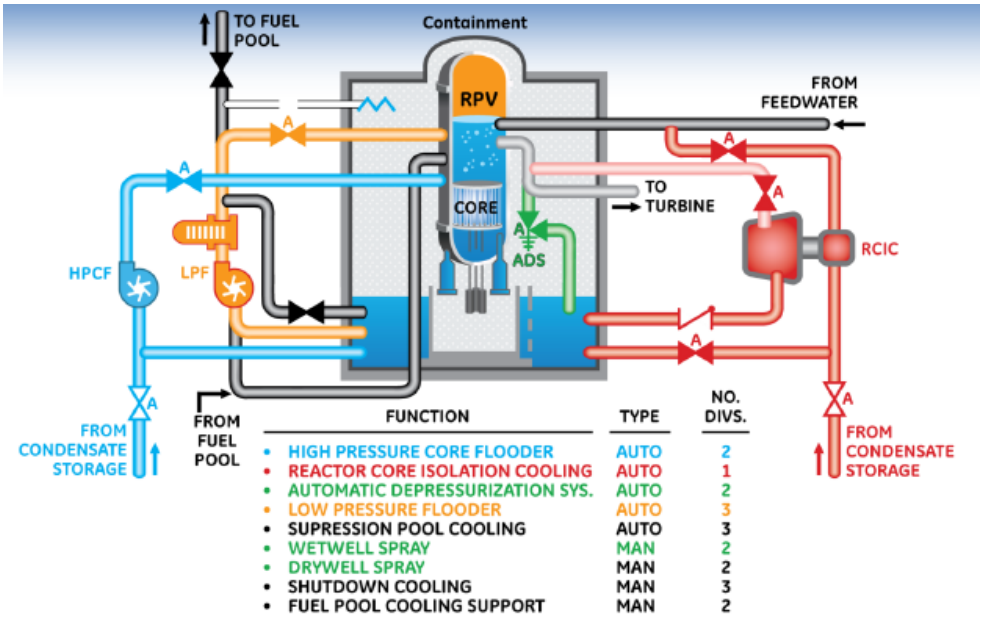
Mr Beard (GE Hitachi NE Intl.) also discussed evolution of the containment, in which now the ABWR uses the so-called *pressure suppression component* to absorb the energy following DBA, LOCA and also act as a source of water of ECCS. The 50-year old technology was compared as equivalent with the PWR newest adoption of the In-Containment Refueling Water Storage Tank (IRWST). He addressed a question on chasing safety versus optimizing design. Safety in the design of a NPP should be optimised by considering maintaining a reasonable cost of design. For example, in selecting 3×100% or 4×100% divisions, i.e. N+1 and N+2, and how GE Hitachi NE Intl. managed to find an optimable mix of N+2 but with 3×100% divisions. Referring to the issue of margins, chasing margins will complicate the operation of the plant and that at certain point return on investment or return on increasing margins will no longer improve the safety relative to the cost input. First example is natural driving head with passive plan, which is not suitable for seismic condition. The large heavy volume of water at elevated conditions presents a seismic issue, which requires reinforcement steel. Secondly, for the 24 month fuel cycle, there is a trade-off between cold shutdown margins versus loading additional fissile materials.

For INPRO assessment in the area of safety, the ABWR design is compared with the BWR5/BWR6 as the reference plant. The evolution of the ECCS is shown below. The BWR5 and BWR6 is considered to have 2.5 train configuration (HP + 2 LP) while the ABWR has 3 trains, with each HP and LP has 100% train. For the peak clad temperature, for every DBA the core is kept submerged under water so there is no fuel or cladding heat up. The elements of ECCS comprise one reactor core isolation cooling (RCIC), 2 high pressure core flooders (HPCF) with motor driven emergency core cooling pumps, and 3 low pressure core flooders (LPCF) as part of the residual heat removal (RHR) train. As described, each of the three divisions is capable from the high pressure and low pressure perspective. To enable low pressure injection when there is not large enough break and the HP system cannot make up for the loss, there is also automatic depressurization system (ADS) that can be actuated to depressurize the vessel. This allows the LP injection system to ensure the core is covered with water.



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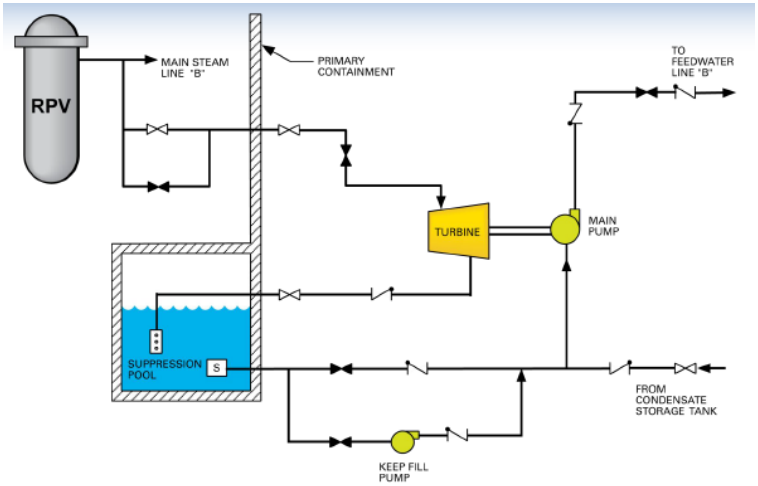
FIGURE 4.3.1. The evolution of the ABWR emergency core cooling system (ECCS)



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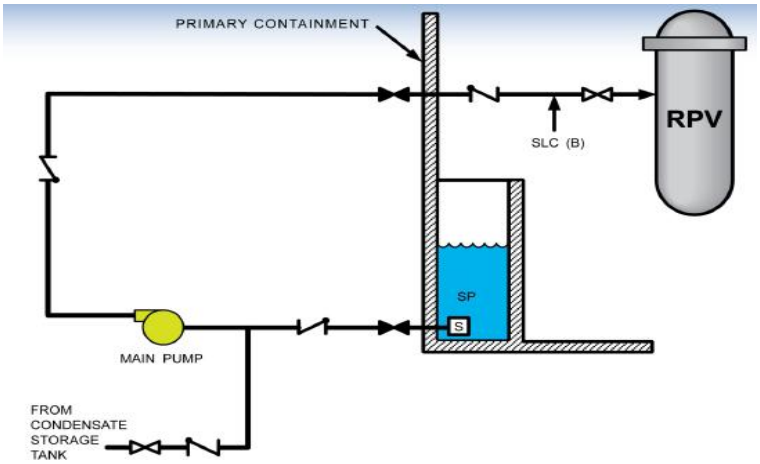
FIGURE 4.3.2. ABWR Emergency core cooling system (ECCS)

- The RCIC is considered as a very significant system in the ABWR and it relies on diverse energy sources. One lesson learned from Fukushima is that AC power is not always available. The RCIC makes up core cooling under high pressure condition, by simply taking the steam generated from the decay heat in the RPV, passing it through the turbine and exhausting the steam into the suppression pool. This will spin the turbine and drive the pump, and suck the water from the condensate storage tank or from suppression pool, and pump back into the RPV. The two units in Fukushima had this type of pumps, but the design has been enhanced with water-cooled and lubricated turbines so the performance will be much better.
- The HPCF can draw water from either condensate storage tank (higher water quality) or suppression pool (lower water quality). If response to transient, injection of water from condensate storage tank to the RPV is preferred. The suppression pool is credited as safe-related source of cooling water for ECCS situation.
- The RHR combines several different functions, altogether there are 6 modes of operation, in which 3 are safety related modes and 3 are non-safety related modes.
- The ADS is used to bring pressure down to containment pressure. The steam is piped from the safety relief valve down to the quencher in the suppression pool where the steam is condensed, thus depressurizing the pressure.



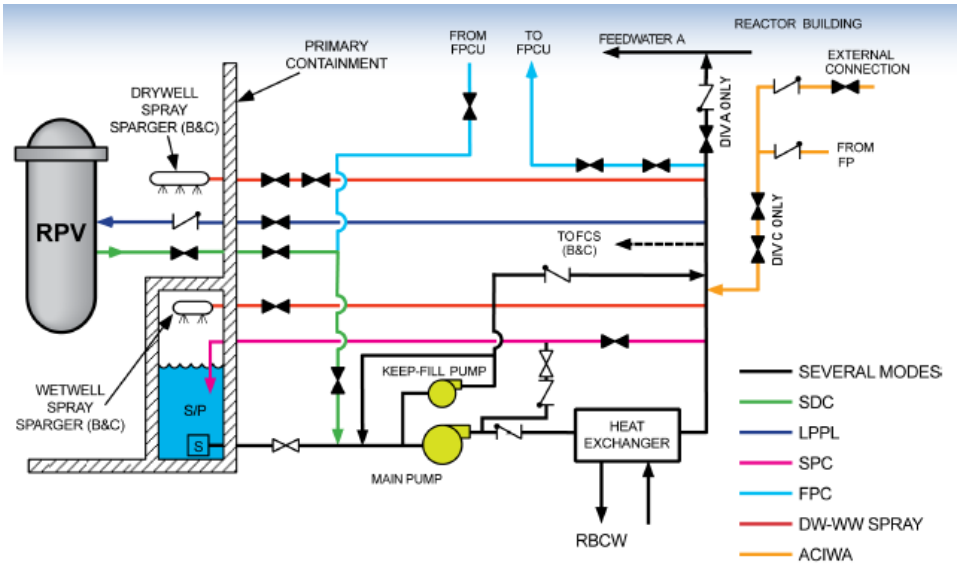
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FIGURE 4.3.3. Schematic diagram of ABWR reactor core isolation cooling (RCIC)



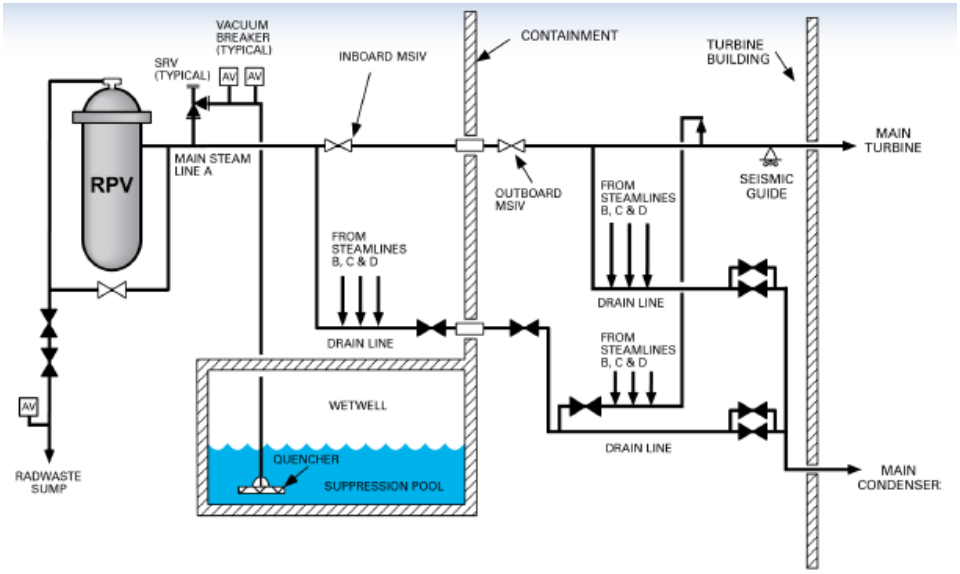
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FIGURE 4.3.4. Schematic diagram of ABWR high pressure core flooder (HPCF)



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FIGURE 4.3.5. Schematic diagram of ABWR residual heat removal (RHR)

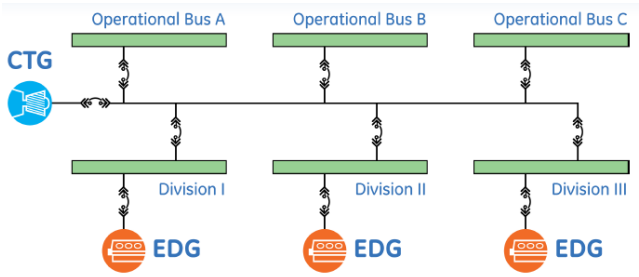


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FIGURE 4.3.6. Schematic diagram of ABWR automatic depressurization system (ADS)

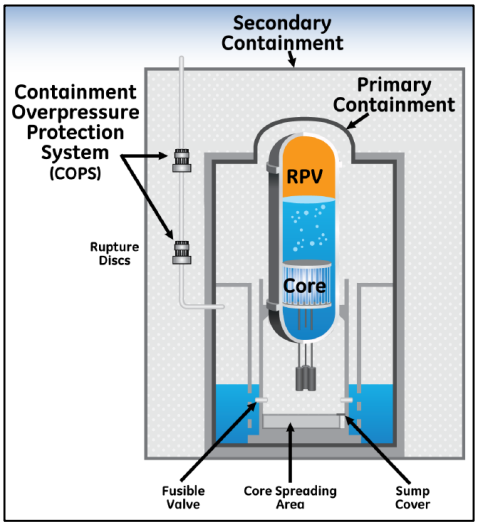
Another key design feature of ABWR is the alternate AC plan, although there is AC independent means of making up coolant to the RPV using the RCIC system. The reactor uses the air-cooled combustion turbine generator (CTG) that will automatically start during loss of off-site power. But it relies on the operator to make the necessary connections or close the necessary breakers.

The ABWR passive features for mitigating severe accidents include: inerted containment, lower drywell flood capability, lower drywell special concrete and sump protection, a suppression pool for fission product scrubbing and retention and containment overpressure protection system (COPS). The passive autocatalytic recombiners (PARS) are under consideration.



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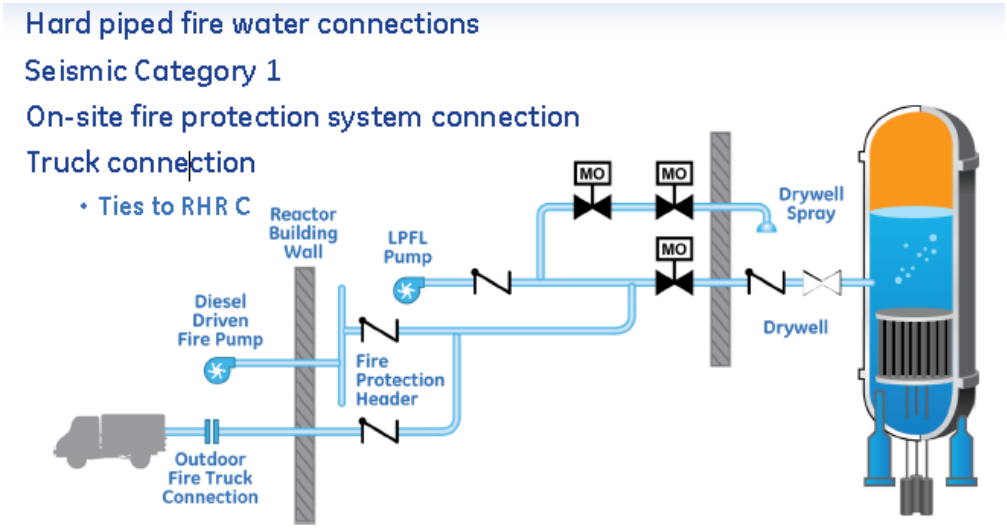
FIGURE 4.3.7. Onsite AC Power with alternate AC plan supplied by air-cooled CTG



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FIGURE 4.3.8. ABWR severe accident features

One of those is the AC independent water addition (ACIWA), which is a means of taking fire water and pump it to various locations within the primary containment or RPV using multiple sources. It has the category 1 seismic rating, comprises hard piped fire water connections and on-site fire protection system connection capabilities as well as truck connection.



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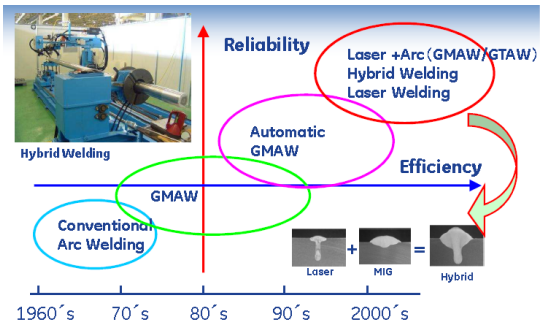
FIGURE 4.3.9. AC independent water addition (ACIWA)

4.3.2. INPRO assessment in the area of safety of ABWR

4.3.2.1. EP 1.1.1.3 Quality of manufacture and construction

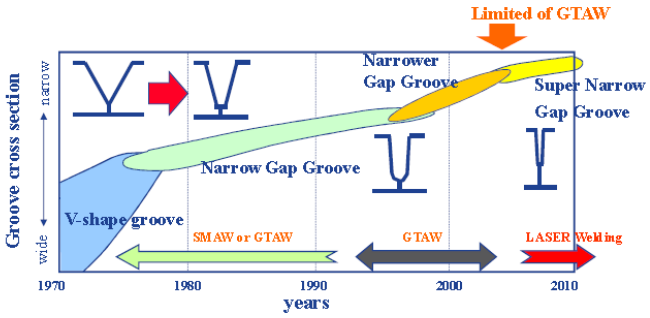
Mr Yasuda (GE Hitachi NE Intl.) explained that for the last 14 years the Hitachi-GE has undertaken manufacturing and construction work of NPP. So manufacture technology such as welding, water jet peening and large/small size machining has been developed to increase the quality of manufacture.

- Welding technology
  Development of welding method was presented to improve reliability and efficiency. Currently, super narrow gap groove is available using laser welding to increase reliability and reduce welding cost. The GE Hitachi NE Intl continuously studies and develops the latest welding technology. Welding is required for example for reactor internals that require precise welding structures with higher reliability and accuracy and lower maintenance requirement. The targets are to minimize deformation after welding, suppress residual stress to minimize the stress corrosion cracking (SCC) risk (thus improving stress distribution at the welding part), improve the reliability and work efficiency and work environment.



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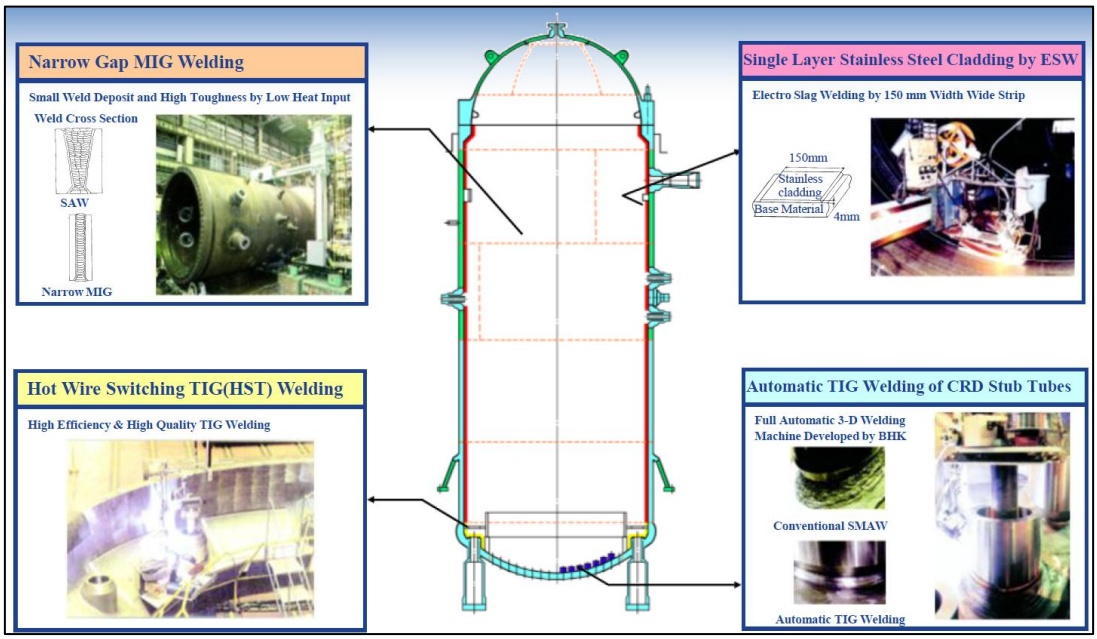
FIGURE 4.3.10. Development of welding method



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FIGURE 4.3.11. Development of narrow gap welding

Several welding technologies used for the ABWR RPV are shown below: narrow gap MIG (metal inert gas) welding, single layer SS cladding by ESW (electro slag welding), hot wire switching TIG (tungsten inert gas) (HST) welding and automatic TIG welding of CRD. The most suitable welding methods are used depending on the application in order to achieve high quality and low cost. Automatic welding is used in most welding parts.

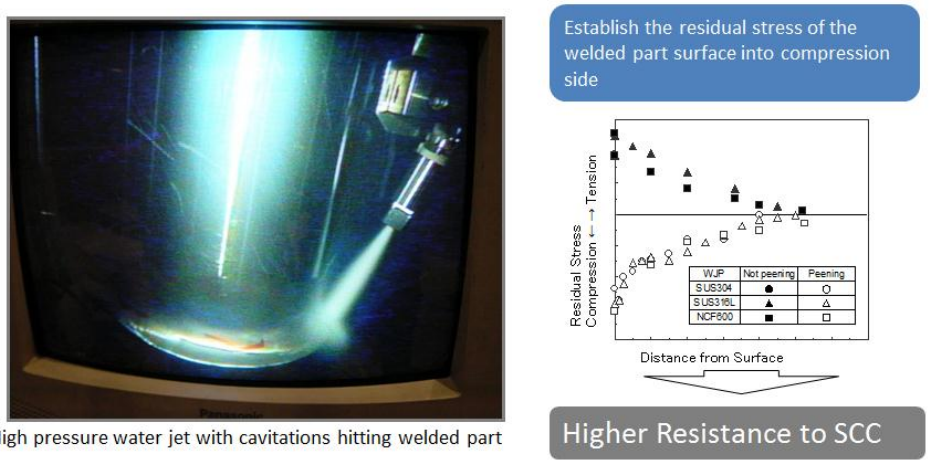


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FIGURE 4.3.12. Welding technologies used for the ABWR reactor

- Water jet peening

Water jet peening is applied after welding to improve the stress distribution of welded part. As shown, by hitting high pressure water jet with cavitation in the welded part, the residual stress of the welded part is changed from tensile side into compression side. So, initial defect for the SCC becomes hard to occur.

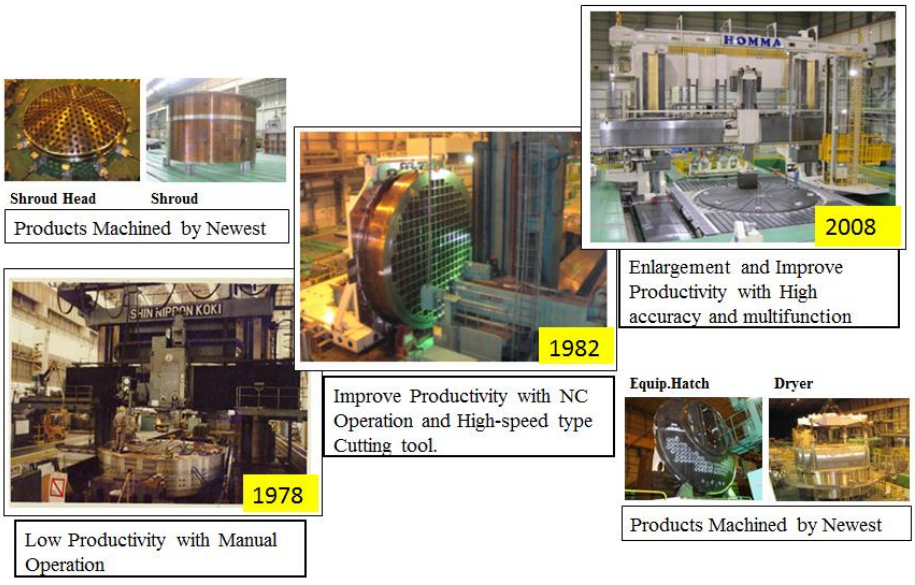


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FIGURE 4.3.13. Water jet peening

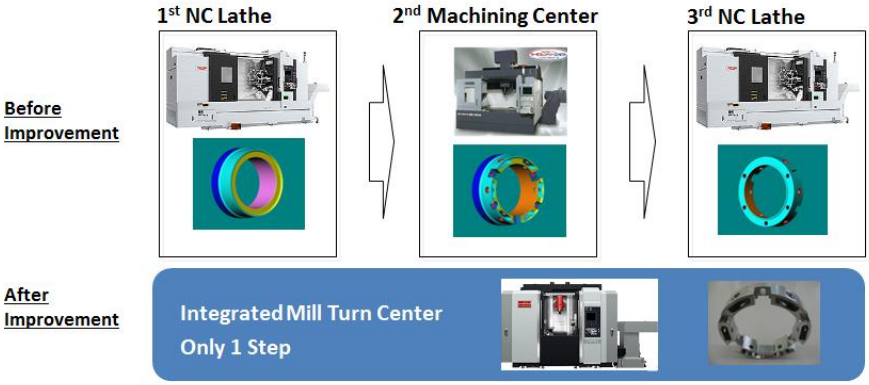
- Large/small size machining

Improvements have been made on the large size machining for enlargement and improvement of productivity with high accuracy and multifunction. On improvements of small size machining, an example is given on the integrated mill turn centre to achieve higher machining quality. The machine combines three steps of machining into a single operation, thus improves machining lead time, capacity and quality.



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FIGURE 4.3.14. Improvement of large size machining



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FIGURE 4.3.15. Improvement of small size machining

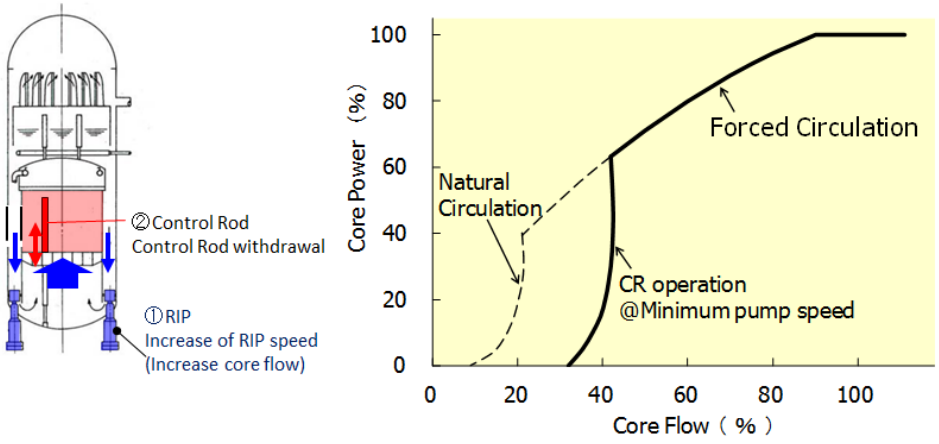
In conclusion, manufacture technology has been developed to increase the quality of manufacture for ABWR. The narrow gap welding technology has been developed to increase reliability and reduce welding cost. Water jet peening achieves higher resistance to SCC. Thus, EP1.1.1.3 is met.

Training programmes for workers (*not presented in the forum, but available on slide notes*)

Training programmes for workers is also offered by Hitachi. The training can begin at an early stage of high school and is thereafter available for all ages. The Hitachi technical high school is aimed at teaching fundamental skills in the fields of electronics, machining and welding and also to teach workmanship. A Japanese Challenge Skill Olympic is also held biennially by the Japanese government to cultivate young technicians of the ages 18-21. Hitachi GE supplies technicians with 2 year training courses for world class challenges to perform in the challenge. 50 former Olympians are currently working in the Rinkai Nuclear Factory as leaders. The investment in the development of workforce has paid dividends in work force of high quality and predictable productivity.

4.3.2.2. EP 1.1.2.8 Use of worldwide operating experience

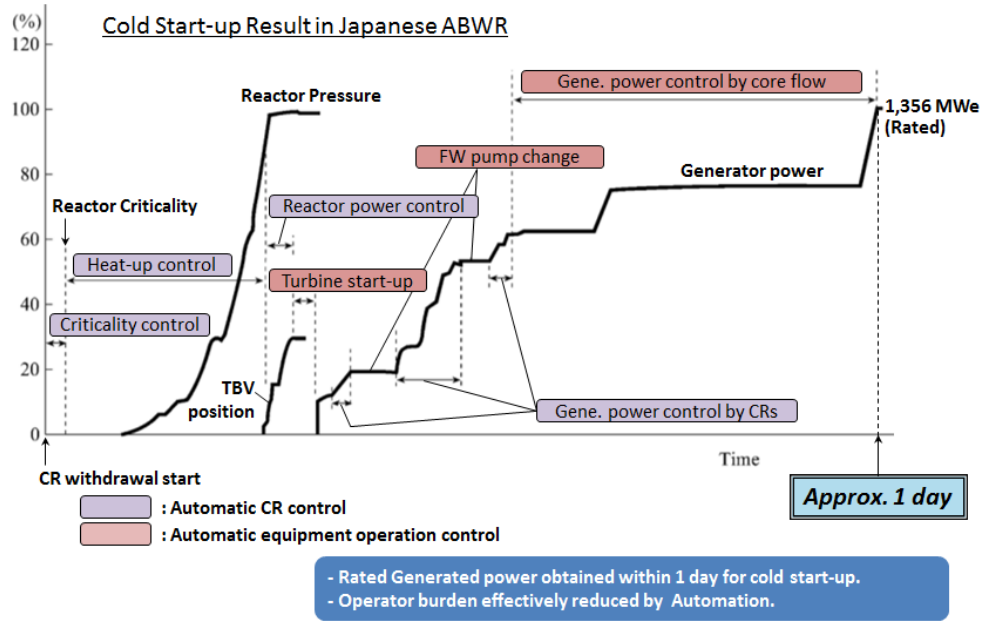
Mr Yasuda (GE Hitachi NE Intl.) addressed operability and safety, and construction technology and experience of the ABWR. The ABW|R has two reactor power control: RIP and control rod. The normal operating control can be achieved by natural circulation, control rod operation and forced circulation – as shown in the graph (core power vs core flow). This shows the reactor power can be easily controlled by altering the core flow rate, along with forced circulation line. Combined with fine motion control rods, a more flexible power operation can be achieved. For this reason, the operation of the ABWR is simple, easy and fast using automatic CR control and core flow control. Also, the ABWR has load-following capability by design.



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FIGURE 4.3.16. Control of reactor power for the ABWR

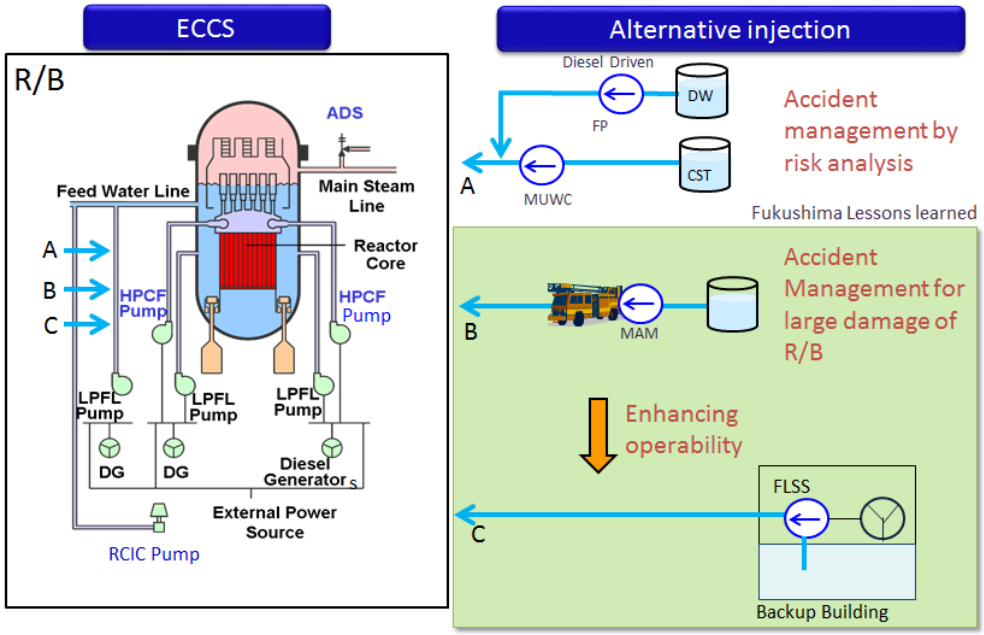
Operability of cold start up in Japanese ABWR is shown below, indicating reactor pressure and generation of power as controlled by automatic CR control and equipment operation control. As shown from a cold start-up, the rated power can be reached within one day. Automation of the reactor systems effectively reduces operator burden.



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FIGURE 4.3.17. Cold start-up result in Japanese ABWR

Mr Yasuda then discussed the safety of the ABWR including the countermeasures, and used the core cooling function of ABWR for illustration. The ABWR has redundant and diverse and independent 3-division ECCS for DBAs. In addition, alternative injection system like diesel driven fire pump is installed which is a countermeasure to vulnerability identified by risk analysis. The designer also recognizes the risk of external hazards by Fukushima accident by preparing for large damage of reactor building by external hazards. So risk management by mobile equipment is being reflected on ABWR as countermeasure for DEC on external events. Also enhancement of structure integrity margin and operability back-up building are being reflected on ABWR.

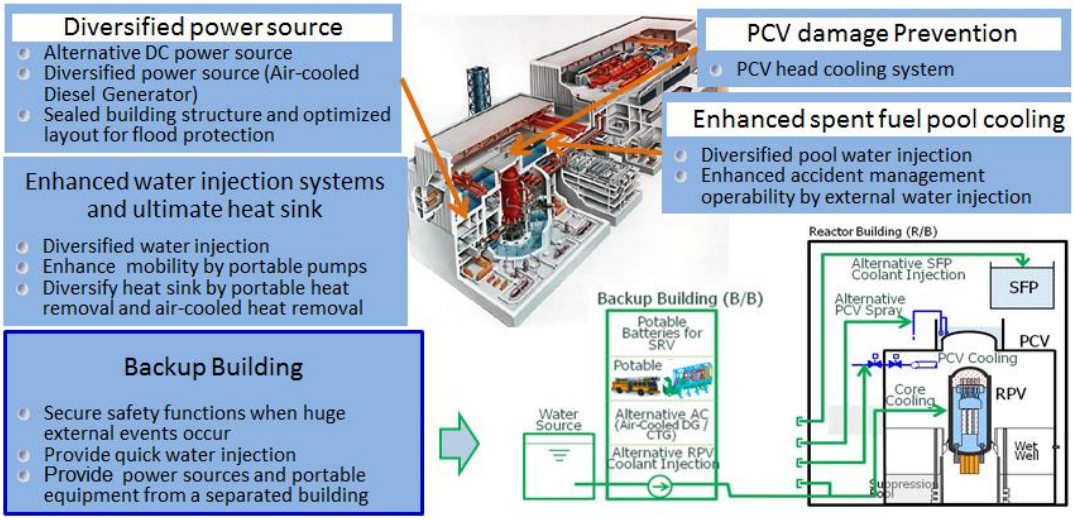


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FIGURE 4.3.18. Safety of ABWR: core cooling

The GE Hitachi NE Intl. recognized the threat of external hazard by Fukushima accident, i.e. common cause failure by external hazards. Diversity of accident management will be strengthened for external events, by adoption of the following countermeasures:

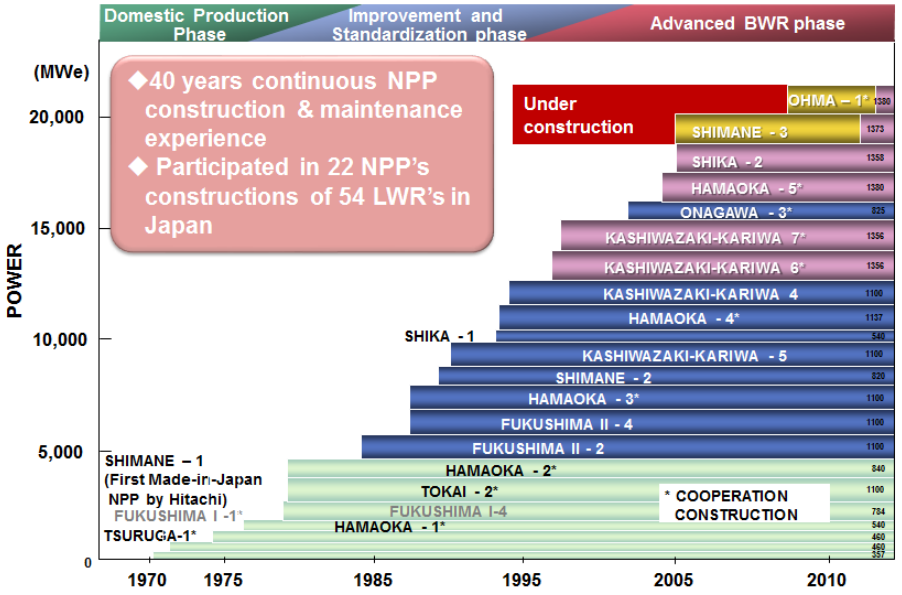
- Diversified power source: Installing alternative DC power, diversified power source (air-cooled diesel generator), sealed building structure and optimized layout for flood protection
- Enhanced water injection systems and ultimate heat sink: Installing diversified water injection, enhancing mobility by portable pumps, and diversifying heat sink by portable heat removal and air-cooled heat removal.
- Backup buildings: Securing the safety functions of the reactor when huge external events occur, providing quick water injection and to provide power sources and portable equipment from a separated building.
- Enhanced spent fuel pool cooling system: Providing diversified pool water injection and enhancing accident management operability by external water injection.
- Enhanced PCV head cooling system, to prevent PCV damage.



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FIGURE 4.3.19. Diversity of accident management as Fukushima countermeasures

Mr Yasuda discussed the ABWR construction technology and experiences. The vendor has 40 years of continuous NPP construction and maintenance experience, and participated in 22 NPPs construction of 54 LWRs in Japan including 4 ABWRs. Five units are currently under construction (including in Taiwan).

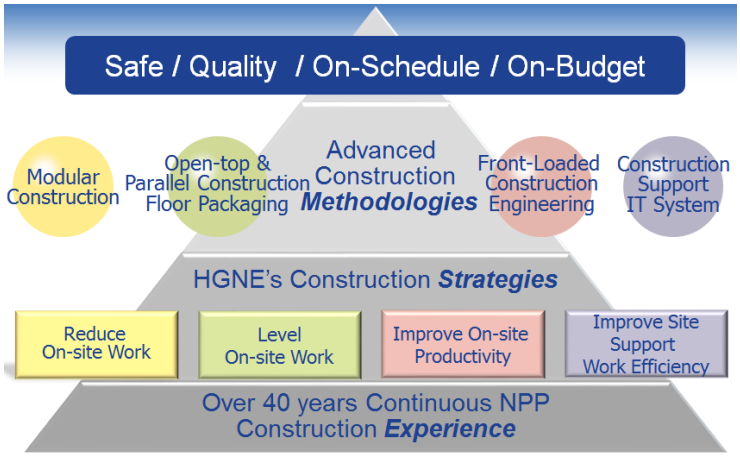


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FIGURE 4.3.20. Construction technology and experiences

Construction strategies consist of 4 elements based on the 40-year experiences:

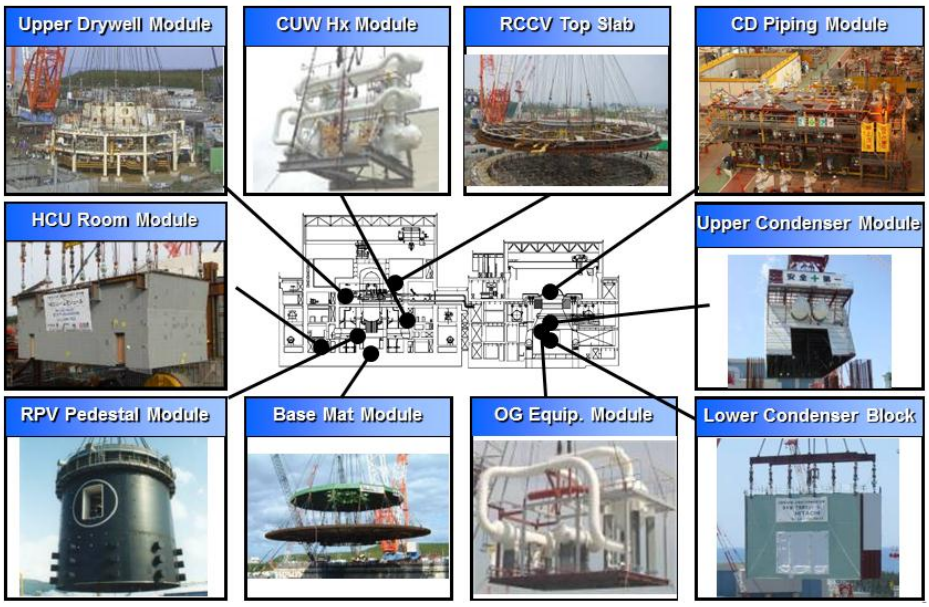
- Strategy-1: On-site work reduction (modular construction)
- Strategy-2: Level on-site work (open top and parallel construction
- Strategy-3: Improve on-site productivity / Front loaded construction engineering (evolution of plant engineering tools)
- Strategy-4: Improvement of site support work efficiency (integrated RFID technology introduction)



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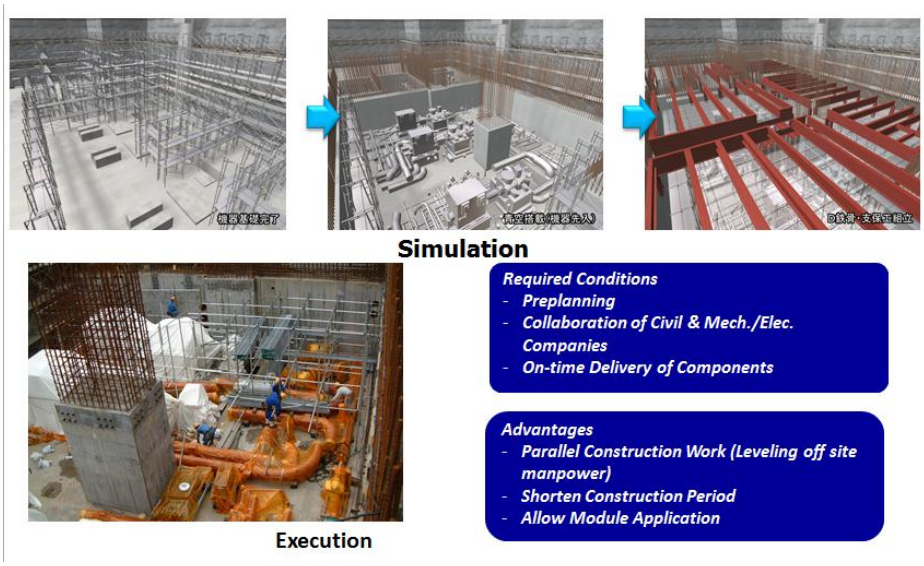
FIGURE 4.3.21. Construction strategies

Module construction is one of the effective methods to reduce on-site work. On the latest ABWR construction, 200 modules are applied. On the improvement of site support work efficiency (4th strategy), real time high accuracy control management work with Radio Frequency in Identification (RFID). In the latest ABWR construction, the RFID is applied for product management / real time progress work management. Product ID label with RFID is attached on the piping cap and sent to construction site. Worker in charge of warehouse management can read the information with PDA.



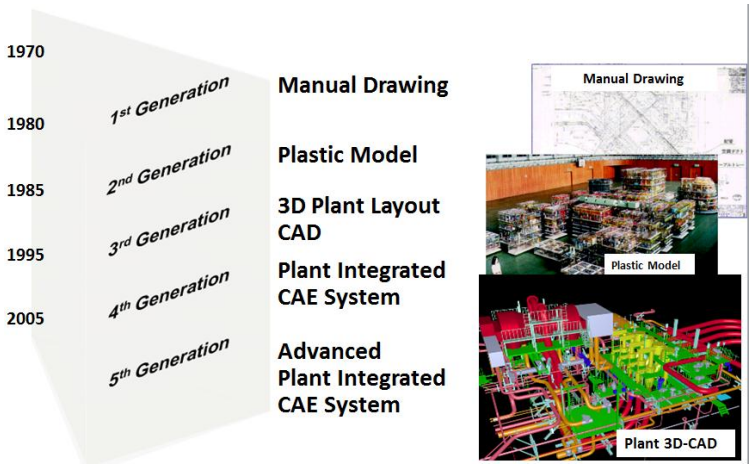
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FIGURE 4.3.22. Strategy-1: On-site work reduction (modular construction)



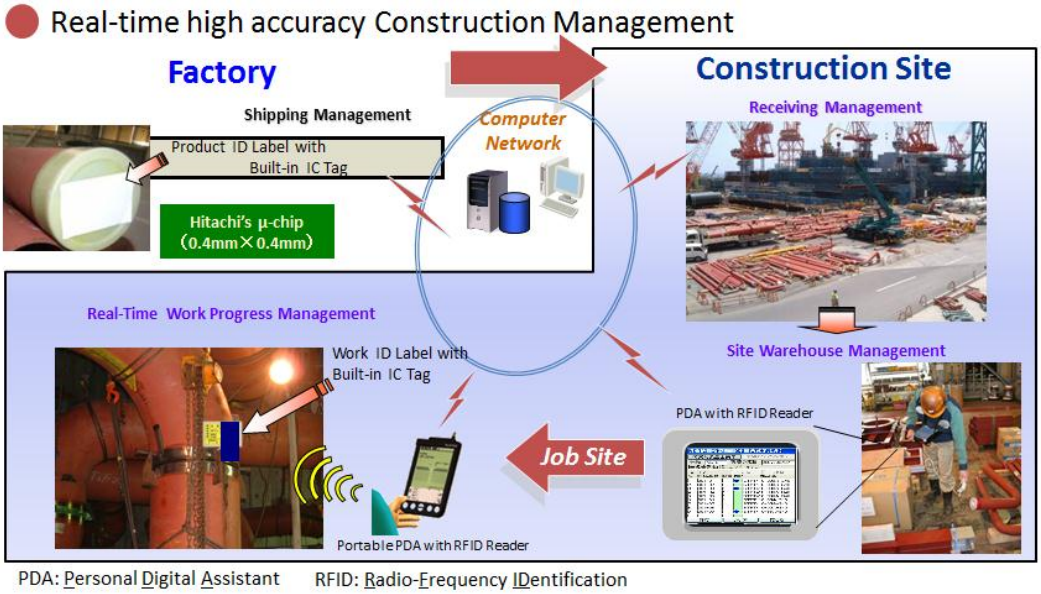
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FIGURE 4.3.23. Level on-site work (open top and parallel construction



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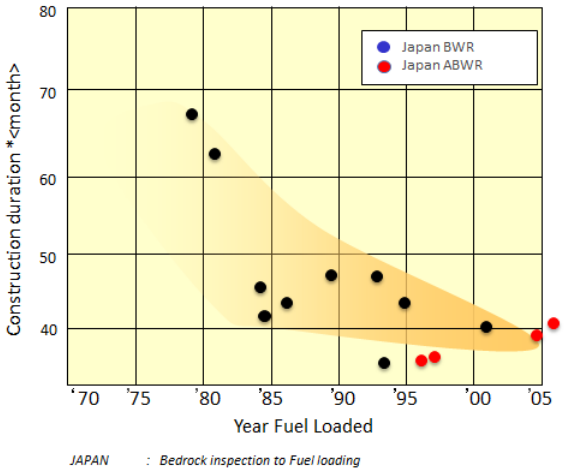
FIGURE 4.3.24. Strategy-3: Front loaded construction engineering (evolution of plant engineering tools)



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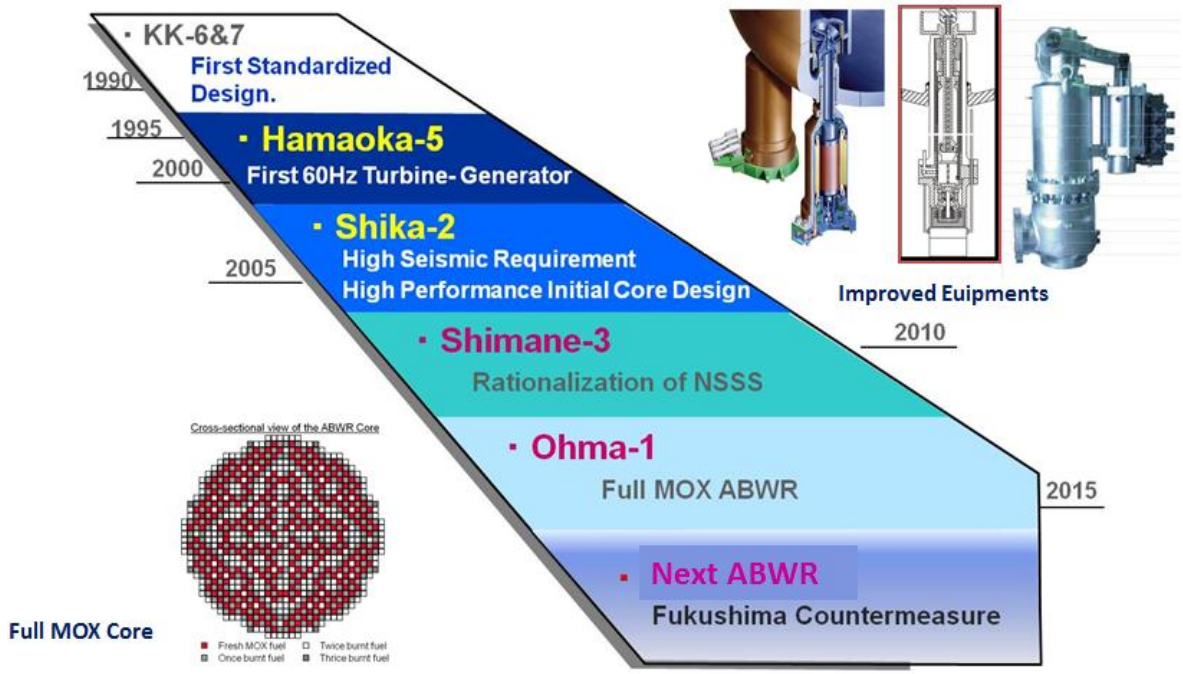
FIGURE 4.3.25. Strategy-4: Improvement of site support work efficiency (integrated RFID technology introduction)

Application of ABWR construction technology realizes short and feasible construction. Continuous evolving technologies with various customer requirements, site conditions and technological advancements are incorporated in every new ABWR.



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FIGURE 4.3.26. Constructability of ABWR – short and feasible construction



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FIGURE 4.3.27. ABWR design evolution

In conclusion, Mr Yasuda indicates that the ABWR can be recognized as the substantiation of the best available technologies from worldwide BWR evolution experiences. The design has been developed to achieve further safety and reliability by incorporating technical lessons learned from the Fukushima accident. The ABWR construction technology has evolved and been improved to realize short and feasible construction. Hence, EP1.1.2.8 is met.

4.3.2.3. CR 1.2.2 Grace period (normal operation)

Mr Beard (GE Hitachi NE Intl.) mentioned key features in the ESBWR to address the criterion are improvement in the I&C system, specifically the steam bypass and pressure control to control the operation of the turbine, and the feedwater level which controls the water level within the RPV. Both are essential because drop in water level will initiate transients. Historically there were a number of transients that

resulted from failures within the I&C, primarily because the previous I&C is single or dual redundant, and not triple redundant. The triplicated fault tolerant digital I&C (with turbine control (steam bypass and pressure control) and 3-element feed water level control) in the ESBWR requires 2 out of 3 logic. It was noted that transient is one of significant contributor to CDF in BWR, as a precursor to accident sequence. In addition, in the turbine island there are also 3×50% condensate pumps, 3×65% feedwater pumps, N+1 condensate and feed water for demineralizers, and standby condensate and feedwater pumps auto start if required.

In the nuclear island, support systems are arranged in 3×100% arrangement. The systems in standby can auto-start in an event in which the running system is disrupted. Mr Beard indicated that these systems are in operation during normal condition to provide higher availability and reliability enabling not to install non-safety equipment to carry out the functions and complicate the mechanism. Each train has on-site-safety-related AC power source, available for power, component and service water cooling, compressed air, chilled water, heating ventilation and air conditioning, etc. There are internal redundancies within many systems.

The suppression pool cooling is automated to prevent inadvertent plant scram at high pool temperature.

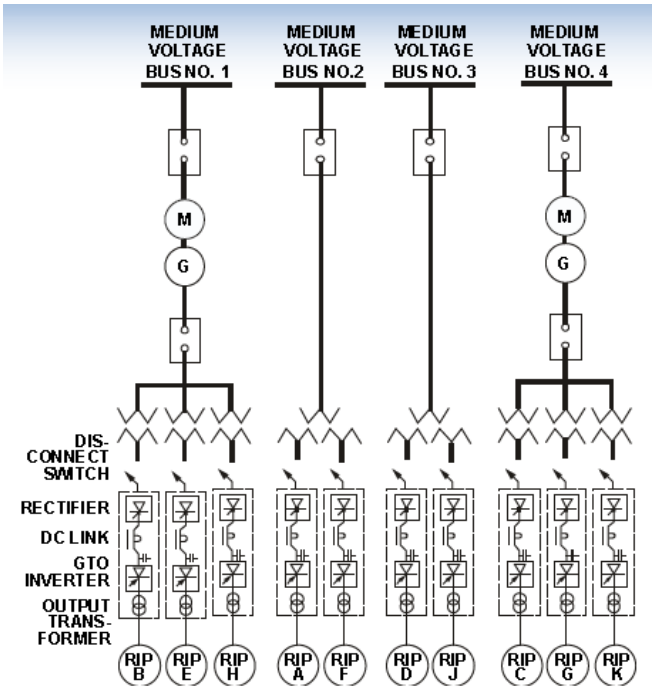
In conclusion, there is no hard number that can be referenced for grace period. Because the way PRA/PSA applied, there are some limitations due to generic values being used. One of the primary insights in the PRA is that the single largest contributor to CDF is loss of AC power or SBO, as high as 70% for ABWR. So the designer tried to decrease the number by adopting the ISC and alternate AC combustion turbine generator. BOP transients are another significant contributor to CDF, and effort was undertaken to look at BOP.

4.3.2.4. CR 1.2.3 Inertia

Mr Beard (GE Hitachi NE Intl.) pointed at both the nuclear island and turbine island to discuss inertia. He discussed the issue on the use of external recirculation in the previous BWR design, in which there are 2 pumps used. In the case either one trips, there will be significant transient. Historically there was a situation in which a condition of low flow and high reactor power occurred, and the reactor became unstable and had to be scrammed. To stay away from that regime of low flow and high power, reliance on 50% of pump power should be avoided. In the ESBWR nuclear steam supply system (NSSS), the reactor recirculation is achieved by using 10 internal pumps. This provides the redundancy, lessens the frequency of transients initiated by the recirculation system, and decreases the probability of entering unstable operation.

The turbine island has the capability to bypass 100% of the steam (100% load rejection capability) to the main condenser in order to rapidly reduce reactor power and decrease output of the main generator, in the order of 60 MWe. It can sustain generator load rejection without turbine trip or scram (island mode of operation). The decision for this feature was driven by the blackout in North America, to allow operator to restore the grid, given the limited power supply.

The recirculation loops are eliminated and replaced by reactor internal pumps (RIPs). There are 10 RIPs providing over 110% rated core flow, achievable even if one pump is out of service. The wet induction motor is of a seal-less design and is continually purged with clean water. The impellers and motor can be removed without draining the reactor. The back seating shaft and blowout restraint hangers provide redundant LOCA prevention. There is a solid-state, adjustable frequency speed control with multiple power supplies to reduce the possibility of significant flow loss. The dose source of recirculating piping has been removed. The ISI piping weld inspections and leakage sources in the drywell have been eliminated. Each RIP motor is driven by a solid-state variable frequency power supply, i.e. adjustable speed drive (ASD). 6 RIP ASDs are powered from constant speed motor-generator sets with flywheels whereas 4 RIP ASDs are powered by medium voltage buses.



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FIGURE 4.3.28. Reactor internal pump (RIP) power supply

In conclusion, the CR1.2.3 is satisfied.

4.3.2.5. CR 1.3.1 Frequency of Design Basis Accidents

Mr Beard (GE Hitachi NE Intl.) pointed out the ABWR was designed to reduce the frequency of DBAs. The criterion is interpreted more to initiating event that would lead to core damage. Transients initiated fault by I&C represent the significant contributor. For the reactor protection system and engineering safety features, multiple I&C are used to handle various functions within the design. The fault tolerant digital I&C incorporates the followings: (i) safety-related is 2 out of 4 same senses parameter, (ii) other key control systems are triple or dual redundant, (iii) lower frequency of I&C initiated transient is used, and (iv) human factors principles are applied to design operator interface.

The ABWR safety-related divisions are separated both physically by 3-hour fire barriers and electrically. Possibility of 'hot-shorts' is minimized. Over 30 m of large pipe below the core has been eliminated. The materials specifications (limit non-desirable trace quantities of materials) for stainless steel piping have been enhanced, along with piping installation requirements. Newly improved welding process is used to limit the stress built in and also compressive stress is induced on the interior of the piping to further mitigate the possibility of SCC.

Interfacing low pressure piping is protected from LOCA with the increased design pressure requirements. Pre-service and in-service inspection techniques are enhanced for better baseline examinations.

In conclusion, CR1.3.1 is satisfied.

4.3.2.6. CR 1.3.2 Grace period (Design Basis Accidents)

Mr Beard (GE Hitachi NE Intl.) pointed out that ABWR is fully automated to perform all actions necessary bring the plant to safe shutdown and maintain it for 72 hours. The fault tolerant digital I&C is designed to execute all actions required for the first 72 hours for transients and DBAs without operator actions. The response to the DEC events is also automated. For SBO, the ABWR qualifies as an AC independence and alternate power plant, and for the initial response is automation of the RCIC. The alternate combustion turbine generator will auto start but it requires operator action from control room to determine which electrical busses it will be connected to.

For the ATWS, there are diverse I&C system controls functions, backup actuation of hydraulic scram, alternate insertion of control rods electrically, and time delayed actuation of Standby Liquid Control (SLC)

to allow chance for electrical run in to complete. The SLC injects sodium pentaborate solution to bring subcritical. Either shutdown means is 100% capable to ensure negative reactivity to guarantee safe shutdown.

On the question whether the operator can override the response, GE's view of point is that if there is a valid safety related command in place, the operator can try. If there is a valid safety related demand coming from the instrumentation system, the I&C will govern the response of the plant.

In conclusion, the CR 1.3.2 is satisfied.

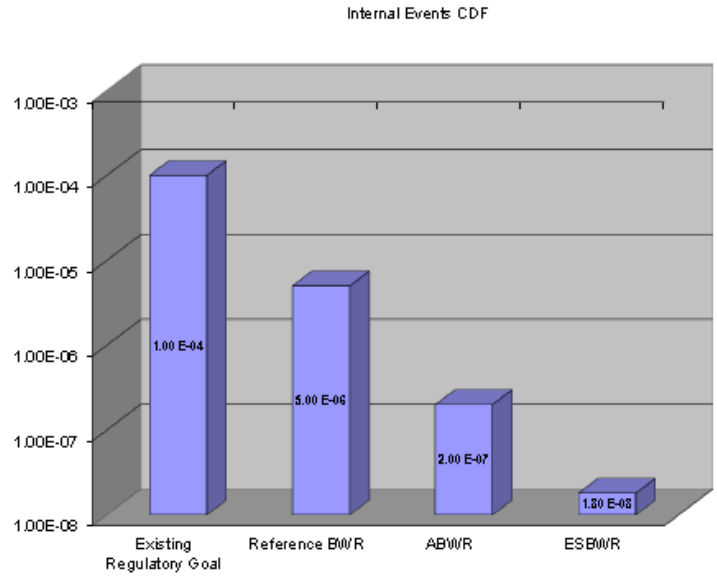
4.3.2.7. CR 1.3.3 Safety features

Mr Beard (GE Hitachi NE Intl.) pointed that some of the features are focused on the nuclear island. Each ESF division has a high and low pressure emergency core cooling (ECC) capability to lessen the chance of having to depressurise the vessel for transients and small LOCAs. The removal of residual heat occurs whenever a RHR division is operated in its many modes, and both the heat exchangers and supporting systems (service water and compounded cooling water) are normally aligned in operation. The interfacing low pressure piping is protected from LOCA with the increased design pressure (ASME) requirements of low pressure system such the system will not rupture (piping may deform). Recirculation piping is eliminated so that there is no large pipe breaks below the core and only small ECC cooling flow requirements are necessitated. The new improved RCIC turbine has been implemented (in Taiwan) which is cooled and lubricated by water and is mechanically governed. The size of the suppression pool strainer is increased and the fibrous insulation material is severely limited. Finally the available net positive suction head (NPSH) is increased so that the performance of the pump is not degraded at elevated suppression pool temperatures.

In conclusion, CR1.3.3 is satisfied.

4.3.2.8. CR 1.4.1 Frequency of major release into containment

Mr Beard (GE Hitachi NE Intl.) presented the comparison of internal CDF for ABWR (and also ESBWR) with reference BWR and existing regulatory goal. The calculated frequency of a highly degraded core, even for the reference plant, is 2E-7, well below the IAEA recommended value of 10^{-5} per unit-year. He cautioned the believable of the magnitude of frequency value as it depends on the assumptions made. However, the CDFs values are useful for identifying areas for possible design improvements. If want to truly compare the safety of design using CDF as primary criterion, then one needs to go to source where there is balance or treat the PSA in a common way. The US NRC has been doing so for various reactor designs.



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FIGURE 4.3.29. Comparison of core damage frequencies

4.3.2.9. CR 1.4.2 Processes

Mr Beard (GE Hitachi NE Intl.) referred to several available passive engineered processes in the ABWR design that would reduce the load on the containment barrier and reduce and/or control the activity in the containment atmosphere. He pointed at the ability to use residual heat removal system to spray primary containment. Two of the RHR trains can spray both drywell and wetwell. The spray function is not credited from safety related, but can be used in the event of core damage to remove radioactive materials from the atmosphere of the containment and transfer back to into the suppression pool. He noted one strong benefit of suppression pool which is for decontamination of / removing radioactive materials from atmosphere. Steam containing radioactive materials and non-condensable gases will be forced down under water, and bubbled up in wetwell airspace. So there is significant decontamination (DF of radioactive gases 10, and ~1000 for particulates). This ability is related to the containment overpressure protection system, which is a passive system using rupture disc. The release point has been designed from the wetwell airspace in the event of the need to initiate release from the containment.

In addition, the ESBWR utilizes inerted containment, lower drywell flood capability, lower drywell special concrete and sump protection, suppression pool, COPs, with PARs under consideration, and ACIWA.

In particular for corium, should the vessel be unable to contain it, there is ex-vessel core cooling. There was a criterion developed by EPRI for a spreading area of 0.2MW/cm^2 and the ABWR has in excess of that requirement. The ABWR also has passive lower drywell flooders. They are fire suppression sprinklers.

In conclusion, CR1.4.2 is satisfied.

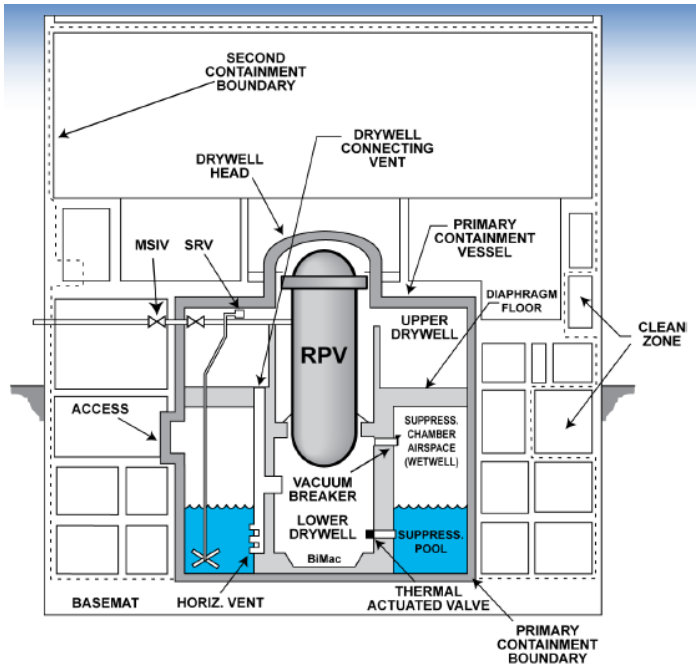
4.3.2.10. CR 1.6.1 Independence of DID levels

Mr Beard (GE Hitachi NE Intl.) noted a term DDD in GE as it is considered not sufficient just to have defence in depth, but also diversity, i.e. the third D stands for diversity. DID can be achieved by increasing redundancy but there is a need to look at the diversity. The largest contributor to core damage in the PSA is loss of AC power events including SBO. The RCIC provides an independent AC supply for cooling the core with automatic actuation and an 8-hour design basis capability, extendable over 24 hours if reasonable operator actions are taken. The air-cooled CTG provides an alternate AC source in case of loss of AC power, i.e. an additional DID. The CTG starts automatically but is connected manually from the MCR and can be connected to all key electrical busses. The ACIWA provides an alternative means of injecting water into the reactor pressure vessel, primary containment, containment spray and spent fuel pool. It utilizes a site diesel driven fire pump using water from on-site fire water storage, and has connections for fire trucks of portable pumps.

In conclusion, CR1.6.1 is satisfied.

4.3.2.11. CR 2.1.4 Confidence in innovation

Mr Beard (GE Hitachi NE Intl.) indicated practical innovation was incorporated in the ABWR design. A new arrangement of the suppression pool vent (also used in ESBWR) was introduced. The connection vent consists of vertical pipes embedded in the reactor pedestal with 3 horizontal pipes penetrating the pedestal under water into the suppression pool. 30 degree full scale tests were performed to verify the complete condensation of steam and that the hydro-dynamic loads were accurately modelled.



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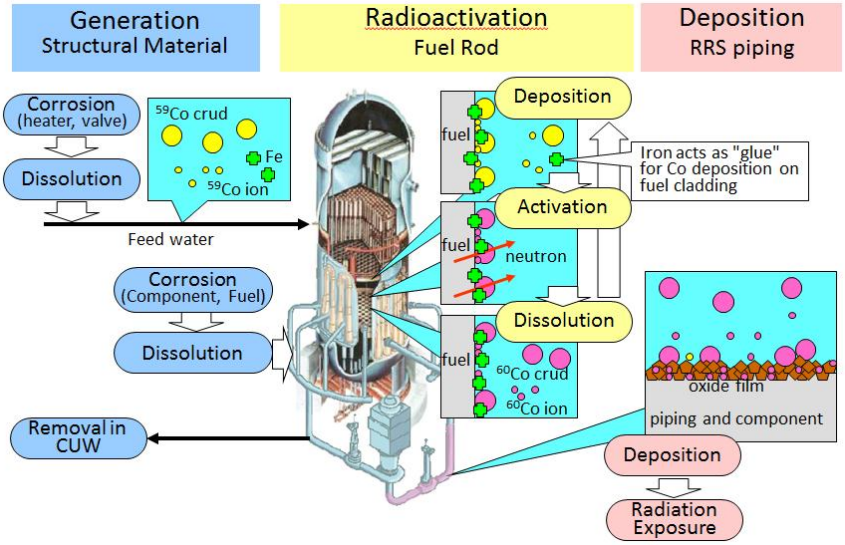
FIGURE 4.3.30. ABWR reactor building and containment

The standby liquid control (SLC) has been improved with new injection location inside the core shroud above the top guide (fuel). Scaled tests have been performed to ensure adequate mixing of the liquid neutron absorber (using air to simulate steam and inject dye to monitor adequate mixing in the vessel is obtained). Enriched sodium pentaborate is used to lessen the possibility of precipitation. The fine motion control rod drives (FMCRDs) are an evolution of the European technology and have undergone advanced duty cycle testing to ensure design 60 year lifetime and verify performance characteristics. The control rod drives feature improved safety with diverse insertion methods: fine reactivity control (electric motor) and 2 drives per hydraulic control (hydraulic scram). The ganged operation improves start-up time. Withdrawal line / scram discharge volume is eliminated. An electro-mechanical brake has been added to prevent rod run out on pressure boundary failure. There is a capability to detect drive / blade separation. The clean water purge flow results in reduced dose. The control rod drives require less inspection / maintenance as no organic piston seals are to replace, i.e. seal-less (magnetic coupling across pressure boundary - versus rotating seal in the previous BWR).

In conclusion, some innovative features have been improved relative to the reference plants and testing program was available to verify the improvement. So CR2.1.4 is satisfied.

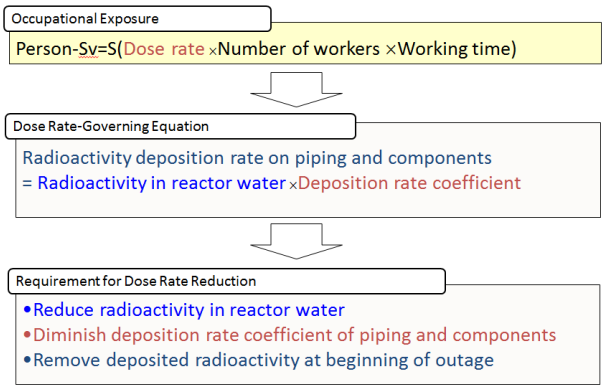
4.3.2.12. CR 3.1.1 Occupational dose

Mr Yasuda (GE Hitachi NE Intl.) presented the assessment of reduction of occupational dose. He started by outlining the formation and migration of radioactivity in BWR, that is, how metallic impurities such as Fe and Co generated by corrosion of structural materials are carried into the RPV, deposited on the fuel cladding surface and activated by neutron radiation. Part of the radioactive corrosion products is released and circulates back into the primary system and deposited on the piping surface posing as the main radioactive source. The occupation exposure to the workers is governed by dose rate, number of workers and working time. The dose rate is mainly governed by radioactive corrosion product concentration in reactor water and the deposition rate coefficient. The basic concept of dose reduction for a BWR is by means of reduction of radioactive corrosion product in reactor water, diminishment of deposition rate coefficient of piping and components, and removal of deposited radioactivity at the beginning of outage.



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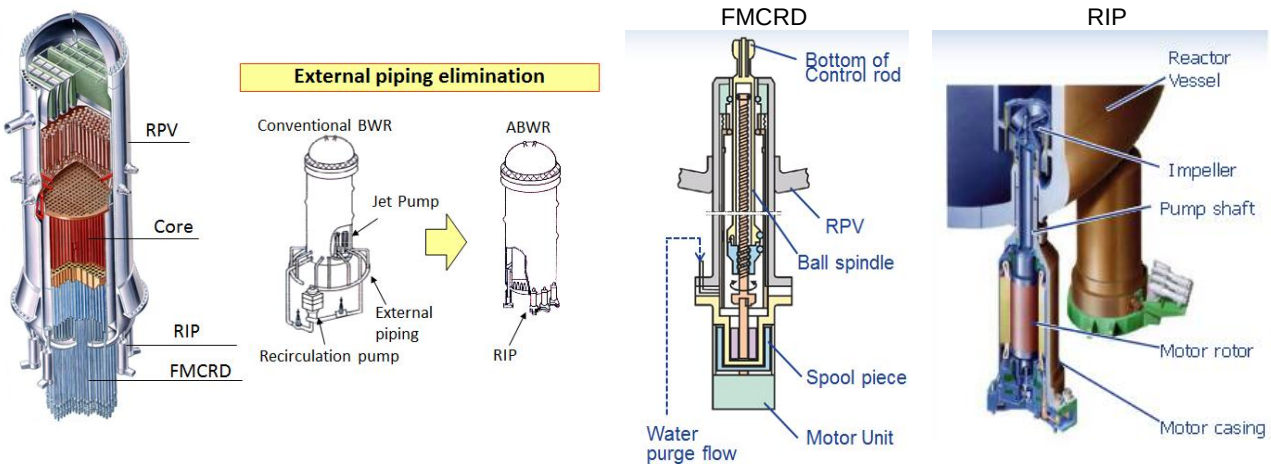
FIGURE 4.3.31. Formation and migration of radioactivity in BWR



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FIGURE 4.3.32. Basic concept of dose rate reduction

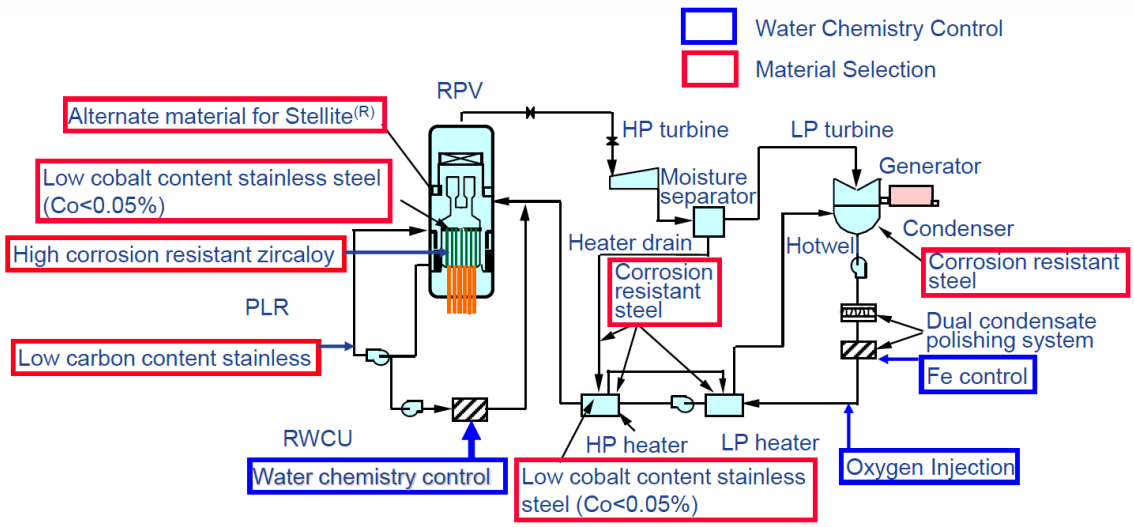
Mr Yasuda indicated that the ABWR system and automatic equipment for maintenance and inspection contribute to reduction of the number of workers and working time. The adoption of RIP (resulting in elimination of external piping) and FMCRD contribute to improvement of maintenance and reduction of occupational dose. By adopting RIP, dose source of recirculation system piping is removed, in-service inspection for recirculation piping weld is eliminated and no high activity recirculation systems valves are required to be maintained and tested. The two-piece design in mechanical portion (upper components and spool piece) of the FMCRD improves maintainability which results in reduced personnel dose (Components to be maintained are limited at spool piece only). The FMCRD is also dismounted by handling machine, which will also drain the water inside it. Its clean water purge flow allows less radiation exposure than that in the conventional BWRs driven by reactor coolant.



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FIGURE 4.3.33. Adoption of RIPs and FMCRD contribute to dose reduction

The direct heat cycle system of the ABWR is presented in the figure below. There are two measures used, which are water chemistry control and material selection. Control of water chemistry is carried out by (i) Dual condensate polishing system to remove corrosion products, (ii) Oxygen injection to reduce carbon steel corrosion, to keep oxide film on piping and components, (iii) Reactor water cleanup system to remove impurities. Components with low Co materials are used to reduce source term. Co-60 is noted as the main radioactive source during outage. Low Co materials are applied to the large source term contribution parts of the plant. It is shown the relative Co-60 production amount has been reduced to less than 50%.



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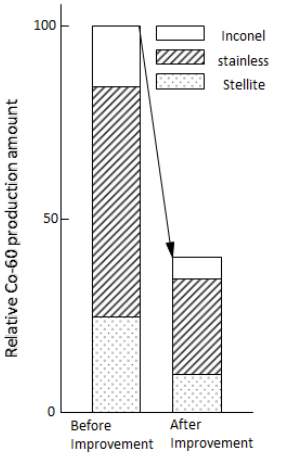
FIGURE 4.3.34. Corrosion production reduction by water quality control and material selection

Material	Inconel(R)	Stainless	Stellite(R)
Conventional	0.22%	0.25%	45-64%
Low Co	≤0.05%	≤0.05%	≤1.0%*

*: alternate material

Material	Component	Applied
Inconel	Spacer spring in fuel	O
	Others	O
Stainless	CR sheath	O
	CR tube	O
	HP feedwater heater tube	O
	LP feedwater heater tube	× (Small contribution)
	RPV internals	× (Small contribution)
Stellite	CR pin, Roller	O
	Valve etc	△ (Partial)

O: Adoption of Low Co material

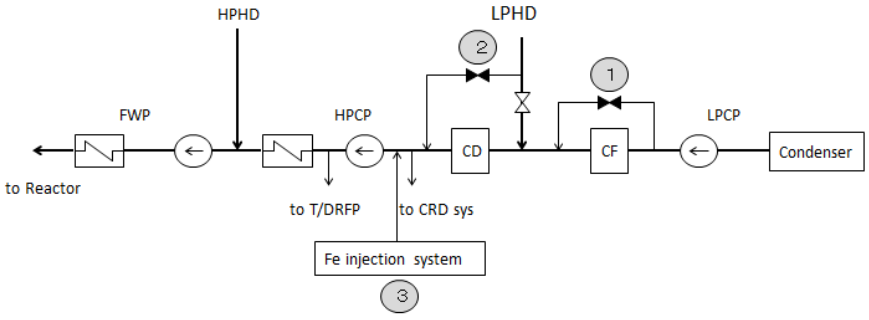


Low Co materials are applied to the large contribution parts of the plant.

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FIGURE 4.3.35. Implementation of low Co materials and effectiveness of Co reduction

Fe control methods are adopted in ABWR to reduce corrosion production. The optimum Fe reduces the radioactivity concentration based on ABWR operational experience. The diagram illustrates feedwater and condensate water lines. There are 3 methods to control Fe concentration as shown below. Condensate filter bypass can easily achieve optimum Fe concentration, and Fe injection into feedwater is also considered easy to control.

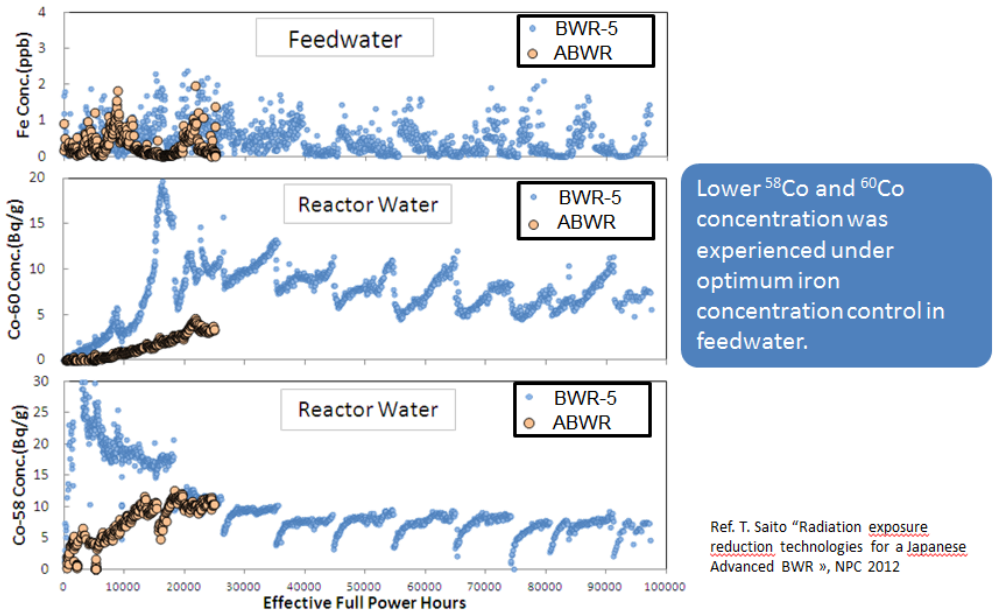


Item	①	②	③
Method	CF Bypass	LPHD mixed in CD Downstream	Fe injection into Feedwater
Feature	- Easy to operate after stabilizing of CD iron removal efficiency	- Effective for increasing small amount of Fe - Rigid	-Flexible, Easy -Short term, Auxiliary
Range of Fe conc. control	0.3-1ppb	0.4-0.5ppb	0.1-1ppb

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FIGURE 4.3.36. Fe control methods to reduce radioactive corrosion product concentration

Mr Yasuda presented the Co-58 and Co-60 concentration in Japanese BWR5 and ABWR. As shown in the figure, low Co-58 and Co-60 was experienced under optimum Fe concentration control in feedwater.



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FIGURE 4.3.37. Comparison of ^{58}Co and ^{60}Co concentration in ABWR and BWR-5 reactor water

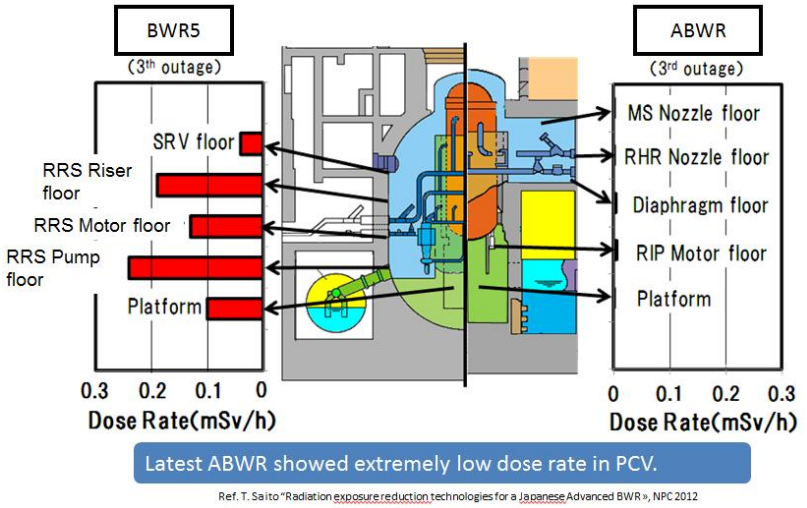
The role of automatic equipment for maintenance and inspections to reduce occupational dose was discussed. The GE Hitachi NE Intl. has developed considerable remote-automatic maintenance and inspection equipment based on proven robotic technologies like CRD handling equipment and in-service inspection equipment. During outage, the FMCRD is inspected and maintained. The FMCRD handling work is performed with the CRD handling equipment installed under the vessel for assembling and disassembling the FMCRD. The disassembled FMCRD is inspected and maintained with the maintenance equipment. The in-service inspection equipment is used to detect defects on the outer RPV (body, support skirts, flanges, nozzles and their corners), pipes and RIP nozzles using ultrasonic probes.



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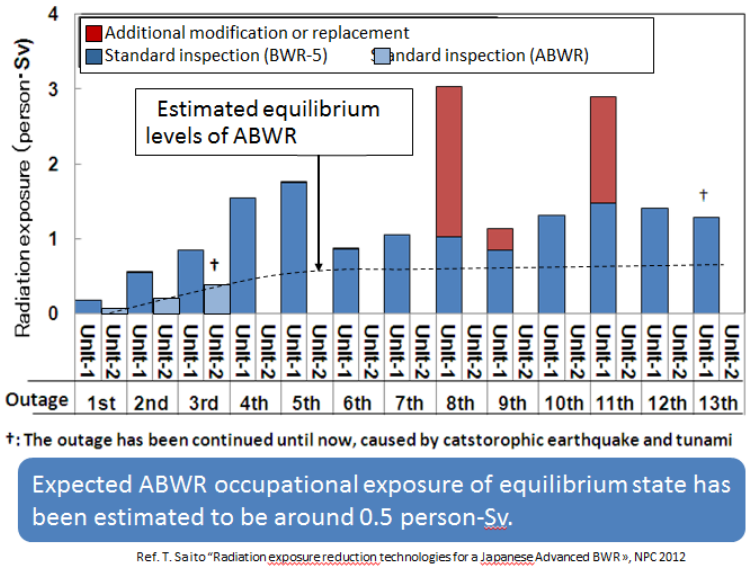
FIGURE 4.3.38. Automatic CRD handling equipment and in-service inspection equipment

The dose rate in the primary containment vessel (PCV) of the ABWR is shown to be extremely low in comparison with that of the BWR5. This is attributed to the absence of external recirculation piping, good materials selection and water chemistry control. The expected occupational exposure for the latest Japanese ABWR is presented below (Co-60 service concentration typically after 5 years of operation). As shown, the occupation dose at the outage is smaller for ABWR, and the expected ABWR occupation dose after reaching equilibrium state is estimated to be ~0.5 person-Sv.



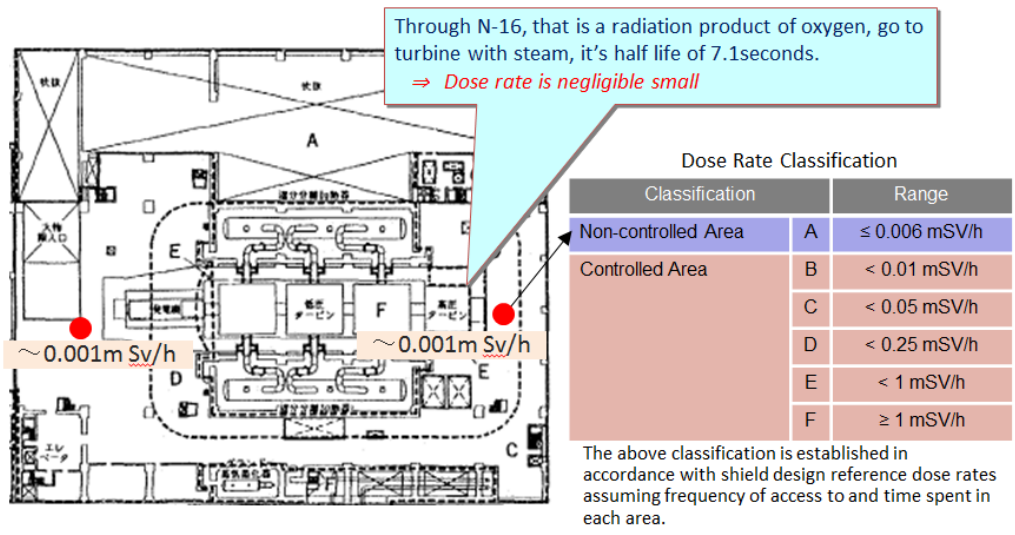
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FIGURE 4.3.39. Comparison of dose rate in primary containment vessel (PCV) for ABWR versus BWR5



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FIGURE 4.3.40. Comparison of occupation exposure of ABWR



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FIGURE 4.3.41. Small radiation exposure in BWR turbine

In conclusion, Mr Yasuda pointed out that adoption of RIP and FMCRD, corrosion production reduction by material selection and water quality control, and the use of considerable remote automatic maintenance and inspection contribute to the reduction of occupational dose, which is estimated to be sufficiently low compared with worldwide statistics. Thus, the CR3.1.1 is met.

4.3.3. Availability of References

The following references are available on the US NRC website:

- Design Certification Document, Rev. 4: http://www.nrc.gov/reactors/new-reactors/design-cert/abwr.html#dcd
- Safety Evaluation Report, NUREG-1503 and NUREG-1503 Supplement 1: http://pbadupws.nrc.gov/docs/ML0806/ML080670592.html
- Certification Rule, 10CFR52 Appendix A: http://www.nrc.gov/reading-rm/doc-collections/cfr/part052/part052-appa.html
- Design Control Document, Rev. 5: http://pbadupws.nrc.gov/docs/ML1100/ML110040323.html (*recently submitted, not yet reviewed by NRC*)
- Reference BWR (Grand Gulf NPP) NUREG-1150: http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1150/ (*US NRC study of core damage frequency for 5 different plants*)

4.3.4. Discussion

Questions	Answers
Mr Koshy (IAEA) asked specific areas in in which GE Hitachi NE Intl. has done improvements subsequent to the Fukushima event.	**Mr Beard (GE Hitachi NE Intl.)** responded that there have not been any specific design changes after Fukushima as these are usually adopted on a site-specific basis, in part because the site threat dictates what some of the alternatives are. The designer believes ABWR already has a significant robustness in its already certified design but there is room for improvement. One of the changes envisioned is to have pre-hard-wired connections for portable electrical equipment to connect into battery charger and safety related busses. There is already substantial capability to pipe fire water to wherever needed. He reminded tsunami is not a threat throughout the world.
Mr Koshy (IAEA) noted that the RCIC turbine continued to operate at Fukushima and generally in a typically design it is supported by a central battery system. He asked whether the philosophy has been changed (for the ABWR) with the focus on RCIC to continue to operate in the absence of AC power.	**Mr Beard (GE Hitachi NE Intl.)** explained that to address the SBO is dictated by the US NRC. The vendor designed the battery for 8 hour capability / SBO. Beyond that, the new improved RCIC turbine uses a mechanical governor for its operation. Even if the batteries deplete, it is easy for operator to continue to operate the pump and inject coolant water into the RPV.
Mr Koshy (IAEA) asked since the reactor is designed for full load-following, recognizing that the secondary system can give transients to the primary system, what level of load-following or whether the design would allow the transmission system operator to take over the control of the load.	**Mr Beard (GE Hitachi NE Intl.)** replied the ability has been included in the design of the I&C system to allow for remote dispatch. This function can be turned on or off depending on the local regulator's acceptance. On the other question, load-following would be in the 100% to 50% power range and it is possible to do a step change of almost 10% power in 10 seconds followed by about 5% per minute after that within that range. The design has included enough transient cycles within the 60 year design of ABWR to do that design operation for 90% of the design life.
Mr Koshy (IAEA) noted that as explained during SBO the RCIC operation is limited to 8 hours and further requires operator assistance. Assuming a SBO what	**Mr Beard (GE Hitachi NE Intl.)** explained that the current thinking (focusing on the US NRC) is using flex strategy, that is, having portable equipment (requiring 7-8 kW generator) pre-designated and pre-sited in very close proximity to the site, with capability to recharge battery and re-establish remote operation.

the level of acceptance needed to extend it to indefinite capability is.	Eventually, other means is required to remove decay heat from the primary containment. This really goes if the site hazards have disrupted safety water or service water system or component cooling water system. If so and they cannot be re-established, then the design has containment vent system to remove the heat.
Mr Seo (KEPCO-E&C) asked that since the ABWR has 100% steam bypass capability to the condenser, whether this would be too much burden or too much heat loss to the condenser.	**Mr Beard (GE Hitachi NE Intl.)** said that taking advantage of the thermal mass of the main condenser, the reactor is able to bypass 100% of the steam for a short period of time, during which the reactor power would have to be brought down into the normal capacity. There are about 3-4 minutes of thermal mass to allow that. There is also automatic power control for the purpose.
Mr. Grudev (Bulgaria) stated that in severe accident management the ABWR does not use a filtering system but only a scrubber for the retention of the radio nucleate, but if the water evaporates then this would result in a dangerous situation.	**Mr Beard (GE Hitachi NE Intl.)** clarified that if the reactor starts to over pressurise the containment it is largely because there are non-condensable gases and it will only be necessary to vent for short very period of time before getting rid of the non-condensable gases and closing the vent. Given the population density in Europe and other areas of the world, regulators may insist on another external vent and this can possibly be added.
Mr Bykov (Rosatom) asked the external impact influence protections on the NPP such as seismic and airplane crash.	**Mr Beard (GE Hitachi NE Intl.)** responded that the ABWR has recently been assessed for its ability to deal with an NRC defined threat of an airplane crash and it satisfied the acceptance criteria. Nevertheless, additional fire door needs to be installed to deal with potential fire due to ingested fuel. In terms of seismic protection, the reactor can withstand 0.3g, following the NRC response spectrum from 0 - 33 Hz. (which is further enhanced in the ESBWR (adding high frequency content) to account for that the NRC criteria may not be that conservative to many areas in the US).
Mr Aziz (Indonesia) would like to know more about how the passive autocatalytic recombiner works.	**Mr Beard (GE Hitachi NE Intl.)** explained that they use a platinum catalyst that when it comes in contact with hydrogen and oxygen, the two will combine generating water and heat.
Mr Susyadi (NPTDS) asked regarding the containment of the pressure protection system. The ABRW uses rupture disk but at Fukushima the rupture disk was troubled. Are there better designs?	**Mr Beard (GE Hitachi NE Intl.)** stated that Fukushima did not have rupture disks in their containment vent and that their containment vent cannot be classified as a hardened vent. This is a philosophical question that needs to be discussed with the regulator. GE believes that it is better to pre-analyse the desirability of venting and provide that capability passively than to rely upon operator action and decision making in the heat of an accident to decide whether or not to vent, and the NRC concurred with this. Having said that, there are valves on the hardened passive vent that can be closed to re-establish the primary containment boundary.
Mr Reitsma (IAEA) pointed out issue on cost and safety. It seems measures for dose reduction in ABWR may not cost a lot by applying engineering and safety principle. He asked for comments on costing and the implementation in the new design.	**Mr Beard (GE Hitachi NE Intl.)** indicated the cost has been very minimum. Besides reduction in potential source material, the water chemistry program has been very effective in keeping the dose low. The criterion was selected to showcase BWR reputation of having low exposure and the achievement the design has achieved now.
Mr Dulera (India) asked to elaborate on the in-service inspectability aspect.	**Mr Beard (GE Hitachi NE Intl.)** said that during the design process 3D CAD is used to ensure that clearances are available for in-service inspection (e.g. UT equipment). Turbines are inspected as part of 10-year maintenance program for turbine, i.e.

	to turn down each section and do the necessary dye penetrant and UT inspections, to ensure performance. Reactor welds are not inspected (internal criteria to GE).
Mr Kahlid (Malaysia) questioned how GE Hitachi NE Intl. ensures quality of subcontractors or suppliers.	**Mr Beard (GE Hitachi NE Intl.)** stated that it is a joint process and when subcontractors are hired, verification is done to make sure they have the proper QA plan in place and take the lead in doing the necessary inspections to ensure that their quality is up to date. But GE has announced and unannounced audits where inspection staff are brought in to check records and randomly inspect a second time to verify that what is being done is done appropriately. Depending on the level of importance there may be an authorized nuclear inspector to come in and complete the necessary steps for ASME certification for pressure vessels.
	During the afternoon discussion, **Mr Beard (GE Hitachi NE Intl.)** added information on shortening the construction schedule. The ABWR FOAK was built in 38.5 months, from the first concrete to fuel. Subsequent construction schedule is longer than that. This is because the optimal schedule requires an extension. The short schedule can be done but costs a lot. Looking at the overall impact on the cost of the plant, at one point it goes negative when the cost is compressed too much. So it depends on the country's need, whether the lowest overnight capital cost or that electricity is available as soon as possible. Common thinking is changing in a lot of cases because of the different economical standards.

4.4. APR1400

Mr Jong Tae Seo from KEPCO-E&C and Mr Han Gon Kim from KHNP presented INPRO assessment in the area of safety of Korean Advanced Power Reactor 1400 or APR1400 design.

4.4.1. Overview of APR1400 Design

Mr Seo (KEPCO-E&C) discussed the design development of APR1400 design. The development strategy started in the 1970s when Korea imported design as well as construction as turn-key projects. The first NPP started PWR 600 MW units in 1978, and several more units were imported from foreign vendors while Korea was building its own design and infrastructure. During the 1980s and 1990s Korea reached nuclear technology localization and became self-reliance and developed its own OPR1000 (Optimized Power Reactor). A total of 12 OPR units were built in Korea. While building OPRs, Korea started developing the next generation, i.e. APR1400 in the 2000s (exactly 2002). After the APR1400 design development, Korea has started to develop the 1500 MWe APR+ design with more reliable and advanced safety systems. The design approval is expected by the end of 2013, and Korea has solid plans to construct APR+ also.

Thus APR1400 is based on the OPR1000 design (2,825 MWth or 1,050 MWe) with an uprated power to 4,000 MWth (1,450 MWe) incorporating advanced design features (ADFs) to enhance safety and operational flexibility. The design met the utility requirements (domestic and world-wide) at the time, i.e. EPRI URD and latest codes and standards. APR1400 design also used proven technology as much as possible for the regulatory stabilization as well as ADFs including passive design features for enhanced safety, constructability, and maintainability. APR1400 received design approval from Korean regulatory body in 2002.

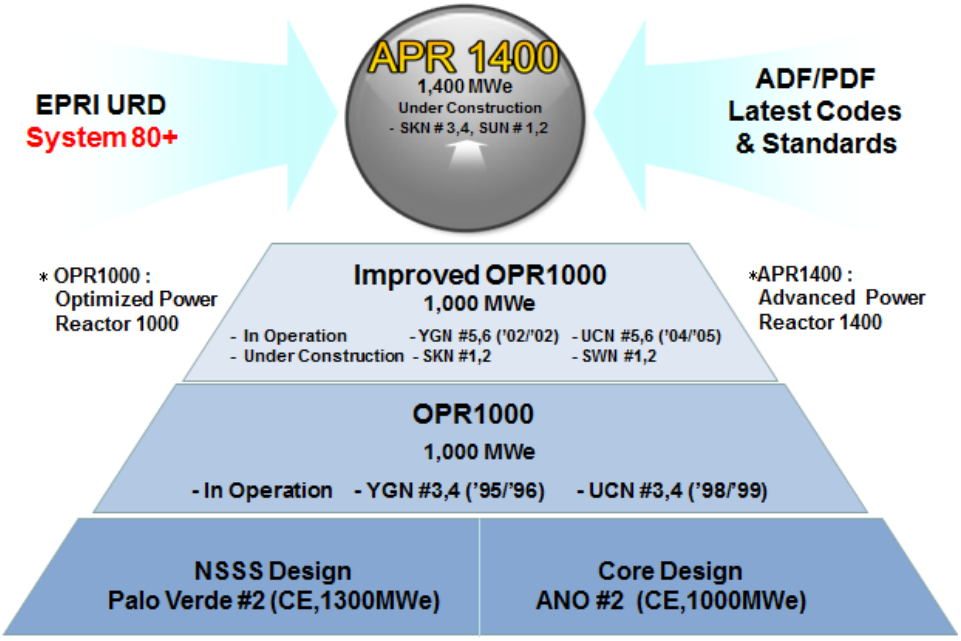


FIGURE 4.4.1. APR1400 Design Development

Mr Seo then discussed the design characteristics of APR1400, as summarized in the table below. The reactor design is unique, i.e. it is a two-loop but with 4 reactor coolant pumps.

TABLE 4.4.1. Design Parameters of APR1400

Parameters	APR1400
Thermal/ Electrical Power	4,000 MWth / 1,450 MWe
Design Life	60 years
Seismic Acceleration	0.3 g
Operating Parameters	
- T_{hot} / T_{cold}	615 / 555 °F
- Operating Pressure	2250 psia
- RCS Flow Rate	1.66×10^6 lb/hr
- Main Steam Pressure (@Full Power)	1000 psia
- Main Steam Flow Rate	8.975×10^6 lb/hr
Safety Parameters	
- CDF	$2.25 \times 10^{-6} < 10^{-5}$/RY
- Containment Failure Frequency	$7.19 \times 10^{-7} < 10^{-6}$/RY
- Thermal Margin	>10%
- Emergency Core Cooling System	4-train, DVI, Fluidic Device in SIT
Performance Requirements	
- Plant availability	More than 90 %
- Unplanned trip	Less than 0.8/year
- Refueling cycle	18 ~ 24 months
Number of assemblies	241
Batch average burn-up	55,000 MWD/MTU
Enhanced operational flexibility	Daily load follow operation capability MOX loading capability up to 1/3 core

The design features the Plus-7 Fuel™, meaning it has 7 advanced design features: (i) Reduced rod bow top Inconel grid, (ii) Mid-grid (9) with mixing vanes, (iii) Bottom Inconel grid with high burn-up capability, (iv) Debris filtering protective grid, (v) Removable top nozzle, (vi) Spacer grid, and (vii) Debris filtering bottom nozzle.

To support the enhanced RV integrity, there is no weld seam in the fuel region shell and low copper contents material is used. The less cobalt in the base material reduces radiation exposure to the operators. The integrated Inner Barrel Assembly (IBA) is welded to the Upper Guide Structure (UGS) upper flange thus

eliminating tie rods, round nuts, snubber flange and block. The integrated Lower Internal Assembly features integrated core support barrel, core shroud and lower support structure in one assembly.

Compared to the previous design, the APR1400 features an Integrated Head Assembly (IHA) integrating the various components such as head area cable tray system, CEDM (Control Element Drive Mechanism) air handling unit, cooling duct, cooling manifold and head lift rig, etc. The IHA design reduces refuelling time, i.e. lifting the RV head is very easy after the detensioning and taking off the 4 seismic supports. The reduced refuelling time in return will decrease the occupational radiation exposure. The IHA design also reduces component storage area and seismic load.

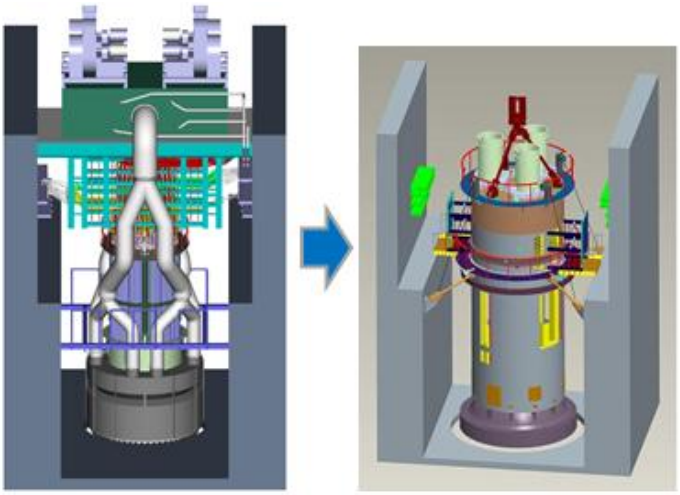


FIGURE 4.4.2. The Integrated Head Assembly (IHA)

The reactor coolant pump is a typical PWR: vertical, motor-driven centrifugal with bottom suction and horizontal discharge and single stage impeller.

The pressurizer is relatively bigger as compared to the power rating. The total volume is 2,400 ft^3 which enhances the capability of absorbing any plant transients. On top of the pressurizer typically there are safety and relief valves but the spring loaded safety valve and motor operated relief valve are now combined into a single Pilot Operated Safety Relief Valve (POSRV). The POSRV performs over-pressure protection and safety depressurization functions. The main valve opens by pilot valve actuation with steam/water/two-phase discharge. There is no drift of the opening set-point. The valve operation is reliable without chattering and leakage, and there is low susceptibility for valve stuck-open.

The design only has two large steam generators (SGs) each handling 2,000 MWth. The SG has integral economizer and 13,102 Inconel 690 tubes with plugging margin of 10%. Improved tube support bars and plate, i.e. increased anti-vibration bars, reduces the flow-induced tube vibration. The modified SG inlet nozzle angle increases space to install SG nozzle dam, hence improving stability in mid-loop operation.

The Safety Injection System (SIS) is one of the key improvements from the OPR design. Its simplified design has mechanically independent 4 trains with 1 Safety Injection Pump (SIP)/train, 1 Safety Injection Tank (SIT)/train, no low pressure pumps, and one injection mode. It has a Direct Vessel Injection (DVI), thus no safety injection water spillage in cold-leg break LOCA, and an increase in the reliability of the injection during LOCA. As shown in the figure, the SIP takes suction from the IRWST (In-containment Refueling Water Storage Tank) located around the bottom periphery of the containment building and injects directly to the Reactor Vessel (RV).

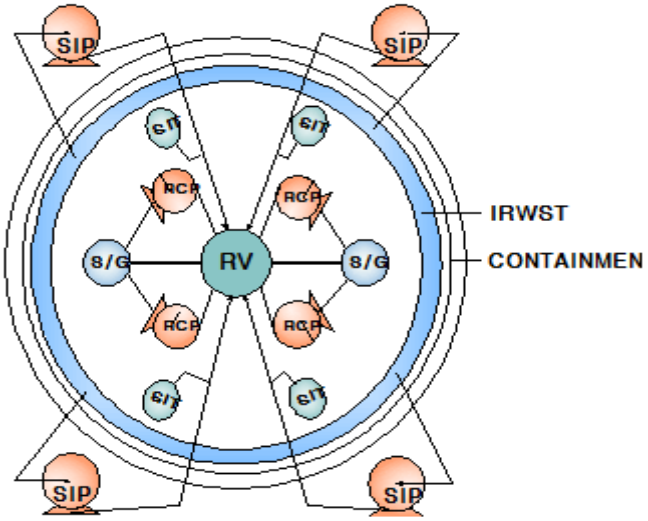


FIGURE 4.4.3. Safety Injection System

The APR1400 design adapted fully digitalized I&C systems as shown below, and all systems are connected to the data communication networks.

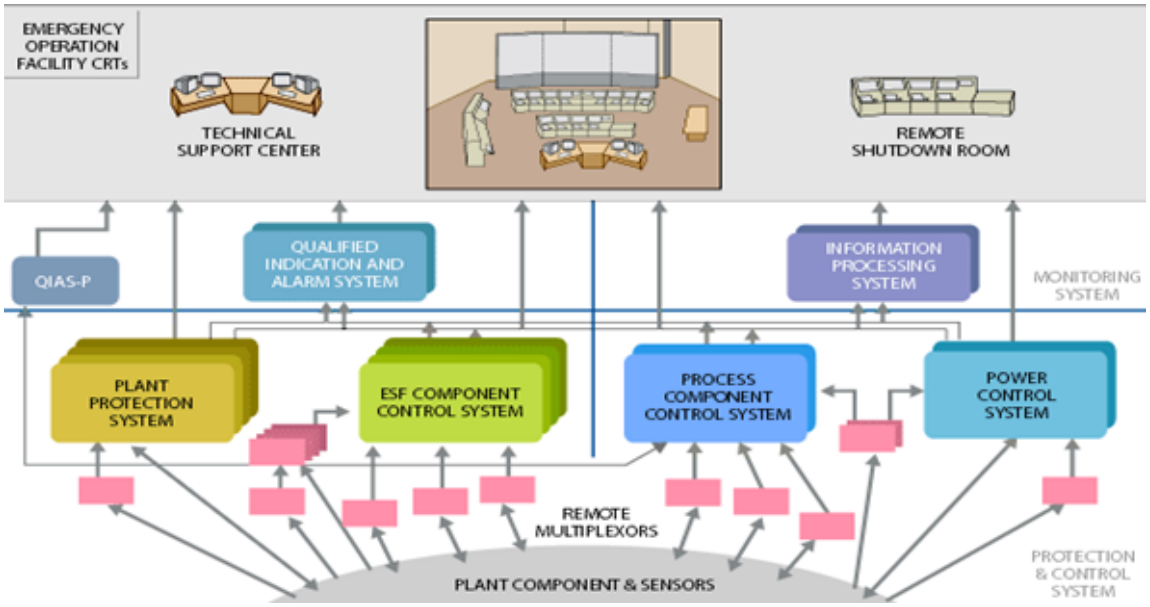


FIGURE 4.4.4. Overview of Man-Machine Interface System (MMIS)

The Advanced Control Room is composed of redundant compact workstation with soft control, large display panel, advanced alarm system, safety console (conventional control board) as a backup for common mode failure in safety I&C systems, and computerized procedure system.

The layout of the APR1400 features a twin-unit concept with a slide along arrangement, i.e. two units in parallel with a compound building in between, having common facilities for the two units. The reactor containment building is surrounded by the auxiliary buildings housing safety and other auxiliary systems.

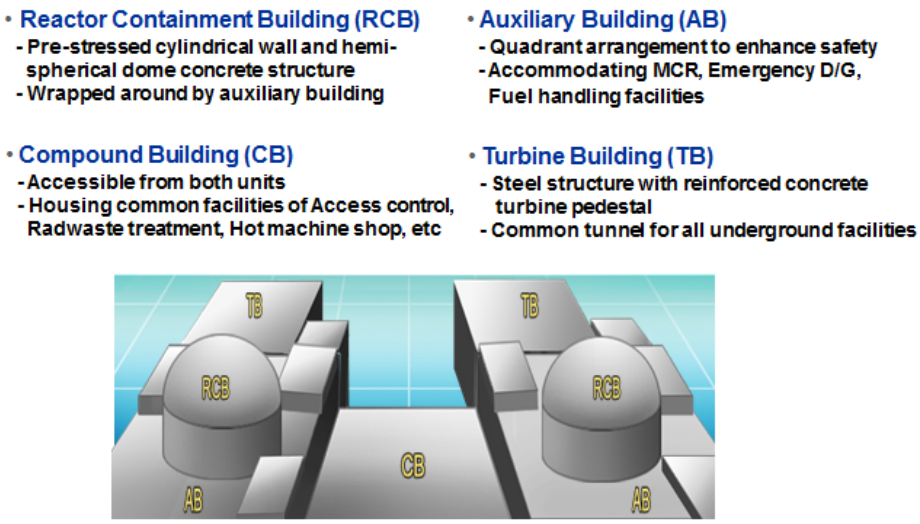


FIGURE 4.4.5. Layout with Slide-along Arrangement (2 units)

The auxiliary building arrangement uses a quadrant approach, i.e. four quadrants, each has one train of safety system physically separated from the others. So there is one SIP and one component cooling water pump (CCWP) in each quadrant. Also there are containment spray pumps (CSP) and shutdown cooling pumps (SCP) which are identical pumps and designed to be interchangeable. So in case of one pump failure, the remaining three pumps can still be used either for the containment spray or shutdown cooling for decay heat removal.

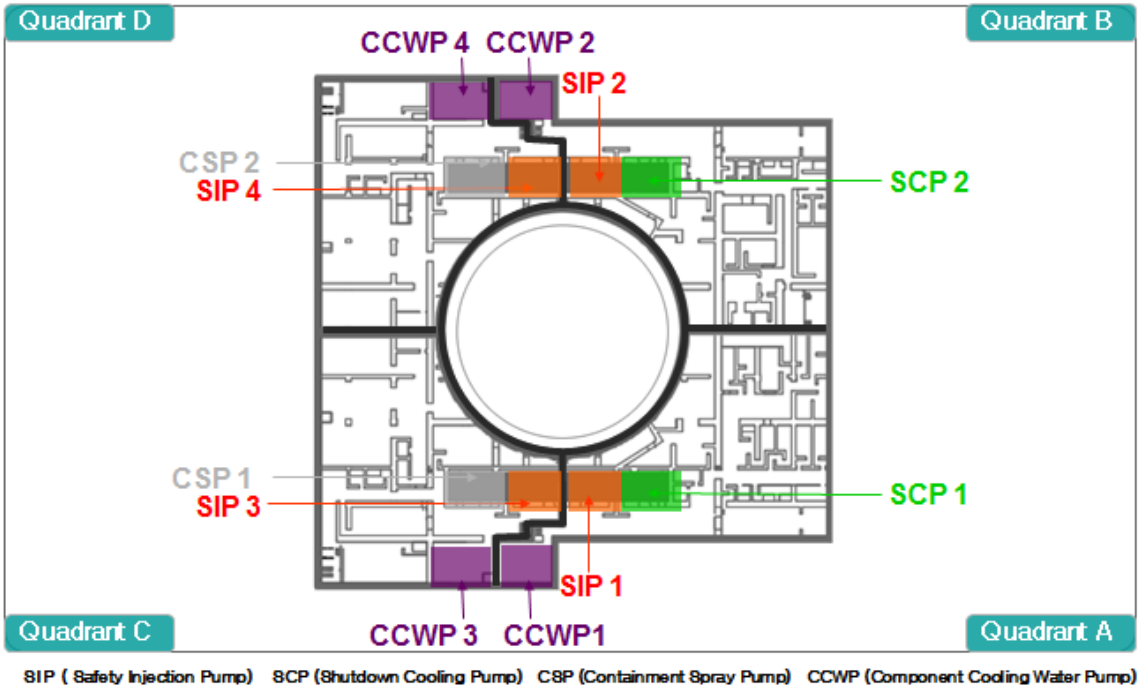


FIGURE 4.4.6. Quadrant Arrangement of Safety Components

Mr. Seo also presented the design verification activities performed during the APR1400 design development.

APR1400 adopted advanced design features that require verification.

- Fluidic device (FD)
  VAPER (Valve Performance Evaluation Rig) is used to evaluate safety injection flow performance of Safety Injection Tank with fluidic device. Inside the SIT there is a passive flow controller called FD. The test facility has the same size with full operating pressure condition as the APR1400.

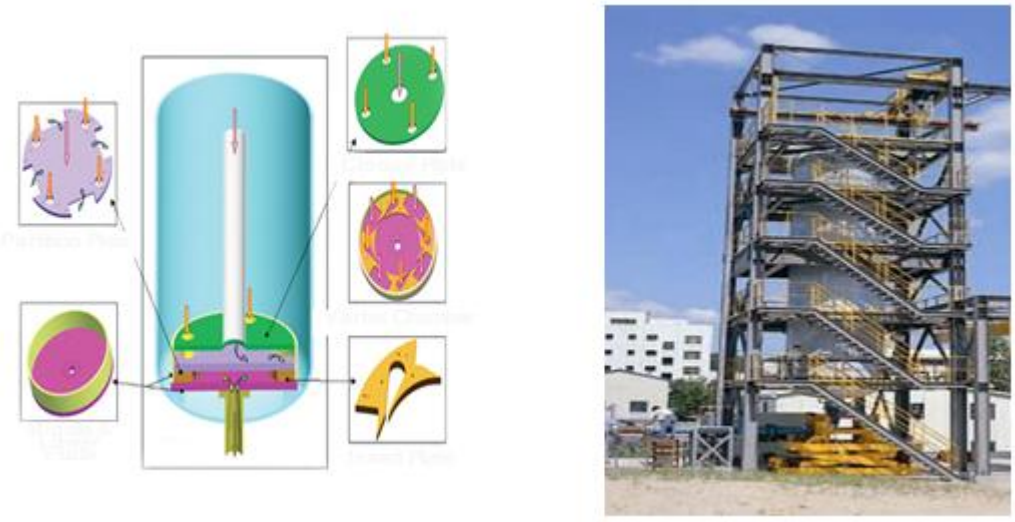


FIGURE 4.4.7. Valve Performance Evaluation Rig (VAPER)

- Direct Vessel Injection (DVI)
  The DVI test facility examines / measures the ECC bypass in reflood phase of cold-leg large break LOCA. The test facility is $^{1}/_{5}$ of the linear scale. It has steam-water (superheated and saturated steam) test capability. It is used to perform multi-dimensional investigation in Downcomer Annulus Simulation.

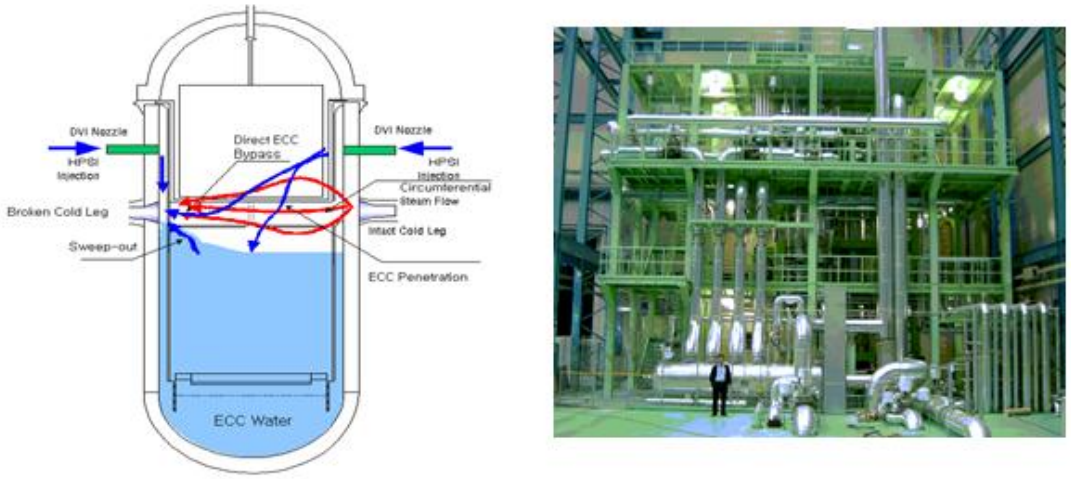


FIGURE 4.4.8. Direct Vessel Injection (DVI) of APR1400

- IRWST and Sparger
  The test loop is used to measure air, steam and water blowdown load on the SDVS and IRWST structure, to investigate thermal mixing between discharged fluid and water in the IRWST.

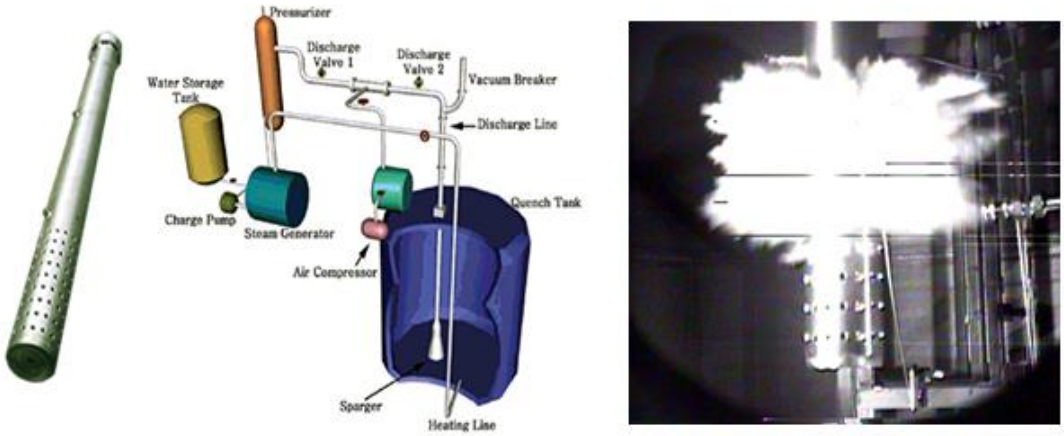


FIGURE 4.4.9. IRWST and Sparger

- Integral test loop of APR1400 – ATLAS
  The design with high power of 4,000 MWth requires integrated test facility. The test loop called ATLAS is a state-of-the-art integral loop test facility, scaled ½ of the height and 1/144 of the area of APR1400. It simulates RV, SG and all major equipment in the reactor to test all major DBAs to verify the design and also the design code. The test data was useful for code benchmarking.

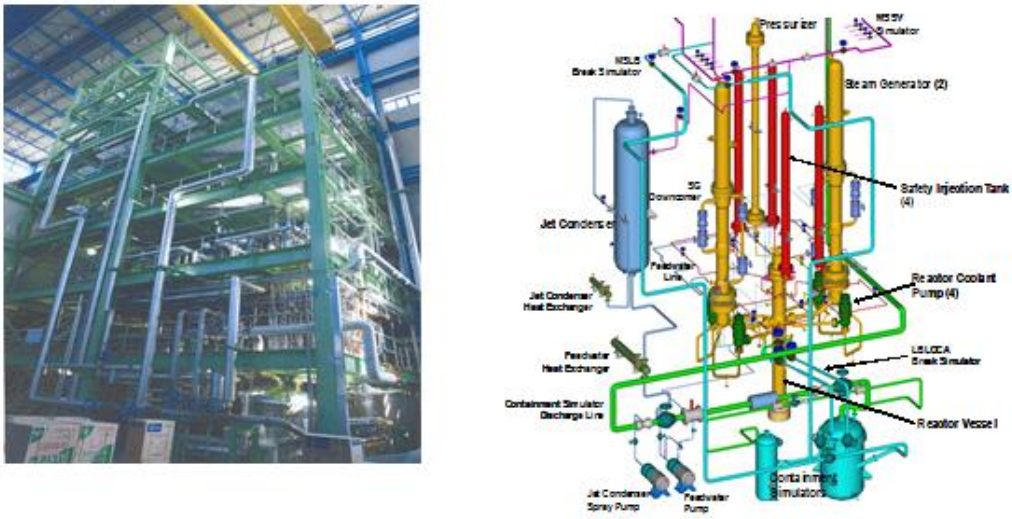


FIGURE 4.4.10. Integral test loop of APR1400 – ATLAS

Finally, Mr. Seo presented the APR1400 construction project status and future plans as follows:

- The first-of-a-kind (FOAK) contruction project of APR1400 is the Shin Kori 3&4 in Korea. The milestone schedule for Unit 3 was detailed, starting from site grading in September 2007 to excavation and construction permit (CP) in 2008, then the first concrete pouring, reactor vessel setting, cold hydro test, hot functional test, and fuel loading (awaiting Operating License (OL)) and commercial operation tentatively in September 2014.

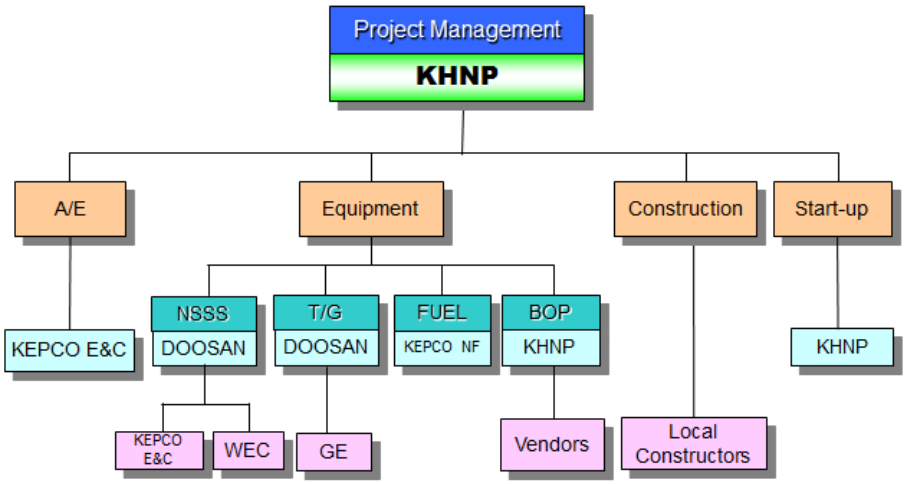


FIGURE 4.4.11. The project structure for Shin Kori 3&4 – the APR1400 FOAK

- Shin-Hanul 1&2 construction permit was obtained in December 2011, the first concrete for Unit 1 was in July 2012, and commercial operation for Unit 1 and 2 is expected by April 2017 and April 2018 respectively.
- Barakah 1-4 in United Arab Emirates (UAE) with PSAR submitted in December 2010, construction permit obtained in July 2012, the first concrete for Unit 2 in May 2013, and expected commercial operation by May 2017-2020.
- Shin-Kori 5&6 with commercial operation fixed by 2019 and 2020 respectively.
- Shin-Hanul 3&4 is under contract negotiation but with the targeted commercial operation by 2020 and 2021 respectively.

Mr Seo also indicated that APR1400 is modified for EU countries according to the EU requirements, and the US NRC design certification is being sought.

4.4.2. INPRO assessment in the area of safety of APR1400

4.4.2.1. CR1.2.1 I&C and Inherent Characteristics

Mr Seo (KEPCO-E&C) noted the criterion asks for preventing any event from progressing to challenging reactor trips, i.e. preventing progression to DBEs; no event reporting necessary. This requires very sophisticated control and monitoring. There were two EPs discussed, continuous monitoring of the plant health and dynamic plant analysis. He emphasized the importance of dynamic analysis to be able to finely tune the control system to prevent reactor trips.

He briefly introduced the history of I&C hardware for the safety and non-safety systems, and noted the digital control and also the fact Korea already manufactures these systems.

TABLE 4.4.2. I&C Platform History

Systems / Plants	Reactor Trip System	ESFAS Systems	Protection Process	NSSS Control	PCS	TBN Control	Main Control
Kori 1	Relay Logic (WEC)	Relay Logic (WEC)	Foxboro H-line	Foxboro H-line	Foxboro H-line	DCS	Conventional
Kori 1 (Upgrade)	Relay Logic (WEC)	Relay Logic (WEC)	Spec200 Spec200m (Foxboro)	Spec200 Spec200m (Foxboro)	Spec200 Spec200m (Foxboro)	DCS	Conventional
Kori 2,3,4 YGN 1,2	SSPS Relay Logic (WEC)	SSPS Relay Logic (WEC)	7300 Analog	7300 Analog	7300 Analog	Mark V (GE)	Conventional
YGN 3,4	Relay Logic (ABB-CE)	Relay Logic (ABB-CE)	Spec200 (Foxboro)	Spec200 Spec200m (Foxboro)	ILS (Forney)	Mark V (GE)	Conventional
Ulchin 3,4 YGN 5,6	Relay Logic (ABB-CE)	Relay Logic (ABB-CE)	Spec200 (Foxboro)	Spec200m/PLC (Foxboro)(Omron)	PCS (Eaton)	Mark V (GE)	Hybrid
Wolsong 1,2,3,4	Relay Logic (AECL)	Relay Logic (AECL)	Analog/PDC (AECL)	DCC X/Y (Computers Control)	Analog/Relay (AECL)	Mark V (GE)	Hybrid
Ulchin 5,6	PLC (WEC)	PLC (WEC)	Spec200 (Foxboro)	PLC (Omron)	AFS100/ECS1200 (HFC)	Mark V (GE)	Hybrid
Shin Kori 1,2 Shin Wolsong 1,2	PLC (WEC)	PLC (WEC)	Spec200 (Foxboro)	PLC (Omron)	HFC6000 (HFC)	Mark VI (GE)	Hybrid
Shin Kori 3,4 (APR1400)	PLC (WEC)	PLC (WEC)	PLC (WEC)	DCS (WEC)	DCS (WEC)	Mark VI (GE)	Compact Workstation
Shin Ulchin 1,2 (APR1400)	PLC (DOOSAN)	PLC (DOOSAN)	PLC (DOOSAN)	DCS (DOOSAN)	DCS (DOOSAN)	Mark VI (GE)	Compact Workstation

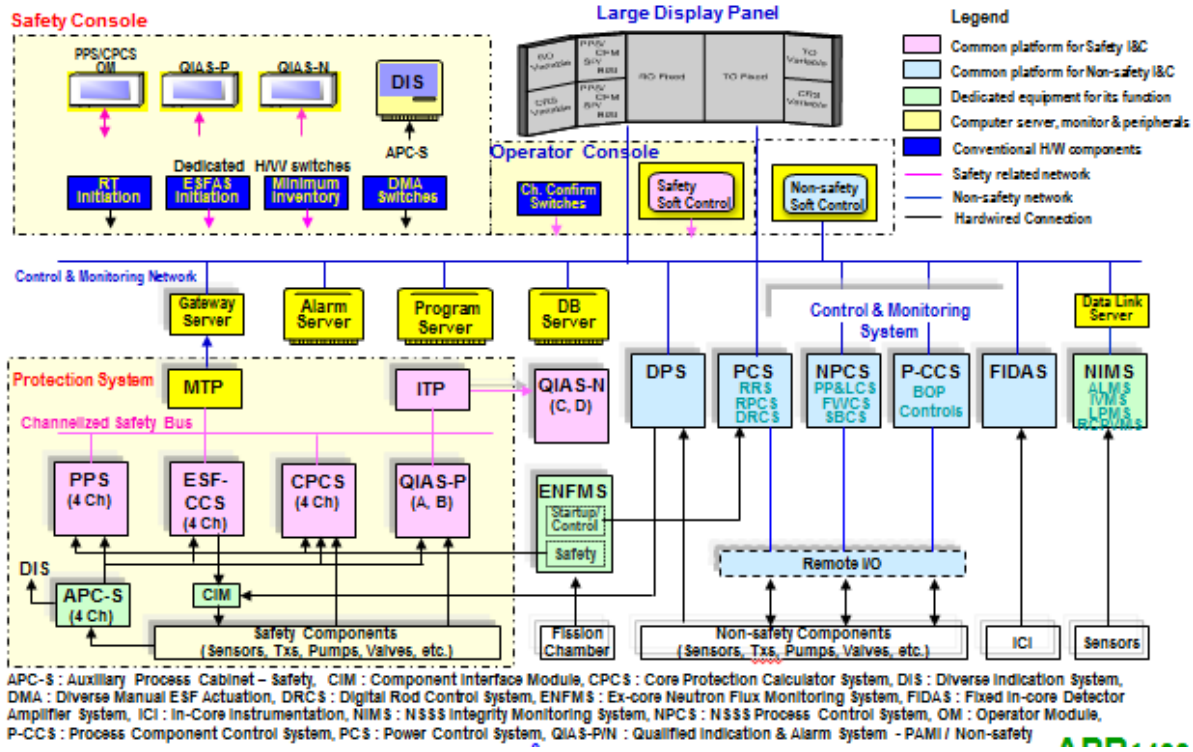


FIGURE 4.4.12. I&C System Overview of APR1400

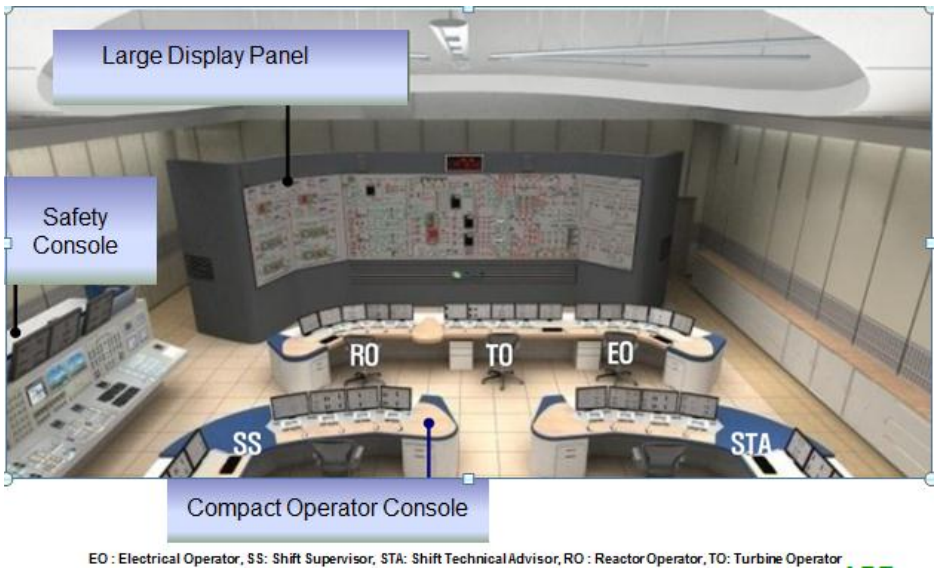


FIGURE 4.4.13. The Main Control Room of APR1400

The following are I&C for APR1400 design:

- Microprocessor based digital I&C technology
  - common PLC platform for safety I&C
  - common DCS platform for non-safety I&C
- Four (4) channel redundancy for safety I&C
  - installed in separate channelized I&C equipment room.
- Multiplexed I/O and redundant communication network
  - from field areas to I&C equipment rooms and MCR
- Diverse Actuation System to cope with the CMF (common mode failure) of safety I&C
- Complete electrical, physical and communication isolation (according to regulatory requirements)
  - between redundant safety channels
  - between safety system and non-safety system
- Protection System
  - Plant Protection System (PPS)
  - Engineered Safety Features-Component Control System (ESF-CCS)
    This includes sophisticated DNBR calculation.
  - Core Protection Calculator System (CPCS)
  - Auxiliary Process Cabinet – Safety (APC-S)
- Diverse Actuation System
  - Diverse Protection System (DPS)
  - Diverse Indication System (DIS)
  - Diverse Manual ESF Actuation Control Switches (DMA switches)
- Control & Monitoring System
  - Power Control System (PCS)
  - NSSS Process Control System (NPCS)
  - Process-Component Control System (P-CCS)
  - Qualified Indication and Alarm System (QIAS) - PAMI
  - Qualified Indication and Alarm System (QIAS) – Non-Safety
  - Neutron Flux Monitoring System (ENFMS / FIDAS)
  - NSSS Integrity Monitoring System (ALMS / IVMS / LPMS / RCPVMS)
  - T/G Control & Monitoring System

For the plant health monitoring system, the APR1400 utilizes the NSSS Integrity Monitoring System (NIMS), the same with that for APWR. The Reactor coolant pump vibration, leak monitoring or loose part monitoring, or internal vibration are designed in NIMS. The NIMS hardware and software have been simplified compared to the OPR design.

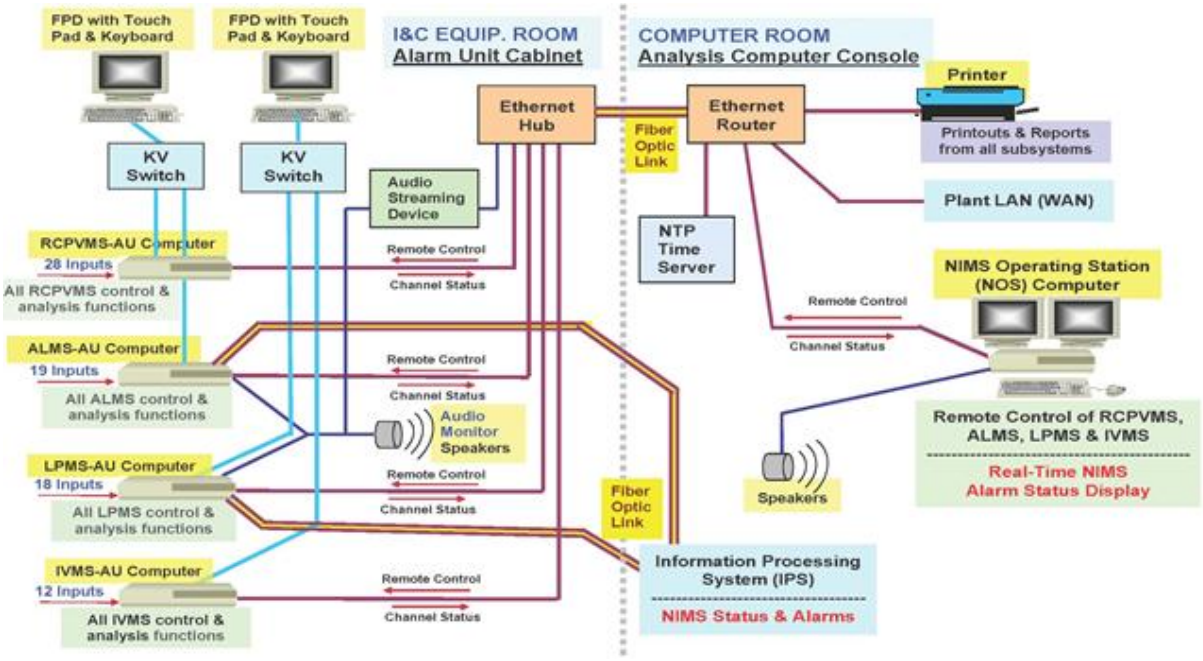


FIGURE 4.4.14. SKN3&4 NIMS Network Architecture

TABLE 4.4.3. NSSS Integrity Monitoring System (NIMS)

Items	OPR1000	APR1400
H/W	➢ PC: Alarm Unit, LPAP, DAD ➢ AlphaServer: LPMS-AC ➢ STD Bus: LPMS-AU	➢ Industrial PC (All Alarm Unit & NOS)
OS & S/W	➢ MS-DOS: LPMS-AU ➢ Win-98: LPAP ➢ Win-XP: RCPVMS/ALMS-AU, NOS/IVMS Computer ➢ OpenVMS: LPMS-AC	➢ OS : Windows ➢ Application : NI LabVIEW™ based S/W
Remote Control	All subsystem excluding LPMS	All subsystems
Cabinet	➢ 2 Cabinets : AUC & ACC ■ AUC : MCR ■ ACC : Computer Room	➢ 1 Cabinet : AUC (I&C E.R) ➢ 1 Console : ACC (Computer Room)

The plant control systems as shown below in the colored boxes are used to control any transients occurring within the power generating system from reactor to turbine side. All control systems are modelled with the best estimate methods. A Nuclear Plant Analyser (NPA) is utilized in the design phase to provide accurate results. It has also graphic interface. All transients can be simulated real time, best estimate wise so that the simulations results can be used in tuning the control systems.

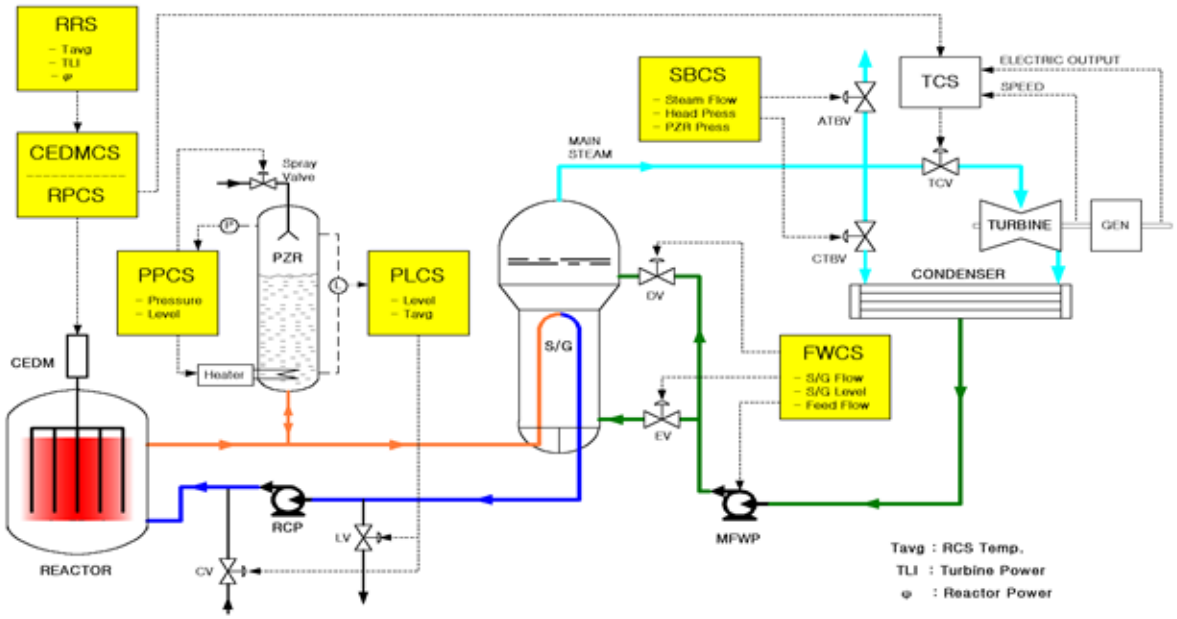


FIGURE 4.4.15. Overview of NSSS Control System

- Nuclear Plant Analyzer (NPA)
  - Win-NPA(Windows-based Nuclear Plant Analyzer)
    - Windows based desk-top engineering simulator
    - Engineering/training tool for operators/designers
  - Characteristics
    - Real-time, best-estimate simulation for NPP
    - Accurate, easy-to-use & easy-to-understand analysis tool
    - Dedicated tools for development & customization
    - Supports external interfaces (DB, I&C devices)

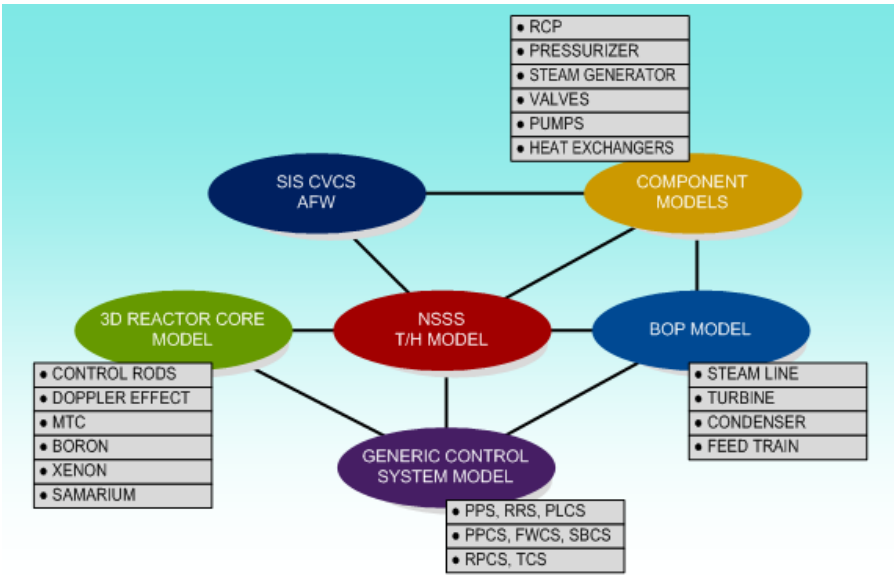
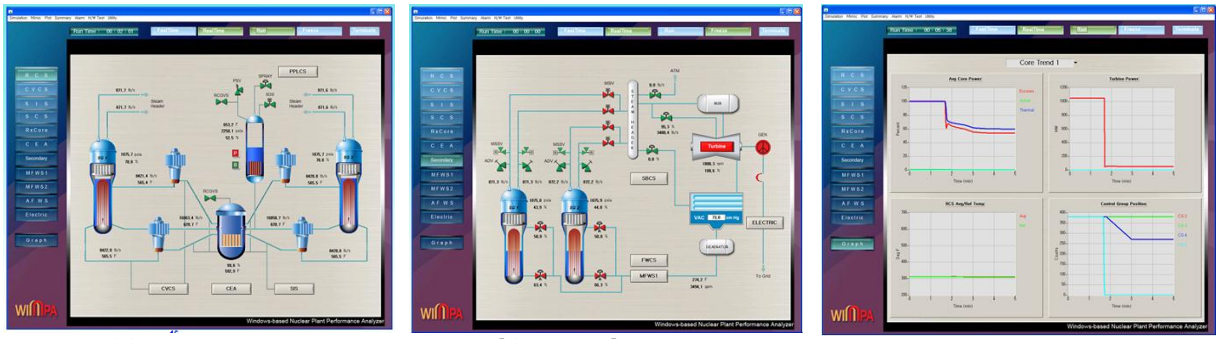


FIGURE 4.4.16. Nuclear Plant Analyzer (NPA) to simulate the control system



(a) RCS Mimic (b) Secondary Mimic (c) Plots

FIGURE 4.4.17. Graphic user interface in Nuclear Plant Analyzer (NPA)

4.4.2.2. EP1.2.1.1 Continuous monitoring of the plant health

The APR1400 design has advanced Plant Health Monitoring Capability over OPR1000 design by having full computerized I&C System and improved NSSS Integrity Monitoring System (NIMS). NIMS is based on the Windows OS and has application of NI LabVIEW™ based S/W. Other computerized plant monitoring systems are available.

4.4.2.3. EP1.2.1.2 Dynamic plant analysis

The APR1400 utilizes advanced Nuclear Plant Analyzer (NPA) for dynamic plant analysis. It is a Window based desk-top engineering simulator and has the characteristics of real-time and best-estimate simulation for NPP transients under normal, abnormal and accident conditions. Two examples of dynamic plant analysis were provided. On load rejection, it is shown that the reactor can continue house load operation without reactor trips. In the case of the other major transient which is loss of one main feedwater pump (pump trip), the reactor can continue to operate with reduced power level. This is made possible given the integrated manner of the control systems and dynamic plant analysis capability.

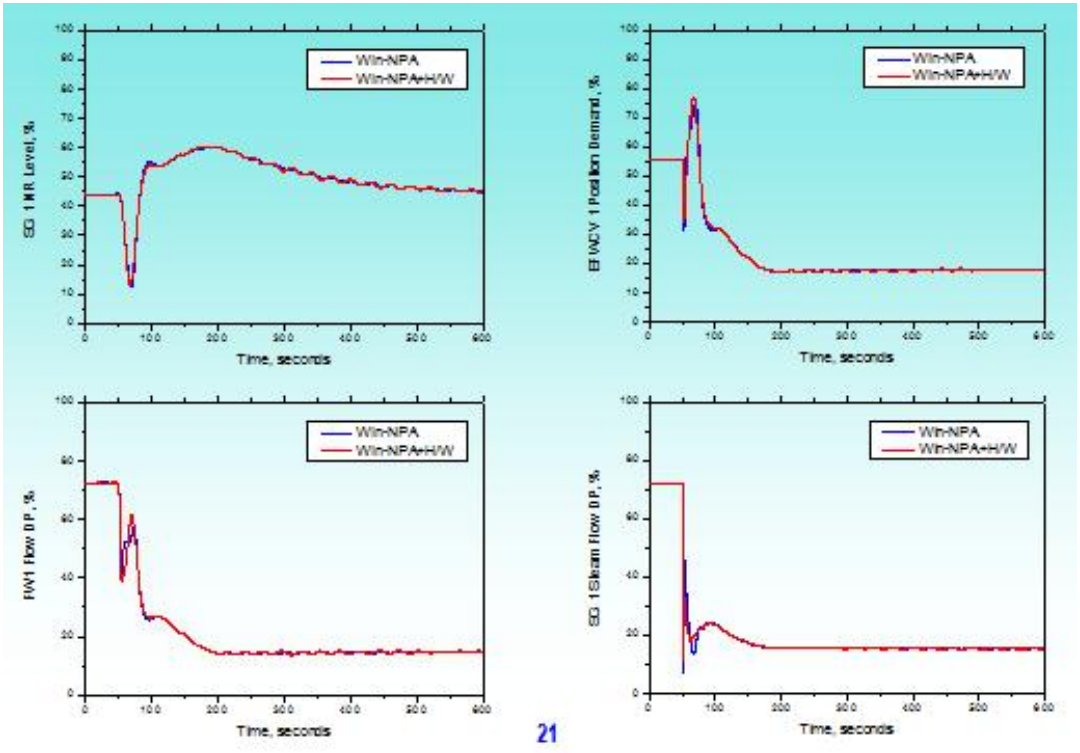


FIGURE 4.4.18. Dynamic plant analysis in the event of load rejection to house load

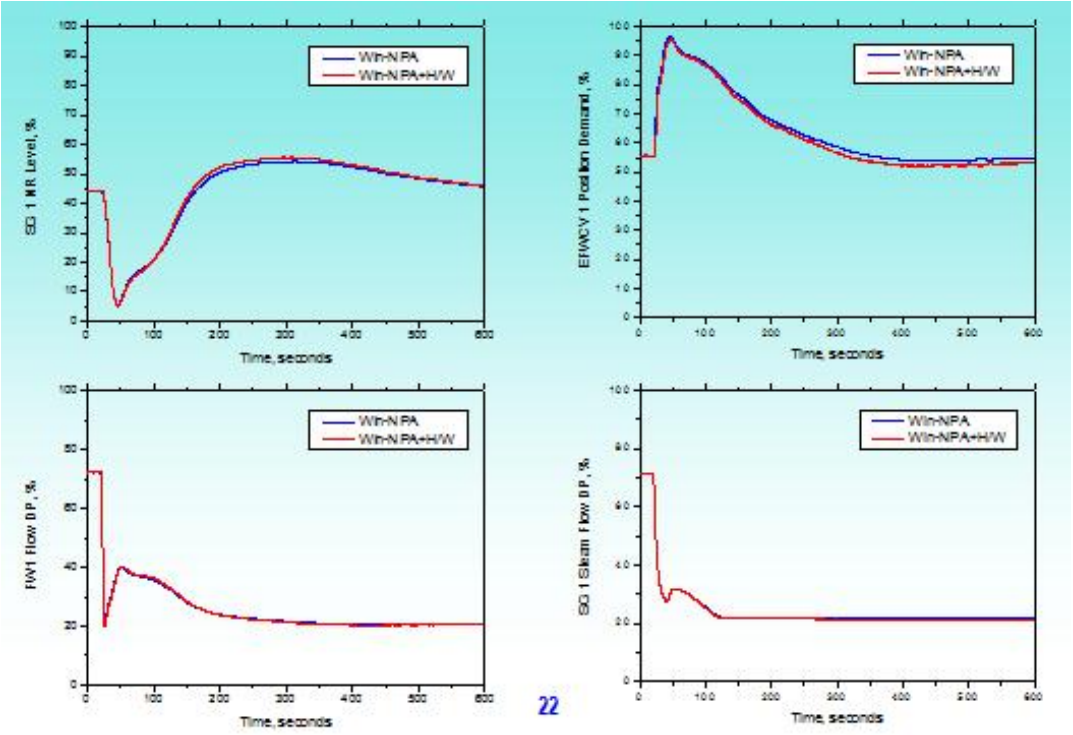


FIGURE 4.4.19. Dynamic plant analysis in the event of loss of a main feedwater pump

In conclusion, Mr Seo stated that the APR1400 I&C utilize full-digital technology and is designed to detect and intercept deviations from normal operational states. The integrated plant monitoring system and control systems prevents unnecessary safety system actuations. Dynamic simulation tool was utilized to verify APR1400 design capabilities. For a full loss of load and loss of a main feedwater pump events, all safety related variables can be controlled within the limit allowing a continued operation without event reporting. Therefore, APR1400 satisfies CR 1.2.1 I&C and Inherent Characteristics.

4.4.2.4. CR1.3.3 Reliability of Engineered Safety Features

Mr Seo (KEPCO-E&C) pointed out the safety system reliability characteristics as stated in the INPRO Methodology:

- Robust Reactor Protection System (RPS) design
- Common cause failure consideration
- Redundant and diverse safety system
- Passive design features
- Single failure consideration

He then discussed the Engineered Safety Feature (ESF) characteristics in the APR1400 design. The ESF systems are provided to mitigate the consequences of the unlikely events such as large break LOCA. The ESF systems control, mitigate, and terminate such incidents and hold exposure levels below applicable limits. The ESF systems include as follow and discussed hereafter.

- Emergency Core Cooling System (ECCS): Safety injection system and Shutdown cooling system
- Containment System: Containment Spray System and Containment Isolation System
- Auxiliary Feed water System
- Safety Depressurization System

1. Safety Injection System

   The safety injection system comprises 4-trains, each with 1 pump and 1 SIT, and inject water directly to the reactor vessel (DVI). The suction is in the IRWST. As shown in the figures, the IRWST is located at the bottom of containment. The DVI nozzles are located above hot and cold leg elevation. The SIT has a passive flow control device called the Fluidic Device to regulate the injection flow rate effectively and extend the injection time of SIT. Hence the Low Pressure Safety Injection (LPSI) function of SCS is removed. The flow rate depends on the Stand Pipe height and resistance of Supply Port and Control Port.

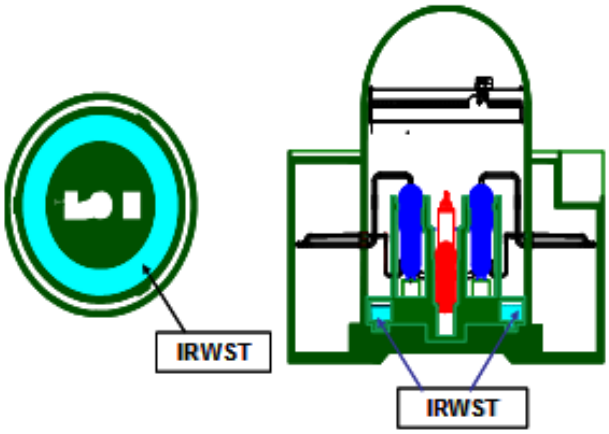


FIGURE 4.4.20. In-containment Refuelling Water Storage Tank (IRWST)

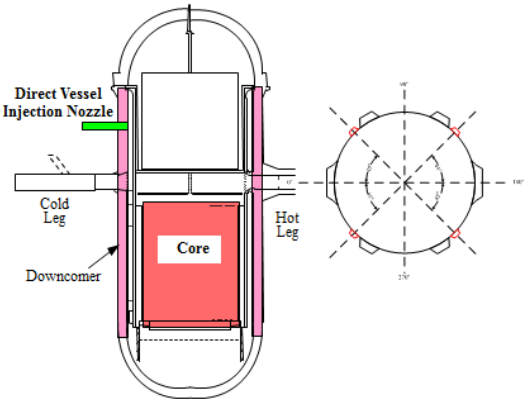


FIGURE 4.4.21. Direct Vessel Injection (DVI)

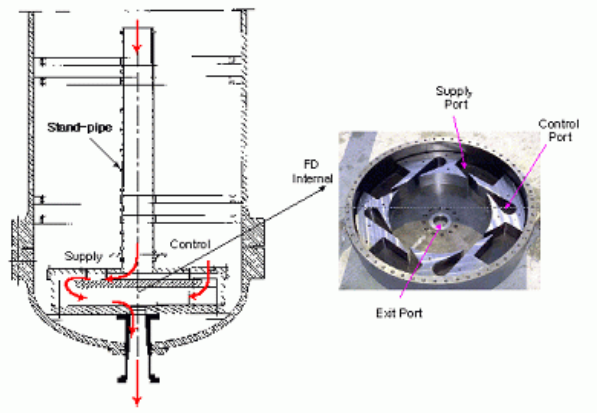


FIGURE 4.4.22. Passive Fluidic Device in Safety Injection Tank (SIT)

2. Shutdown Cooling System

The system removes core decay heat during shutdown, i.e. reduce RCS temperature in post shutdown period from hot shutdown operating temperature to refuelling temperature and cool down RCS to cold safe shutdown condition following SBLOCA, SLB and FWLB. It was noted that there is not much different in the shutdown cooling system for both APR1400 and OPR1000 design.

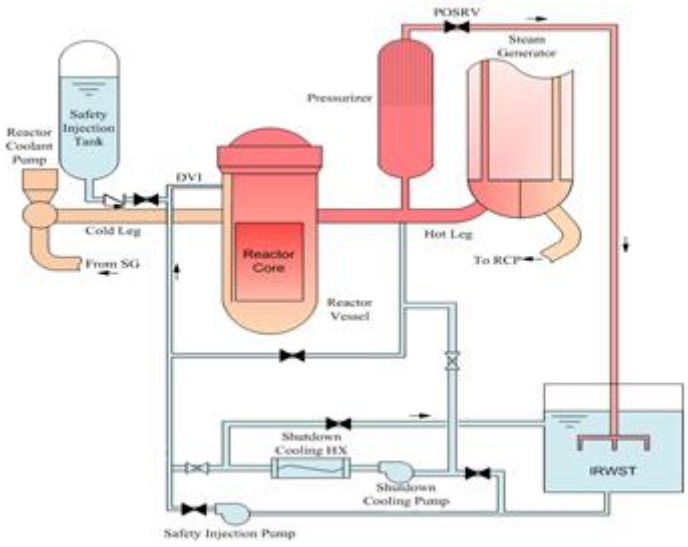


FIGURE 4.4.23. Shutdown Cooling System

3. Safety Depressurization System

The Pilot Operated Safety Relief Valve (POSRV) performs over-pressure protection and safety depressurization (during design basis transient). The main valve is opened by pilot valve actuation, so there will be no drift of the opening set-point. The valve operation is known to be reliable without chattering and leakage, and also it has low susceptibility for valve stuck-open.

4. Containment Spray System

The containment spray system is automatically actuated by the Containment Spray Actuation Signal (CSAS). Upon SIAS (Safety Injection Actuation Signal) or CSAS, the CS (Containment Spray) pumps start taking suction from the IRWST, and the containment spray header isolation valves open on CSAS, and spray water flows into the containment spray headers. It was noted that the shutdown cooling pump which is identical to the CS pump, and once CS pump becomes unavailable then it takes suction from the IRWST and pass it to the spray nozzle.

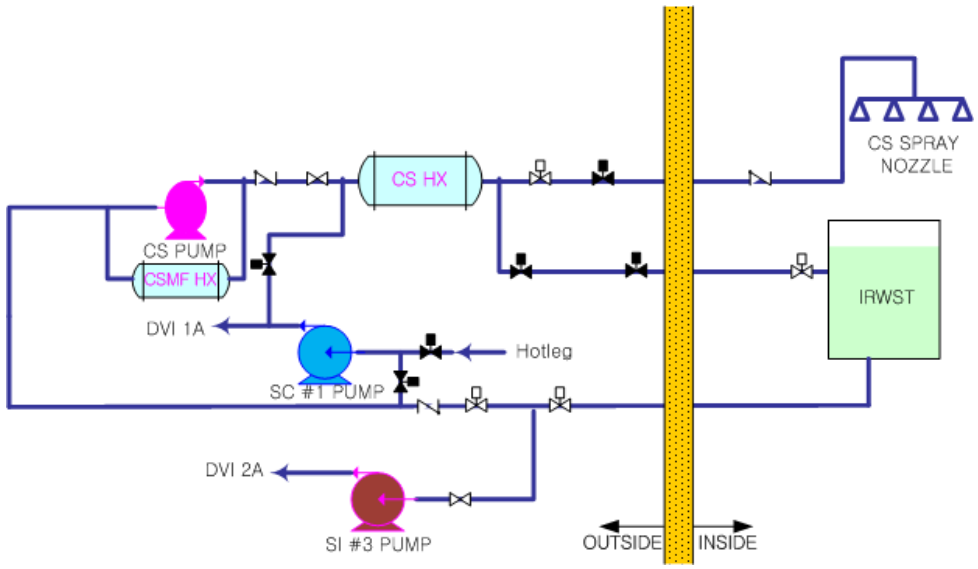


FIGURE 4.4.24. Containment Spray System

5. Containment Isolation System

The containment isolation system is a means of isolating non-safety related fluid systems that pass through containment to confine any radioactivity following a design bases event. It has a double barrier with automatic & leak-tight closure. There are leak testing and periodic operability test performed. It was noted that the containment isolation system is not much different between APR1400 and OPR1000 design.

6. Auxiliary Feed water System

The auxiliary feed water system has been improved in the AR1400 design. There are 4-Trains with 100% capacity for each train. Redundancy and diversity has been incorporated, e.g. for one SG there are 1 turbine-driven pump and 1 motor-driven pump in separate lines, taking suction from auxiliary feedwater storage tank into the SG. There is also physical and electrical independency.

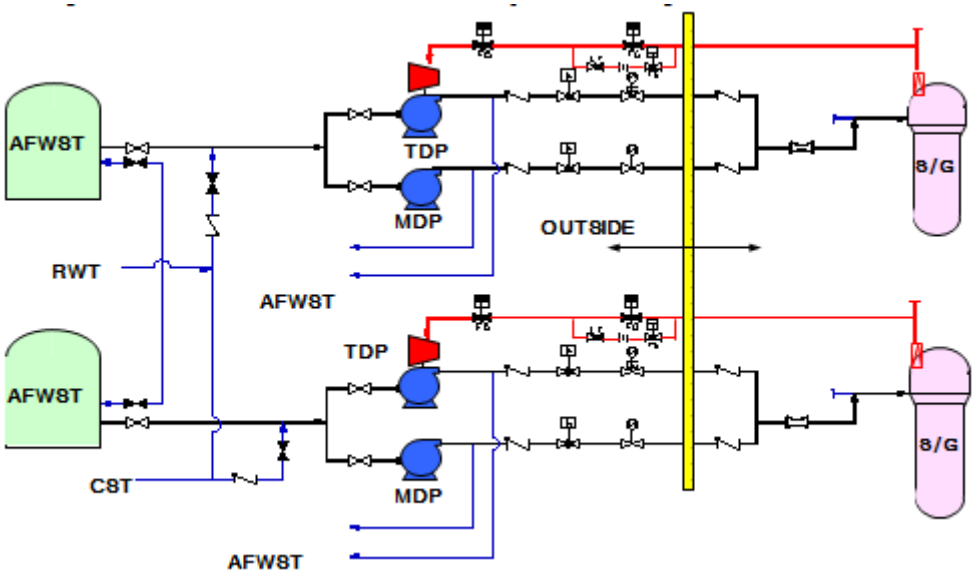


FIGURE 4.4.25. The 4-Train Auxiliary Feed water Systems (AFWS)

The Plant Protection System (PPS) initiates reactor trip and/or the ESF actuation whenever the monitored process values exceed the pre-defined limits. In the case the PPS generates reactor trip signal, the signal is passed to the reactor trip switch gear system (RTSS) which will interrupt power supply to control rods. In the case the PPS generates ESF Actuation System (ESFAS) signal, the signal is passed to the ESF-Component Control System (ESF-CCS) group control (GC) via Control Panel Multiplexer (CPM). The ESF-CCS controls the operation of the ESF components such as valves and pumps. The Core Protection Calculator System (CPCS) computes the DNBR (Departure from Nucleate Boiling Ratio) and LPD (local power density) value using process values such as RCP speed, CEA position and ex-core neutron flux. There is also the Auxiliary Process Cabinet-Safety (APC-S) which receives safety field sensor signals and distributes them to PPS, ESF-CCS, CPCS QIAS-P and DIS.

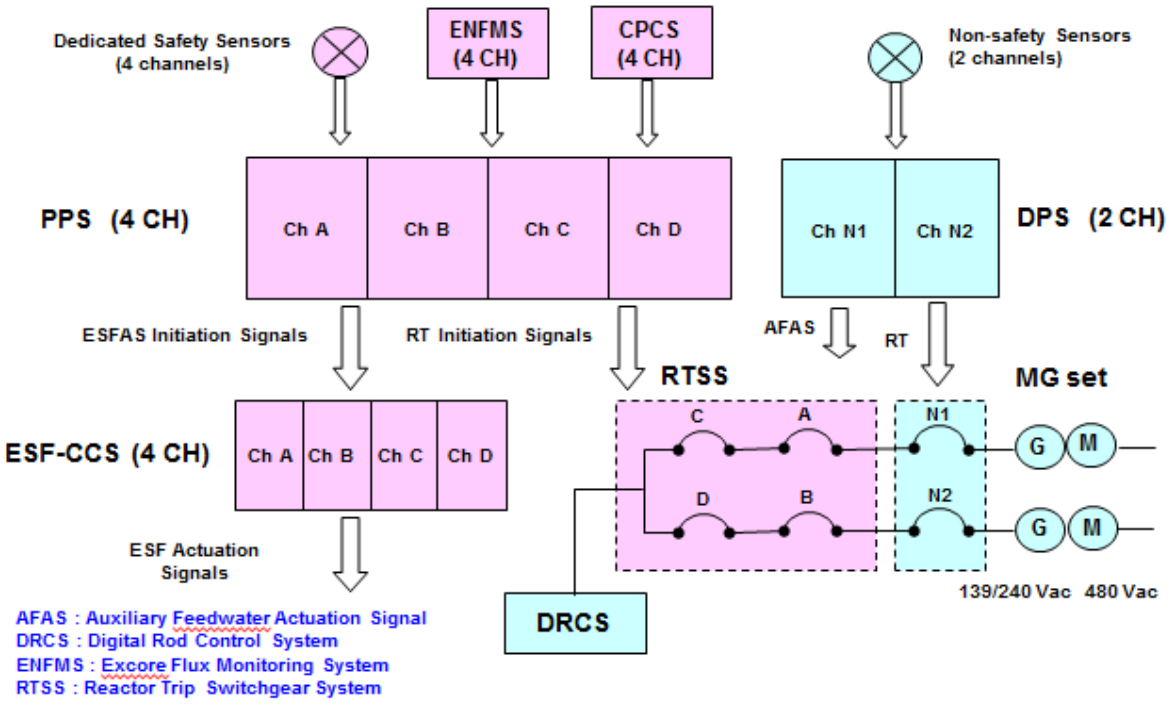


FIGURE 4.4.26. Plant Protection System (PPS)

Diverse Actuation System

- Diverse Indication System (DIS) – in the safety console
- Diverse Manual ESF Actuation Control Switches (DMA switches) – in the safety console
- Diverse Protection System (DPS)

The DPS is a non-safety system as a backup for safety grade plant protection system. It has two channels, opens separate breaker in the M-G set. The DPS uses separate inputs as well as actuates separate trip breakers.

The DPS itself was designed for the ATWS mitigation required by 10CFR50.62 and provides capability to cope with common mode failure (CMF) of the digital PPS/ESF-CCS (SECY 93-08). It has energized-to-actuate design. On the loss of power, the DPS fails to non-actuated conditions. So the DPS is diverse equipment from the PPS, from sensor output to final actuation device. There are 2 DPS controllers used: 2/2 coincidence logic and 2/2 power interruption scheme to prevent inadvertent actuation on single failure of one (1) DPS channel.

For INPRO evaluation on ESF improvements, Mr Seo (KEPCO-E&C) pointed out 6 major advanced design features of the APR1400 as compared to the OPR1000 as follow:

- 4-Train high pressure Safety Injection System and no low pressure Safety Injection System.
- Direct Vessel Injection
- In-containment Refuelling Water Storage Tank (IRWST)
- Safety Injection Tank with Fluidic Device
- Simplified Safety Depressurization System with POSRV
- Interchangeability of Containment Spray and Shutdown Cooling Pumps.

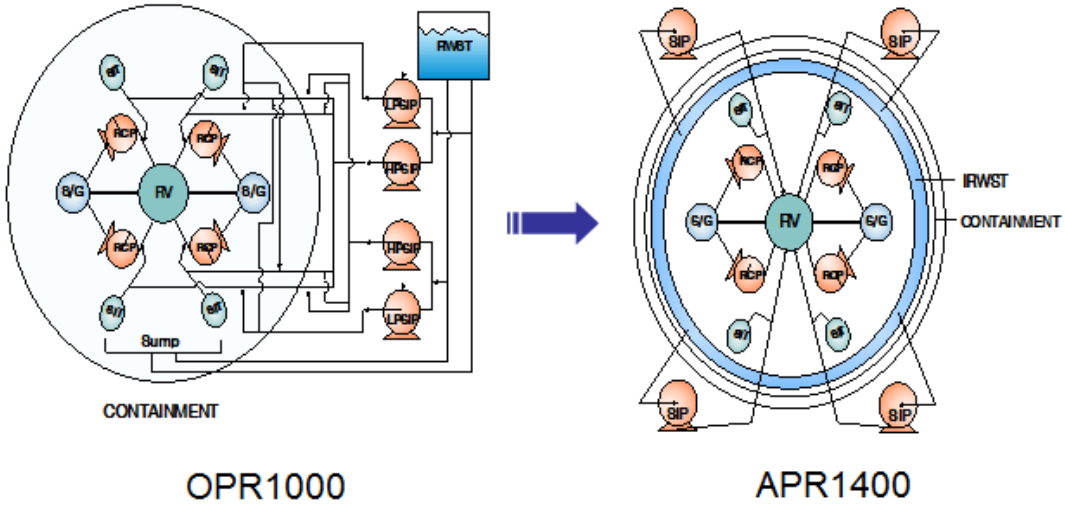


FIGURE 4.4.27. Comparison of 4-train system in APR1400 versus OPR1000

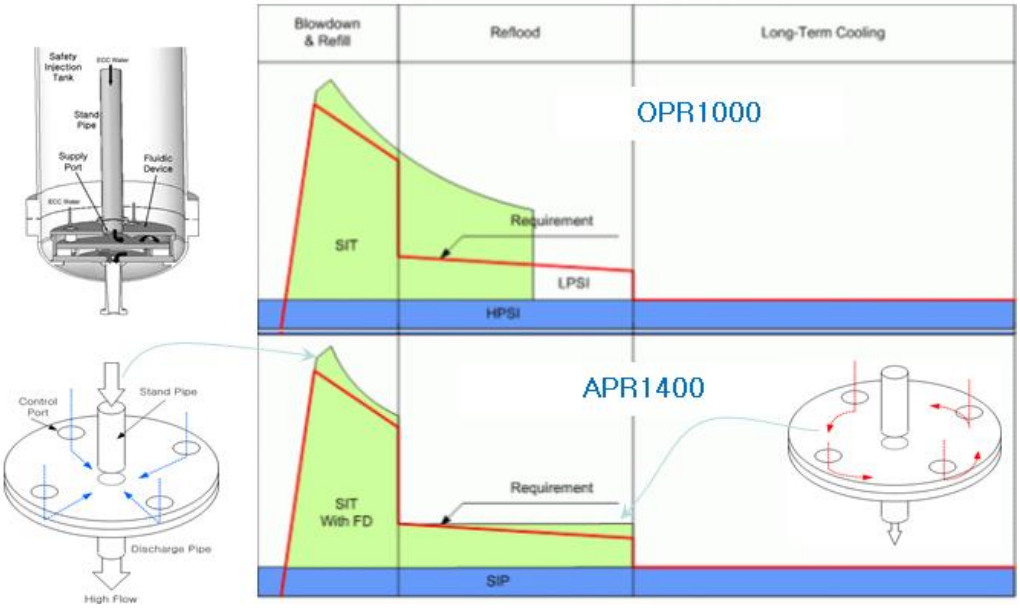


FIGURE 4.4.28. Comparison of the Passive Fluidic Device in APR1400 versus OPR1000

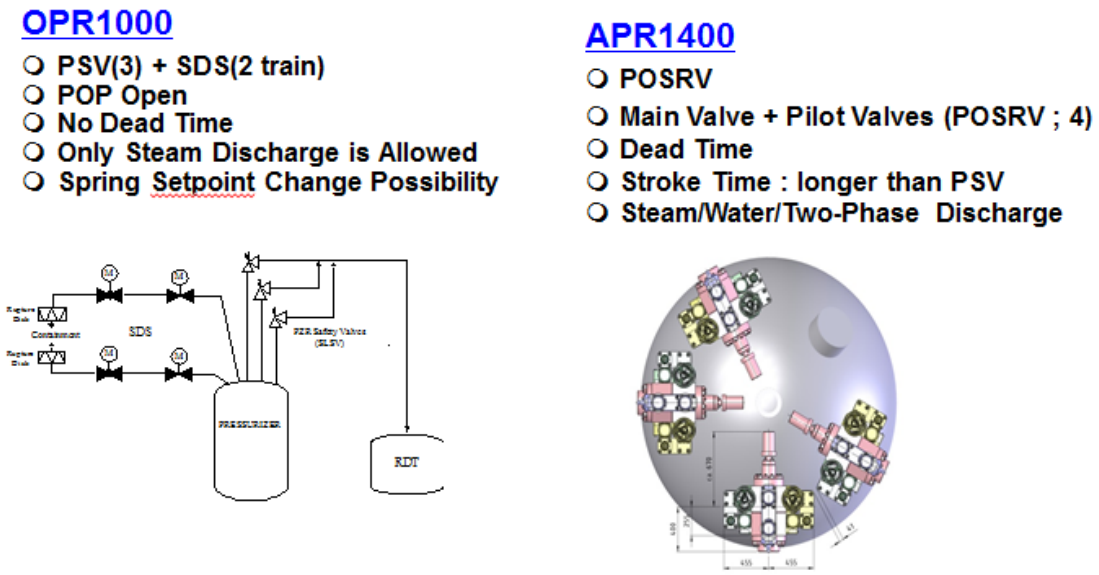


FIGURE 4.4.29. Comparison of the safety depressurization system in APR1400 versus in OPR1000

In addition, improvements in the PPS include (i) full digital with data network for the protection system, i.e. PPS, ESF-CCS, CPCS and APC-S, and (ii) provision to cope with CMF in digital PPS for Diverse Actuation System, i.e. DIS, DMA switches and DPS.

In conclusion, Mr Seo (KEPCO-E&C) made the following points. The APR1400 ESF has been dramatically improved as compared to the reference plant OPR1000, by means of simplifying system configuration with passive and reliable components. A lot of verification tests have been performed. In addition, the digital Plant Protection System (PPS) provides reliable actuation of reactor trip and ESF actuation with CMF provisions. Therefore, the APR1400 satisfies CR1.3.3 “Reliability of Engineered Safety Features”.

4.4.2.5. CR1.4.3 In-Plant Severe Accident Management

Mr Seo (KEPCO-E&C) pointed out that INPRO criterion refers to DID Level 4. The accident management actions should include injection of boron into the core, depressurization of RPV, restoration of water injection into RPV, restoration of heat removal, spraying into the containment atmosphere, and containment venting. The accident management procedure should also include demonstration of feasibility and effectiveness and sufficient training. He noted the training is not designer’s aspect, but utility’s scope.

Severe accident design characteristic

The APR1400 is designed to cope with severe accident phenomena: over pressurization, high pressure melt ejection, direct containment heating, steam explosion, molten corium concrete interaction and hydrogen mixing and combustion.

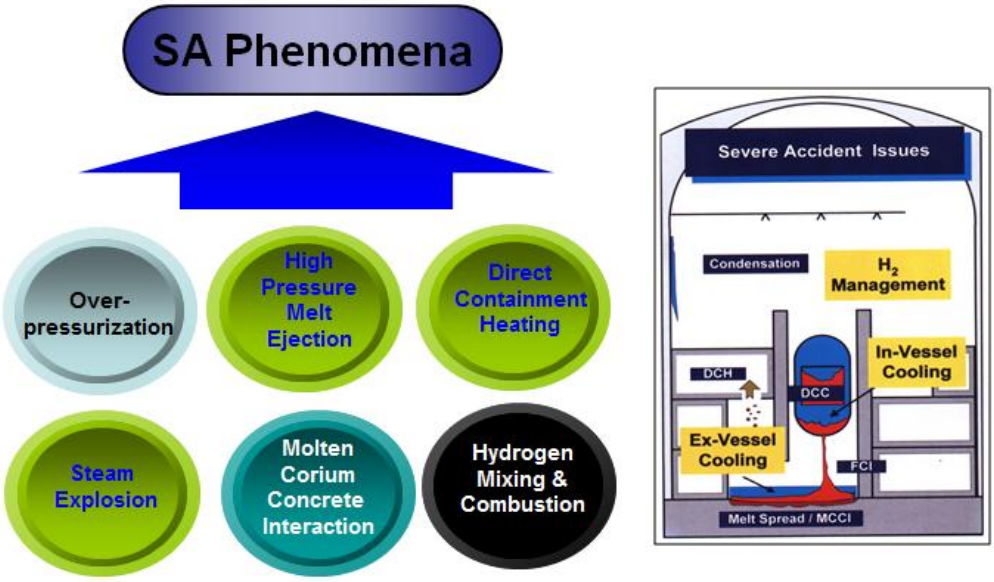


FIGURE 4.4.30. Severe Accident Phenomena

The Severe Accident Mitigation design features that can be used for severe accident phenomena:

- Containment with IRWST and HVT (Hold-up Volume Tank)
- Safety Depressurization with POSRV and Vent System
- Reactor Cavity Design
- Convoluted Flow Path
- Cavity Flooding System (CFS)
- In-Vessel Retention (IVR) / External Reactor Vessel Cooling (ERVC)
- Hydrogen Mitigation System
- Containment Spray (CS) System and Emergency Containment Spray Back-up System (ECSBS)

1. Containment design

   The APR1400 containment is a large dry containment with pre-stressed concrete structure, with steel-lined inner surface for leak tightness. The height is 229.5 ft, diameter 150 ft, thickness 4 ft and design pressure 60 psig. The IRWST eliminates switch-over operation during LOCA, that is, all spillage will be eventually collected down to the IRWST so it can continue injection mode operation. It serves as heat sink for feed and bleed operation in case of total loss of a feed water flow event.

2. Rapid depressurization

   Rapid depressurization can be achieved by using the POSRV. It depressurizes the RCS below 250 psia (1.7 MPa) before the reactor vessel rupture to prevent High Pressure Melt Injection (HPME) and Direct Containment Heating (DCH). In total there are 4 POSRV and two 3-way valves. During severe accident, there is a way to directly discharge the steam of pressurizer (RCS) to the SG compartment, i.e. to the containment atmosphere, using the 3-way valves in the discharge lines of the POSRV.

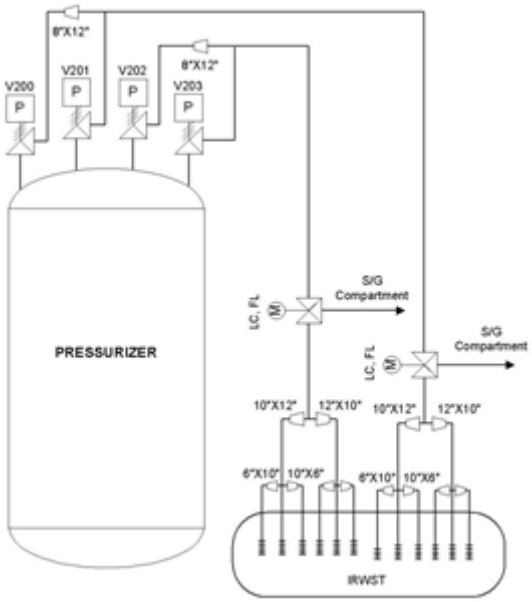


FIGURE 4.4.31. Rapid depressurization using POSRV

3. Reactor Cavity Design and Convoluted Path for Corium control

   Reactor cavity design has enough spreading area for the molten core. The corium has to travel through the convoluted flow path before it reaches the containment atmosphere.

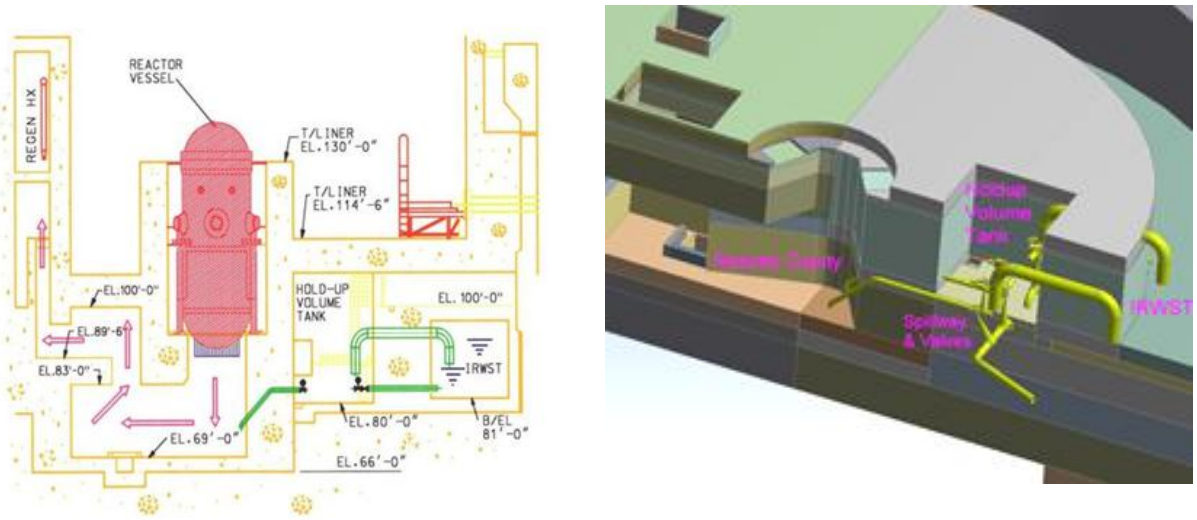


FIGURE 4.4.32. Reactor cavity design and Convoluted Flow Path for molten core

4. Reactor Cavity Flooding System

   Flooding reactor cavity to cool molten core uses the water source from the IRWST by gravity force. The cavity floor area is >0.02 m^2/MWt. The cavity area design meets the SECY-93-087 (US NRC requirement).

5. The IVR / ERVC strategy

   As a provision for the severe accident management strategy, the APR1400 is designed to use the IVR (In-Vessel Retention) through the ERVC (External Reactor Vessel Cooling). In order to cool outside the reactor vessel, we have to fill the reactor cavity up to the RV flange. Then submerging reactor vessel lower head can be achieved to cool and to retain molten core in reactor vessel. The water source is the IRWST and the Shutdown Cooling Pump (SCP) and Boric Acid Makeup Pump (BAMP) are used to fill the reactor cavity.

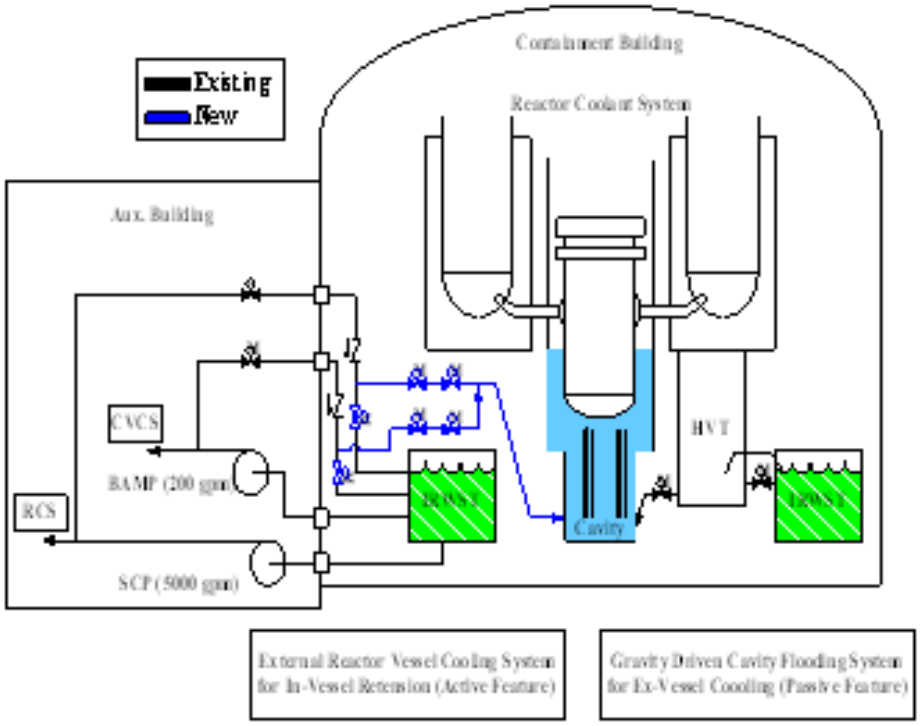


FIGURE 4.4.33. Reactor Cavity Flooding System and IVR/ERVC Strategy

6. Hydrogen mitigation System

   The function is to maintain hydrogen concentration below the design criterion. It employs 30 Passive Autocatalytic Recombiners (PARs) and 10 glow plug type igniters located throughout the containment.

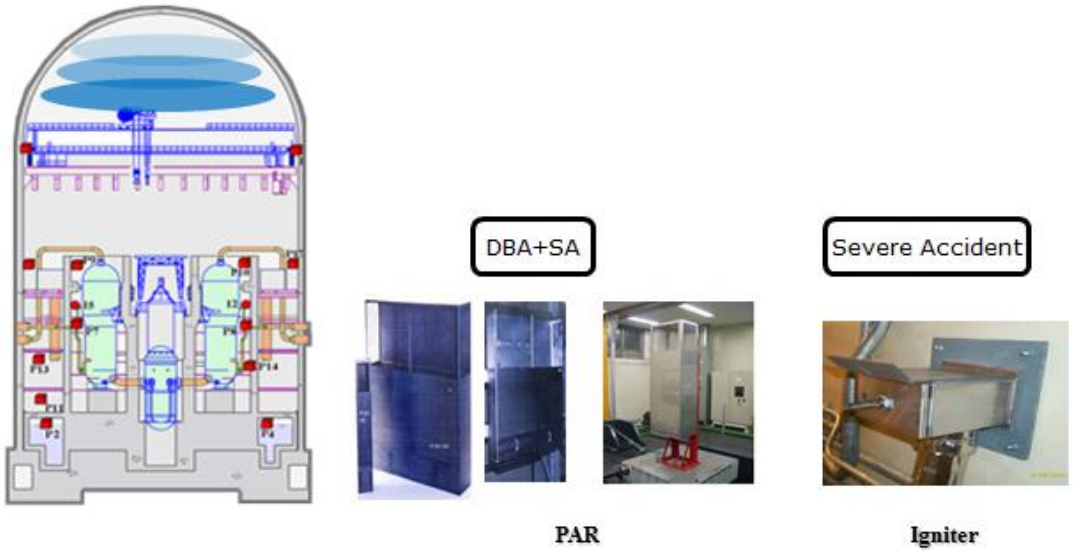


FIGURE 4.4.34. Containment Hydrogen Control System

7. Containment Spray (CS) and Emergency Containment Spray Backup System (ECSBS)

   Normal spray system is used to control containment pressure for DBAs. APR1400 design also incorporates the ECSBS that has a separate spray header and nozzles. In case of normal spray is unavailable, there are flanges outside the containment to hook up to fire engine truck pumps or mobile pumps to provide additional water from external water sources to spray to the containment. External water sources can be Reactor Makeup Water Tank, Demineralized Water Storage Tank and Fresh Water Tank, Raw Water Tank. They can be located away from the containment building.

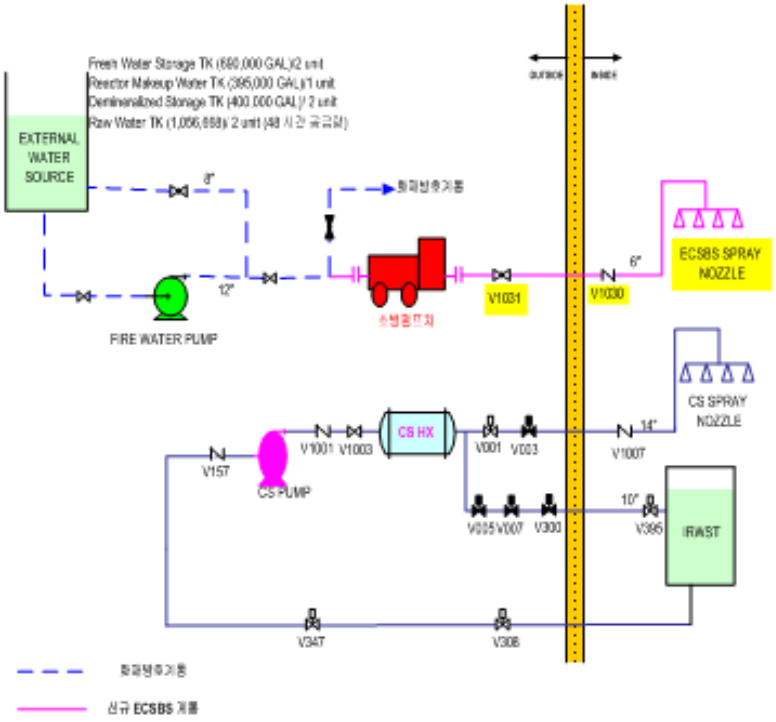
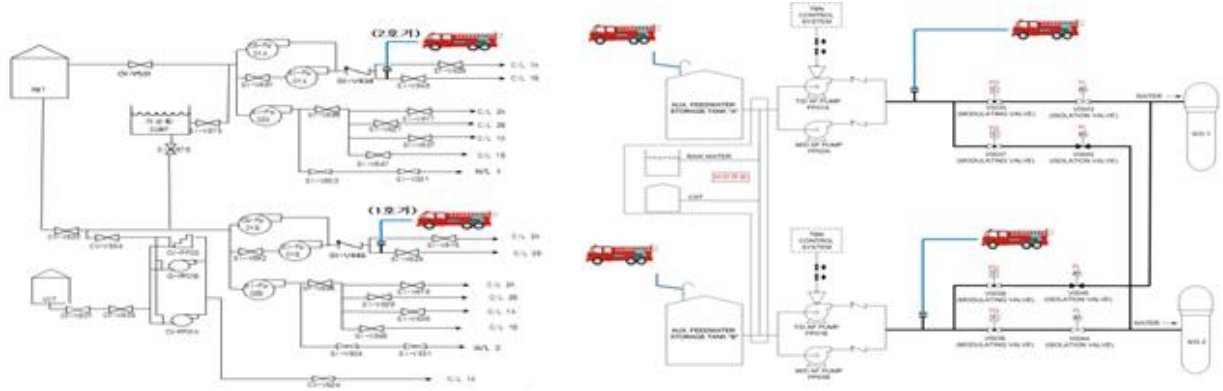


FIGURE 4.4.35. Emergency Containment Spray Backup System (ECSBS)

Emergency Make-up System

After Fukushima accident, the Korean regulatory body requires emergency make-up system, in case all design means of providing cooling water to primary and secondary systems is lost. In this case water from external sources can be delivered by fire engine truck. Emergency make-up provisions are also available for the Spent Fuel Pool.



(a) Primary loop injection make up system connection.

(b) Secondary loop injection make-up system connection

FIGURE 4.4.36. Emergency Make-up System for Primary and Secondary System

Severe Accident Management Guideline (SAMG)

The APR1400 SAMGs are developed for plant specific application, reflecting the APR1400 design features for the severe accident prevention/mitigation as well as all available instrumentation for plant monitoring. The SAMGs consist of 1 Entry Guideline, 7 Recovery Guidelines, 1 Exit Guideline and 2 Supporting Guidelines. The 7 Recovery Guidelines consist of:

1. Reactor Coolant System Depressurization
2. Steam Generator Feeding
3. Reactor Coolant System Injection
4. Reactor Cavity Flooding

5. Fission Product Release Control
6. Reactor Containment Control
7. Reactor Containment Hydrogen Control

Each recovery guideline is supported by the corresponding Technical Basis Document.

TABLE 4.4.4. APR1400 Severe Accident Management Guideline (SAMG): Recovery Guidelines

No	Title	Action	Purpose	System and Components
1	RCS Depressurization	RCS direct depressurization	• Prevention of direct containment heating • Containment integrity • Core cooling	• SDS • Pressurizer auxiliary spray
		SG depressurization		• Main steam ADVs
2	SG Feeding	SG feed water injection	• Heat sink for RCS • SG integrity • Prevention of containment bypass • Reduction of fission product release	• Any feedwater pumps
		SG depressurization		• Main steam ADVs
3	RCS Injection	Water injection into RCS	• Core cooling • Prevention of reactor vessel failure • Reduction of fission product release	• SIPs • CSPs • Charging pumps
4	Reactor Cavity Flooding	Water injection into containment	• Heat sink for RCS • Containment integrity • Containment basemat integrity • Reduction of fission product release	• CSP/ECSBS • IVR-ERVCS • Cavity flooding system
5	Fission Product Release Control	Containment depressurization	• Reduction of release from containment • Reduction of release to aux. building	• CSP/ECSBS • Containment fan cooler
		Steam dump	• Reduction of release from SG	• MS ADV and TBV
		Auxiliary building venting	• Reduction of release from aux. building	• Venting system in aux. building
6	Reactor Containment Control	Actuation of containment heat sink	• Containment integrity • Reduction of release from containment • Environment of instrument and equipment	• Containment spray • Containment fan cooler
7	Reactor Containment H_2 Control	H_2 control	• Prevention of H_2 explosion	• Igniter • Recombiner
		Intentional H_2 burning		• Spark equipment
		Inertness of containment		• Stop heat sink operation • RCS valve opening

In conclusion, Mr Seo (KEPCO-E&C) indicated that the APR1400 is designed with advanced Severe Accident Mitigation design features. Severe Accident Management Guidelines (SAMGs) are developed for plant specific application which is severe accident prevention/mitigation as well as available instrumentation for plant monitoring. The design incorporates sufficient equipments with recovery guidelines to prevent large release to environment during severe accident. He assured the forum that the APR1400 SAMG is much

upgraded compared with that of the OPR1000, though the OPR SAMG was not discussed in the presentation. Therefore, the APR1400 design meets CR1.4.3 “In-Plant Severe Accident Management”.

4.4.2.6. CR4.2.1 RD&D and CR4.2.3 Scaling

Mr Kim (KHNP) presented assessment for UR4.2 on RD&D for understanding. The APR1400 has advanced safety design features in comparison to the currently operating PWRs, for example, Safety Injection System, IRWST and Sparger, POSRV and Rapid depressurization system, IVR-ERVCS (In Vessel Retention through External Reactor Vessel Cooling System) for severe accidents, and modern MMIS. All the advanced design features are evaluated according to the INPRO user requirement and criteria on RD&D. For the purpose of the meeting, the RD&D of two of the features were evaluated, namely the Direct Vessel Injection (DVI) in Safety Injection System (SIS) and the Passive Fluidic Device in Safety Injection Tank (SIT).

Mr Kim (KHNP) first discussed the design characteristics of the APR1400 Safety Injection System (SIS) in comparison with that of the OPR1000. The APR1400 employs a Direct Vessel Injection (DVI) versus the Cold-leg injection used in OPR1000. The APR1400 has mechanically and electrically independent 4 trains with no cross-tie between the trains, offering easy maintenance. Each train consists of one safety injection pump (SIP) and one safety injection tank (SIT). The OPR1000, on the other hand, has 2 trains with each train connected by cross-tie line. In the APR1400 design, a passive fluidic device is installed in the SIT to regulate discharge flow rate in passive manner. The adoption of fluidic device eliminates the use of large capacity low pressure safety injection pumps (LPSIP). The IRWST is located at the lowest part of containment, thus there is no recirculation mode change necessary.

The APR1400 uses the direct vessel injection (DVI), i.e. injection of emergency water directly to the reactor vessel - horizontally position 83″ above the cold leg, whereas the OPR1000 employs cold leg injection (CLI), i.e. injection of water to the cold leg 60° at the RCP discharged leg. The conceptual advantage of DVI design is that emergency water can be used more effectively. In CLI design, when large cold-leg break accident occurs, one safety injection train connected to broken cold leg cannot be used. So, no such assumption is needed in the DVI design because all emergency water is available during LOCA.

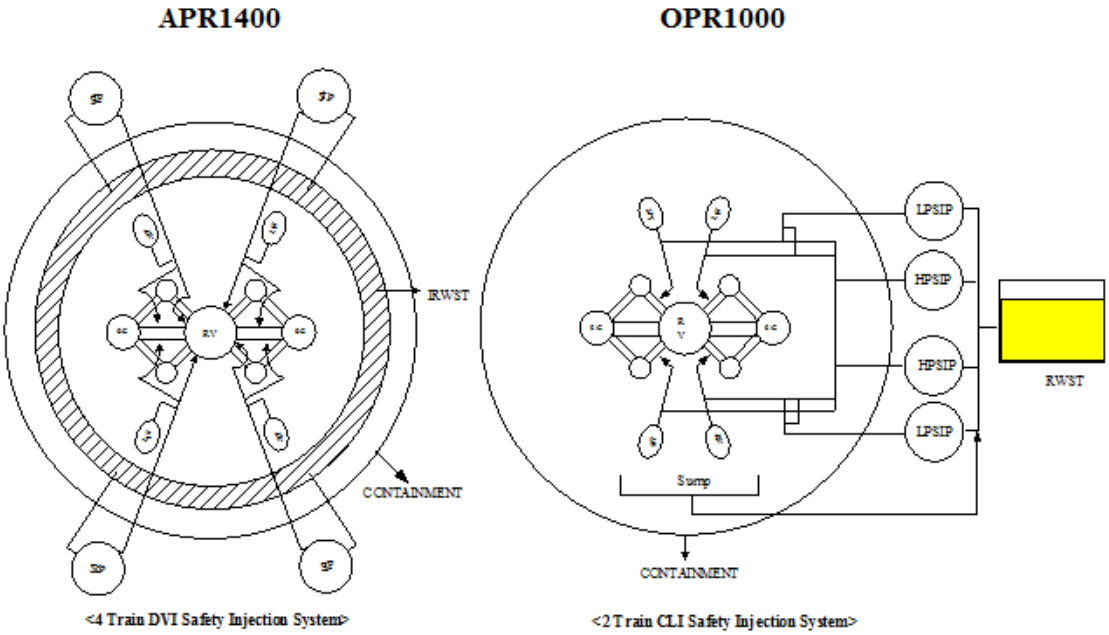


FIGURE 4.4.37. Safety Injection System: Direct Vessel Injection (DVI) for APR1400 versus Cold Leg Injection (CLI) for OPR1000

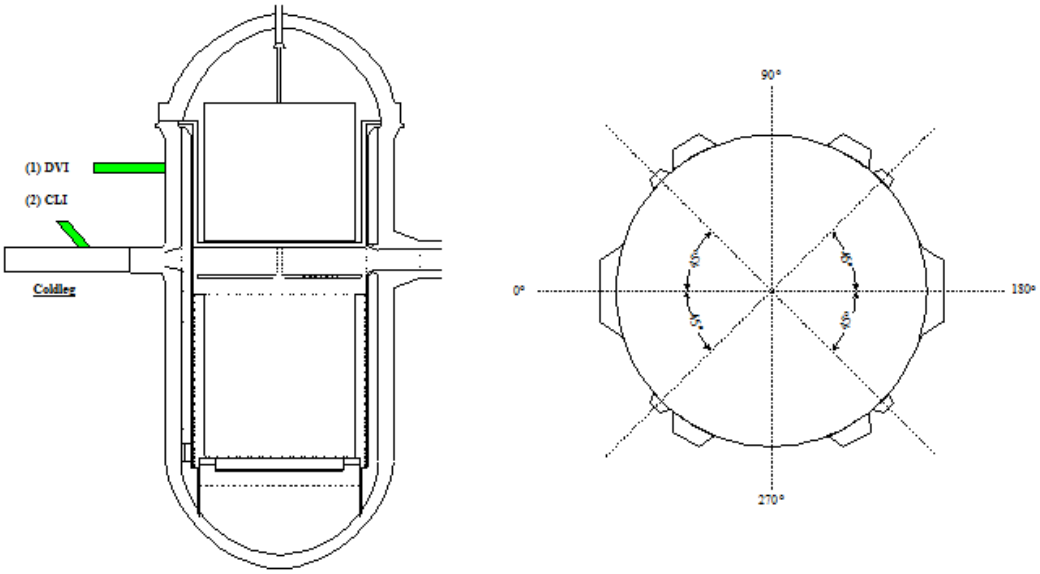


FIGURE 4.4.38. Injection location for Safety Injection System: Direct Vessel Injection (DVI) for APR1400 versus Cold Leg Injection (CLI) for OPR1000

The SIT (accumulator) has a role to refill reactor vessel lower plenum rapidly during early phase of LBLOCA. At initial condition, it is pressurized to 600 psia by nitrogen gas (1800 ft^3 (51 m^3) for APR1400. The problem is that there is too much water to fill the lower plenum. To overcome this problem, a passive fluidic device is used in the SIT. The use of the passive fluidic device utilizes water more efficiently to remove the low pressure SIP (LPSIP). Mr Kim (KHNP) clarified further by considering that assuming in the event the cold leg is broken, all water inside the RCS is discharged to the containment because the pressure in the RCS is 2,250 psia compared to the atmospheric pressure of the containment. When the RCS is empty, the fuel is adiabatically heated-up. To keep the fuel cladding temperature within the limit, water in the SIT is injected to the RCS by pressure difference.

The SIT is pressurized to 600 psia by nitrogen gas and when LBLOCA occurs, the RCS is depressurized to atmospheric pressure. The only function of SIT is to fill the reactor vessel lower plenum during LBLOCA.

The graph below shows comparison between the conventional SIT with LPSIP flow rate and the ideal SIT with Fluidic Device flow rate. The black line indicates flow rate from the conventional SIT. The problem of the current design (SIT with LPSIP) is too much water is injected to the RCS. Most of the water is spilled out to the containment. The desirable flow rate curve (SIT with fluidic device) is represented by the red line. It is shown that at the initial stage, large amount of water should be injected to reactor vessel to fill the lower plenum of the reactor vessel. Afterward, small amount of water should be injected for long time to remove low pressure SIP.

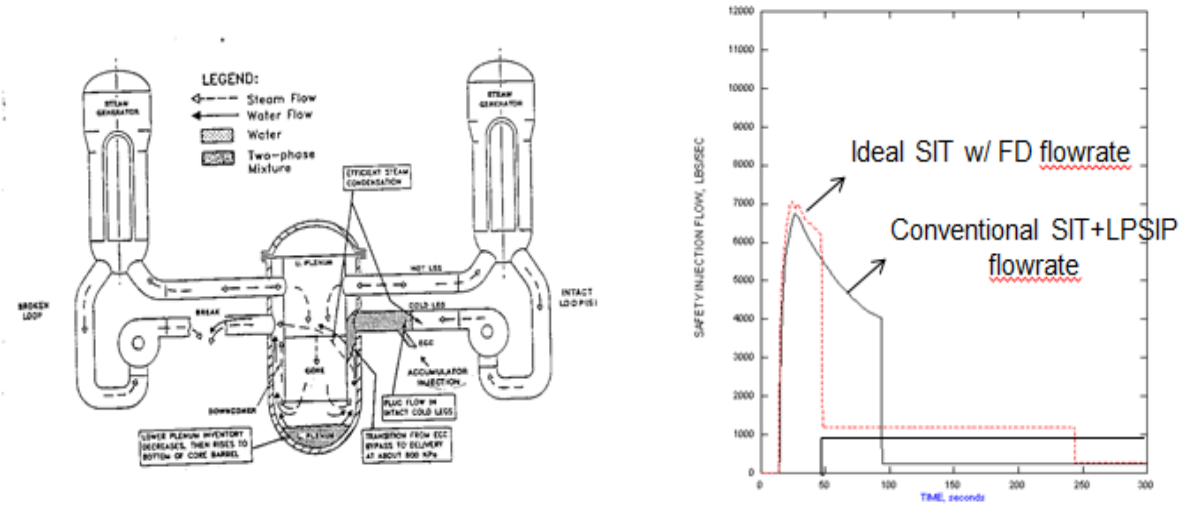


FIGURE 4.4.39. Safety Injection Tank (SIT)

The fluidic device has no electrical source and moving part, and it is installed at the bottom of the SIT. The principle of the fluidic device is vortex resistance. There is stand pipe up to certain elevation of water level. If water level is higher than the stand pipe elevation, water flows through the stand pipe. Then flow direction is perpendicular to the exit. So, small flow resistance is generated and large amount of water flows to the exit. If water level is below than the stand pipe elevation, water flows through the control port only. Then the flow direction is tangential to the exit. So, high vortex resistance is generated and small amount of water flows to the exit.

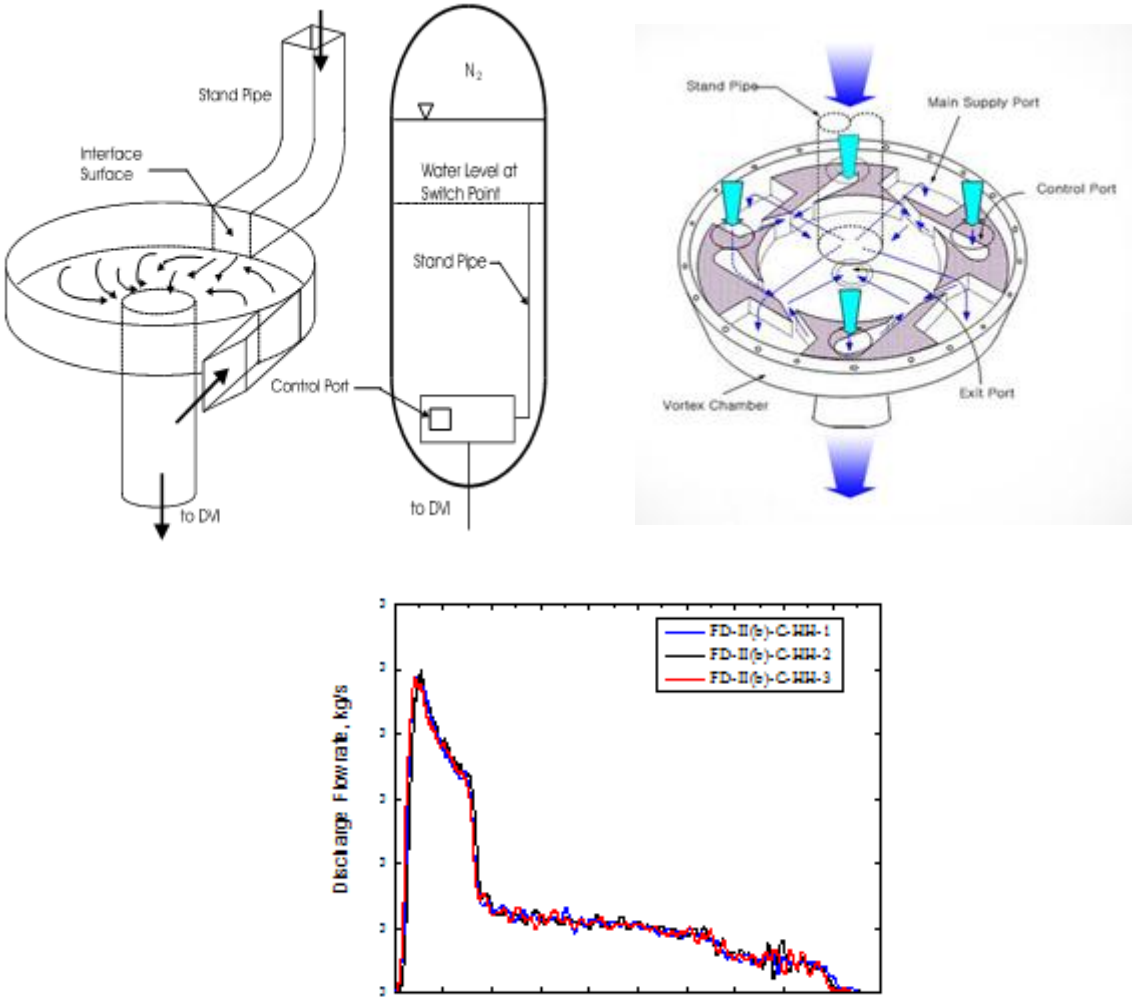
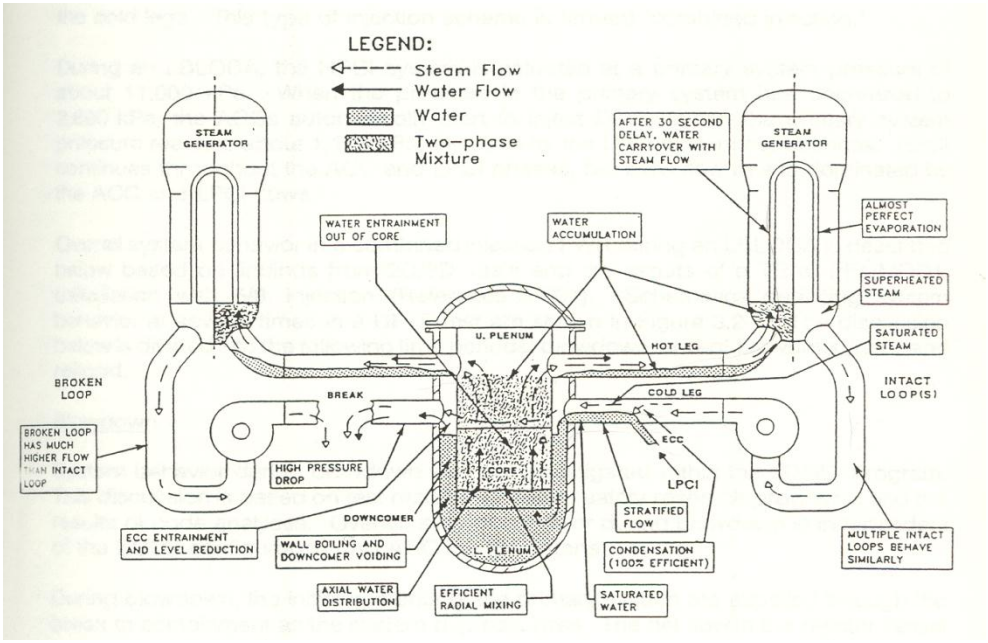


FIGURE 4.4.40. Schematic of passive fluidic device in SIT and the actual experimental results showing vortex resistance (flow rate as a function of time)

Evaluation results for DVI design

During reflood phase of LBLOCA, steam generated in the core flows through upper plenum, hot leg, steam generator, cold leg, and downcomer before it is eventually discharged through the broken cold leg. When the steam reaches the downcomer, the steam flows in horizontal direction to the broken cold leg. Emergency water injected through DVI line should flow downward to contribute to the core cooling. Therefore, steam and water are crossing at right angle in the downcomer. This crossing can make various multi-dimensional phenomena such as ECC water bypass, sweep-out, steam condensation etc. So, there is a need to identify these phenomena and to quantify bypass rate for design and licensing.



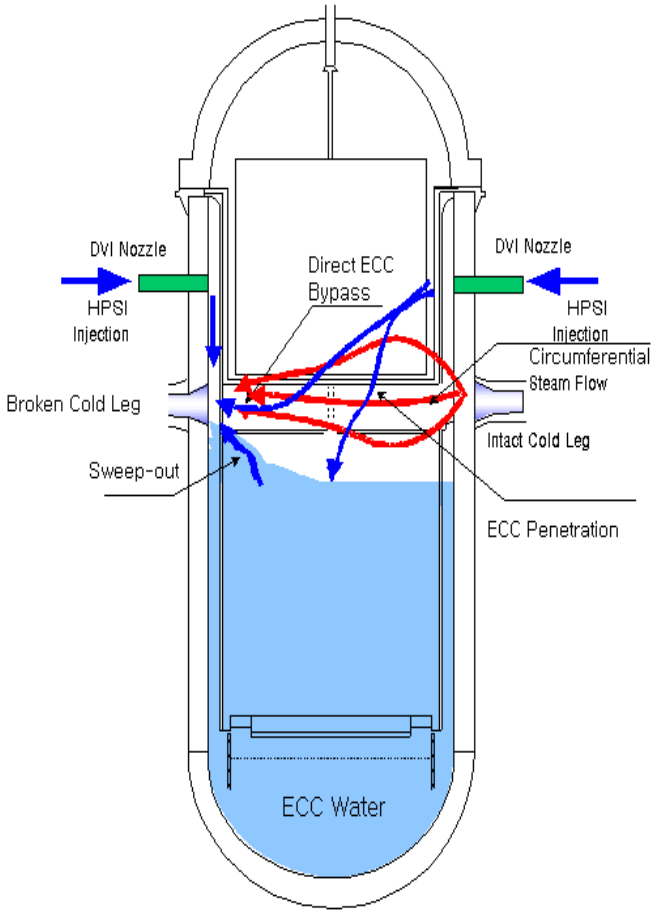


FIGURE 4.4.41. ECC Bypass Mechanism during the Reflood Phase

To understand the phenomena and to quantify ECC bypass rate, two kinds of tests were performed:.

- Direct ECC Bypass (DEB) test

  Some of injected water is bypassed directly to the broken cold leg by fast horizontal steam flow during reflood phase of LBLOCA, termed direct ECC bypass. Steam water interaction at the upper dowmcomer region is important because the DVI line is located 83″ above the cold leg center line.

- Void Height Test : DEB + Sweep-out

Fast steam flow sweeps out the water accumulated in the lower downcomer. Due to these phenomena, the downcomer water level can be lower than the cold leg elevation. The difference between cold leg bottom elevation and water level is called void height. If void height is large, water head is reduced and reflooding rate could be reduced.

Mr Kim (KHNP) showed the test matrix used for ECC Bypass. Korea constructed three different test facilities: (i) 1/7 scale SET facility, DIVA – air-water test facility; (ii) 1/5 scale SET facility, MIDAS - mainly steam water test facility but can be used as air-water test facility, and (iii) 1/288 integral effect test facility, ATLAS. The first two SET facilities which have different scale ratio were constructed to understand scaling effect. Two different tests were formed with same scaling ratio, air-water test and steam-water test to understand steam condensation effect. The integral effect test was performed to understand overall core and system behavior of APR1400.

TABLE 4.4.5. DVI test matrix for ECC Bypass

Test Case		**Test Scale**	**Major Parameters of Interests**	**Test Objectives**
SET	Air-Water Test	1/7	ECC Bypass Mechanism Direct ECC Bypass (DEB) Void Height (VH) ECC Bypass Fraction	Phenomena understanding
		1/5		Condensation Effect
	Steam-Water Test	1/5	DEB, VH ECC Bypass Fraction Condensation/Subcooling	Parameter quantification Code Validation
IET	Steam-Water Test	1/288	Core behavior ECC Bypass Fraction Condensation/Subcooling	Parameter quantification Code Validation

Mr Kim (KHNP) then discussed the three test facilities:

1. 1/7 Scale SET : 'DIVA' Facility, APR1400, Air-Water

   The DIVA stands out for Downcomer Injection Visualization and Analysis. A linear scaling law is applied, with the exact scaling ratio 1: 7.07. The downcomer gap size is 3.6 cm. The test section is transparent. The main purpose of this test is for visual observation on ECC injection and air-water interaction. The figure below shows the DIVA facility and water injection behavior according to water velocity as well as ECC water bypass behavior according to the air velocity.

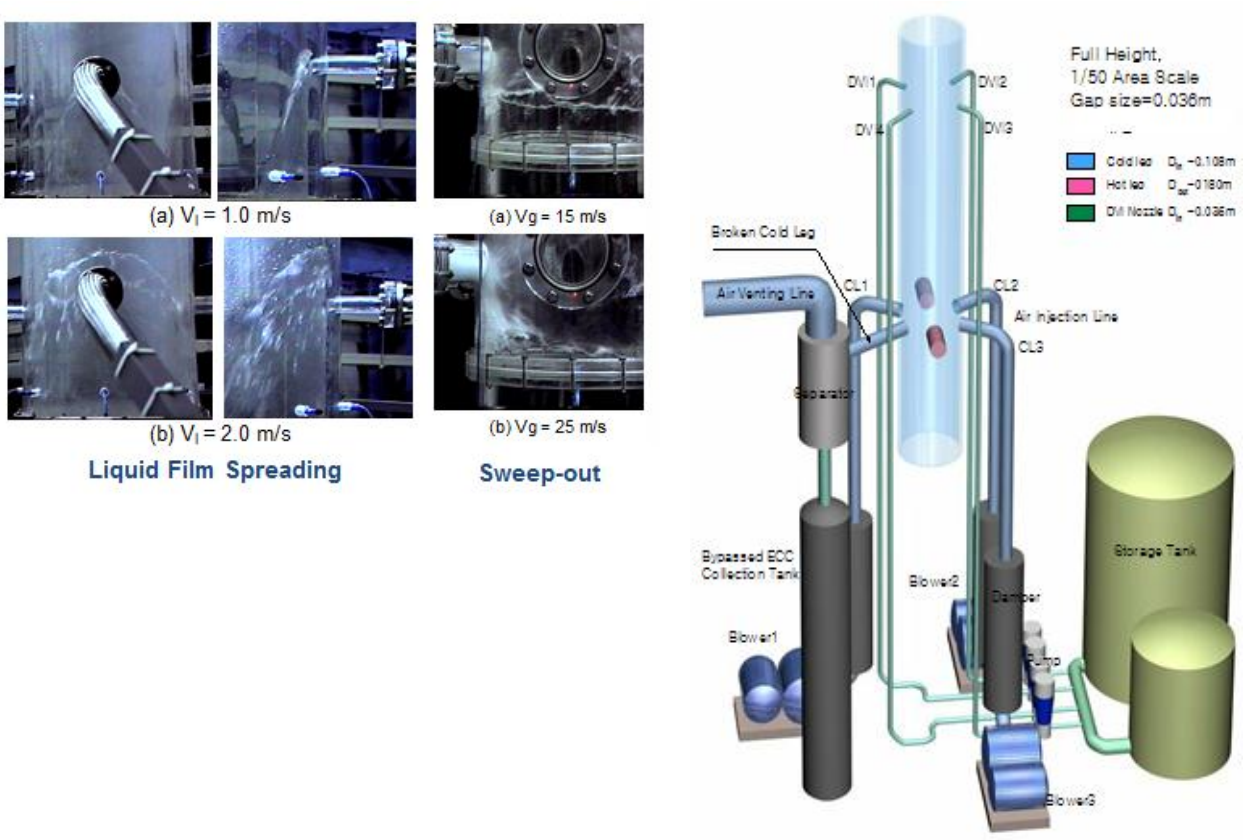
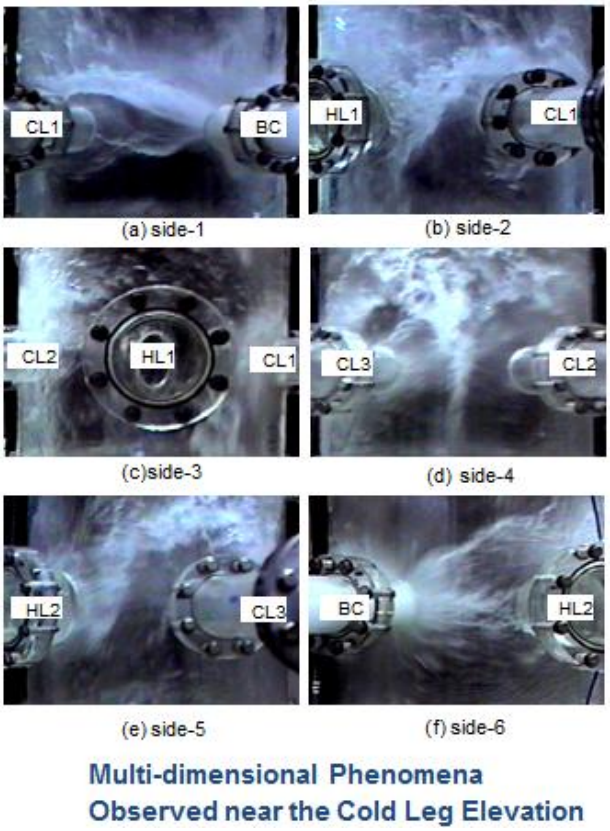


FIGURE 4.4.42. 1/7 Air-Water Test Facility (DIVA)

The cross-sectional view of APR1400 reactor vessel is shown below with two hot-legs and four cold-legs, including the broken cold leg No 4. Side 1-6 correspond to each of the visual observation on the left. Major findings of hydraulic phenomena are as follows. Gas jet impingement is formed. Around the gas jet, liquid slug is formed. Near the broken cold leg, ECC water cannot penetrate into the lower downcomer, i.e. zero penetration. Near the hot leg, ECC water penetrates into the lower downcomer, i.e. local penetration zone (side 2 and 5) (hot leg blockage effect). Flow patterns observed in the downcomer include cross flow (downward liquid film and transverse gas flow), co-current transverse annular wispy flow, and ECC penetration region.

From the visual observation, it can be concluded that if broken cold leg is Cold leg No. 4, the bypass rate of nearest DVI nozzle No. 4 is fairly large, and the bypass rates of the other DVI nozzle are small. When a single failure assumption is applied, only 2 SIPs



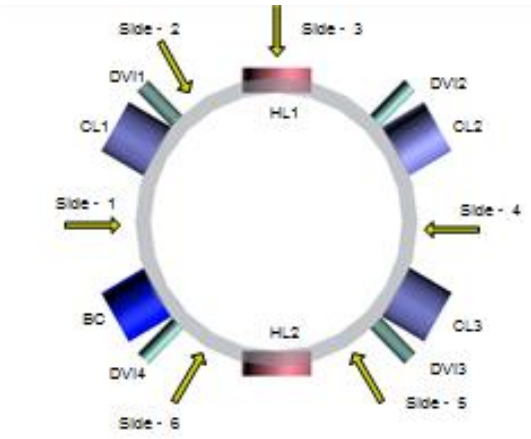


FIGURE 4.4.43. 1/7 Air-Water Test: Visual Observations

2. 1/5 Scale SET: 'MIDAS' Facility, APR1400, Steam-Water or Air-Water

MIDAS stands for Multi-dimensional Investigation in Downcomer Annulus Simulation. The working fluid is steam-water (superheated or saturated steam) or air-water. As shown in the schematic diagram below, the core and downcomer are installed separately to make it easy to measure lots of physical parameters in the downcomer in detail. The downcomer gap of the test section is 51.8 mm, and the internal diameter of the downcomer vessel is 938.8 mm. The exact scaling ratio is 1:4.93 linear scale (DVI internal ∅ = 43.8 mm, Cold Leg internal ∅ = 182 mm).

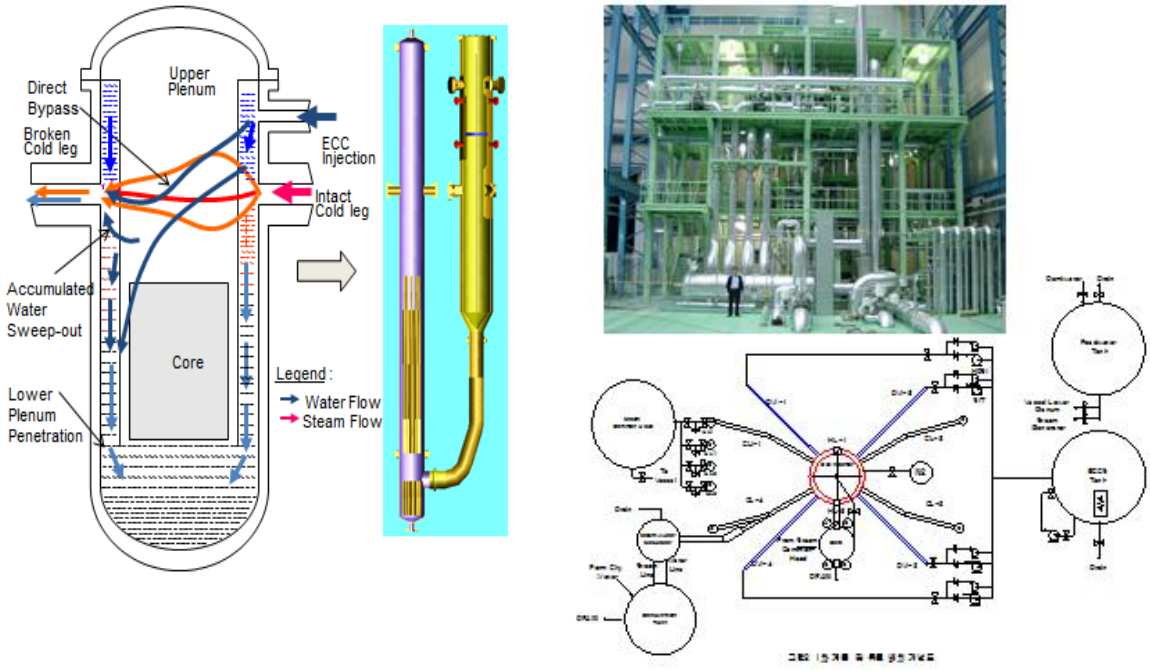
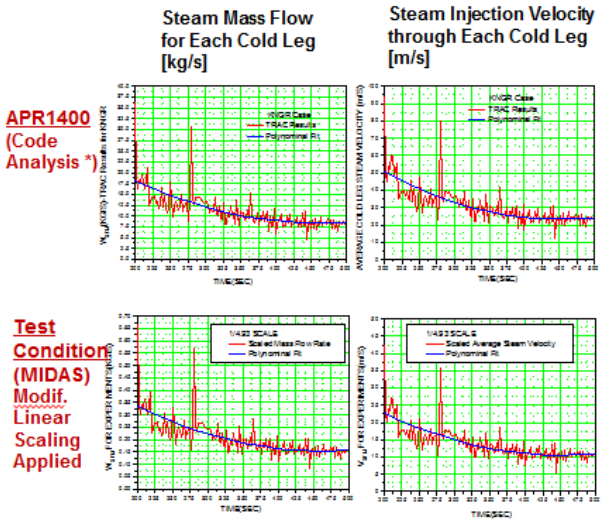


FIGURE 4.4.44. 1/5 Steam-Water Test Facility (MIDAS)

A scaling methodology, so called modified linear scaling method, was developed. At first, linear scaling method is deemed more suitable for ECC bypass phenomena than volume scaling method by the observation of visualization test. Modified linear scaling is basically same as linear scaling except velocity, flow rate and time. To keep ECC bypass phenomena, velocity scale is reduced by square-root of two from the original linear scaling. Major physical parameters such as steam mass flow rate and velocity are adjusted from APR1400 analysis results to test condition according to scaling method in order to setup initial and boundary condition of test facility.



Scaling Parameter	Volume Scaling	Linear Scaling	Mod. Linear Scaling
D/C Height, l_{oR}	1	l_{oR}	←
D/C Dia., d_{oR}	d_{oR}	l_{oR}	←
Aspect Ratio, l_{OR}/d_{oR}	l_{OR}/d_{oR}	1	←
Area, a_{oR}	d_{oR}^2	l_{oR}^2	←
Volume, V_{oR}	d_{oR}^2	l_{oR}^3	←
Flow rate, m_{oR}	V_{oR}	l_{oR}^2	$l_{oR}^{5/2}$
Velocity, U_{OR}	1	1	$l_{oR}^{1/2}$
Time, t_{OR}	1	l_{oR}	$l_{oR}^{1/2}$
Gravity, g_{OR}	1	$1 / l_{oR}$	1

FIGURE 4.4.45. Modified Linear Scaling Methodology. To keep ECC bypass phenomena, velocity scale is reduced by √2 from the original linear scaling.

Mr Kim (KHNP) then presented the direct ECC bypass test results for APR1400. As seen below, the left graph shows ECC water bypass ratio according to steam velocity. The *y*-axis is bypass fraction and *x*-axis is non-dimensional superficial steam velocity. It was noted from the graph that when vapor velocity is greater than certain value, ECC bypass rate is sharply increased. According to LBLOCA condition of APR1400 is near the threshold area. The black dots show steam-water test results and the red ones show air-water test results. The differences are the steam condensation effects, that is, under the same vapor velocity, bypass rate for air-water test is larger than that of steam-water test because some of the steam is condensed when it interacts with sub-cooled water. Therefore, final vapor velocity is reduced. Meanwhile, the graph on the right shows ECC bypass rate according to ECC water injection location. As shown, when ECC water is injected near the broken cold leg, ECC bypass rate is much higher than other injection location.

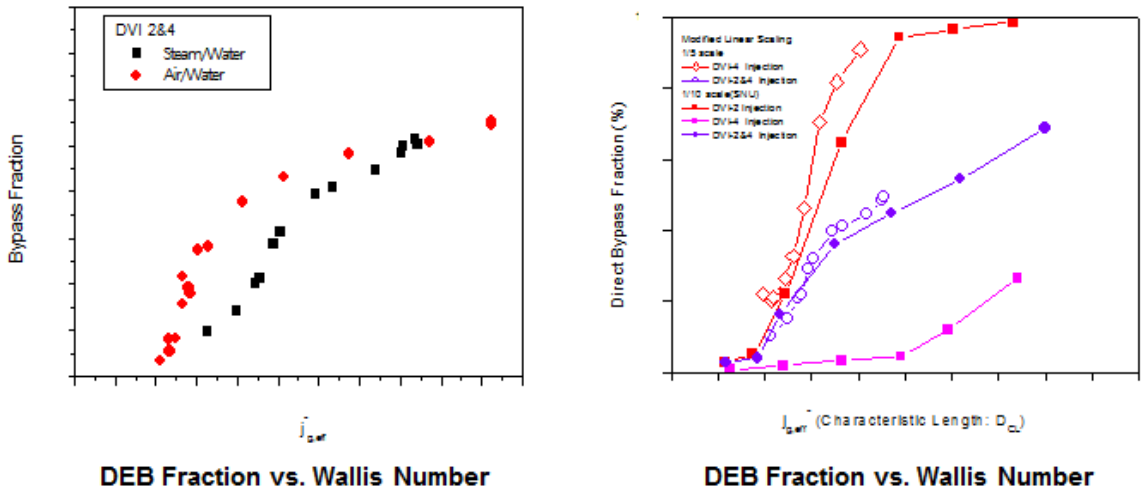


FIGURE 4.4.46. Direct ECC Bypass Test Results for APR1400

The void height test results for APR1400 are presented below. Each colored dot shows data for certain steam mass flow rate. For example, black dot means 0.15 kg/s per each cold leg, red means 0.19, green means 0.26, and blue means 0.35. Simply speaking, black dots represent the slowest steam velocity and blue dots represent the fastest steam velocity. The APR1400 plant condition is equivalent to black dots. As seen in the graph, void height is proportional to steam velocity. However, void height is independent of ECC water injection location. This graph shows that void height during LBLOCA condition is negligible under APR1400 plant condition.

Injected ECC water is highly subcooled water. When subcooled water interacts with superheated steam, subcooling degree is reduced. This graph shows this phenomenon. If steam mass flow is very large, subcooling of ECC water is almost lost. However, in APR1400 plant condition, the graph shows that sufficient subcooling degree is maintained.

The graph on mass balance error for the test indicates that test has been performed very accurately.

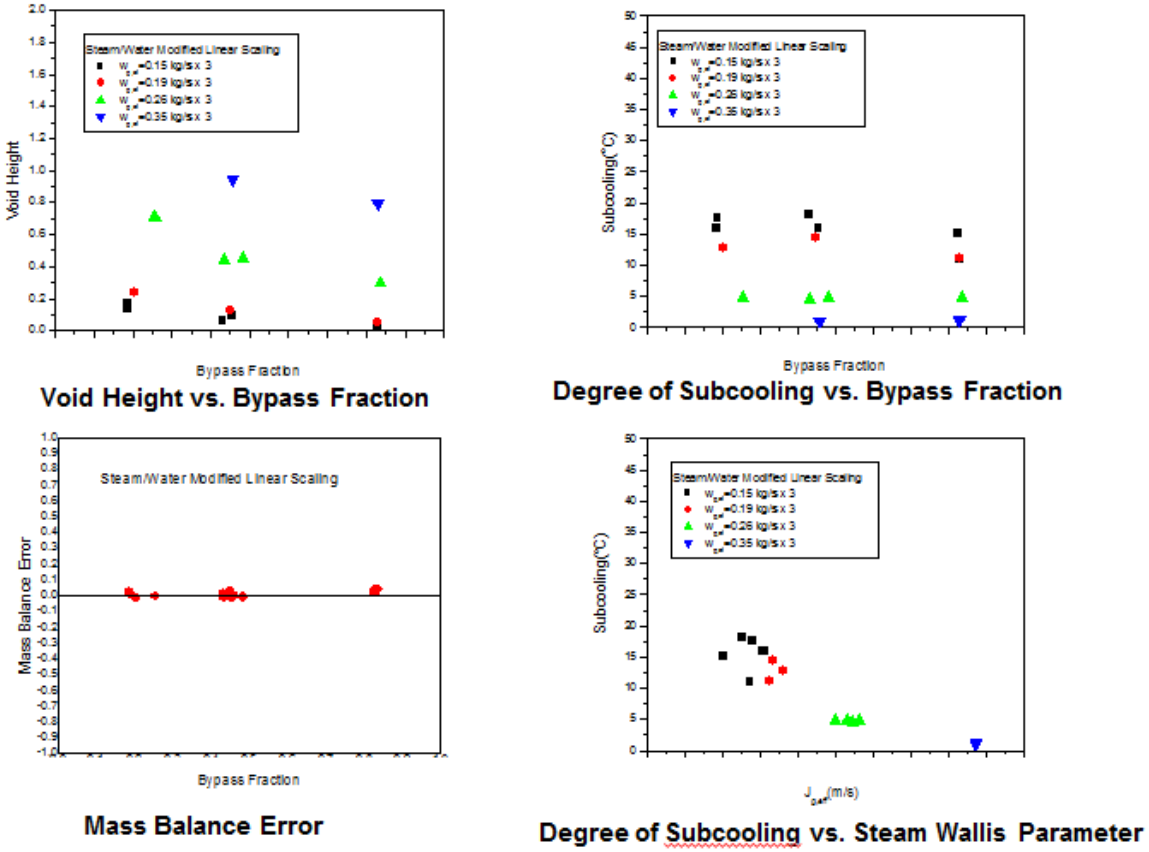


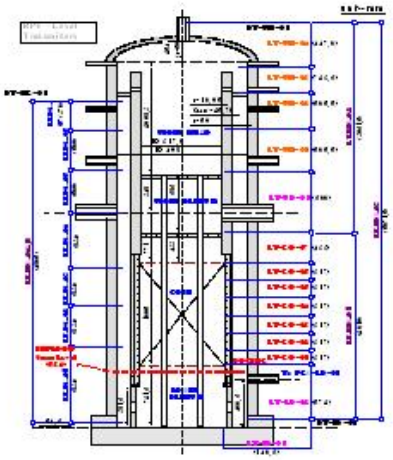
FIGURE 4.4.47. Void Height Test Results for APR1400

3. 1/288 Scale IET: 'ATLAS' Facility, APR1400, Steam-Water

ATLAS stands for Advance Thermal-hydraulic Test Loop. Various separate effect tests (SETs) have been performed to understand and to quantify ECC water bypass phenomena during LBLOCA reflood condition. It was concluded that APR1400 SIS design has sufficient margin on ECC water bypass. Then integral effect test (IET) loop was constructed to understand the overall plant behavior and to validate computer code for APR1400 accident analysis. The figure below shows reactor vessel of ATLAS facility.

The design pressure of ATLAS is 18.7 MPa and the design temperature is 370°C. Height and length are scaled down to half and 1/12 respectively. So, volume is scaled down to 1/288. Reactor core is simulated by electrical heater. Maximum heater power is 2 MW. ATLAS facility can simulate LBLOCA, SBLOCA, SLB (steam line break), FLB (feed line break), ATWS, TLOFW (total loss of feed water), and SBO. And all simulations for these accidents for APR1400 have been done. For this meeting, only test results for LBLOCA were presented.

- Design Press. : 18.7 [Mpa]
- Design Temp. : 370 [C°]
- Height, Length scale : 1/2, 1/12
- Volume scale : 1/288
- Max. core power : 2MW
- Tests : LBLOCA, SBLOCA, SLB, FLB, SGTR, ATWS, TLOWF



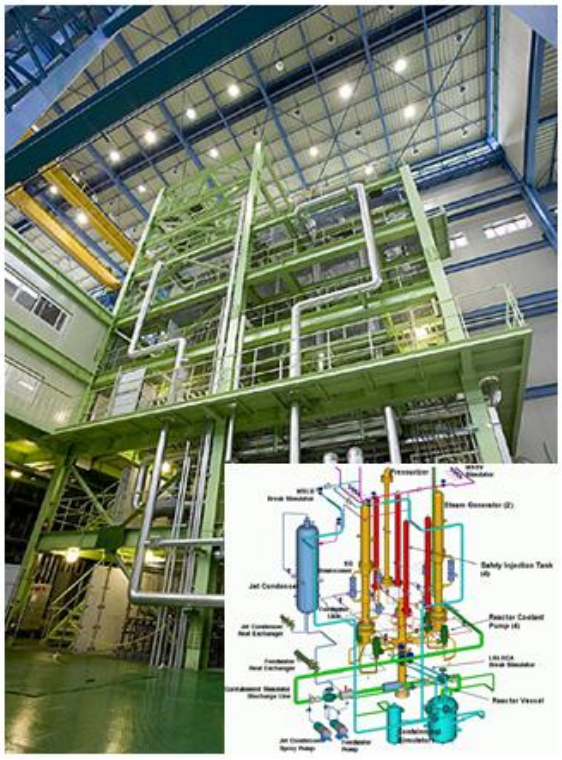


FIGURE 4.4.48. Integral Effect Test - ATLAS

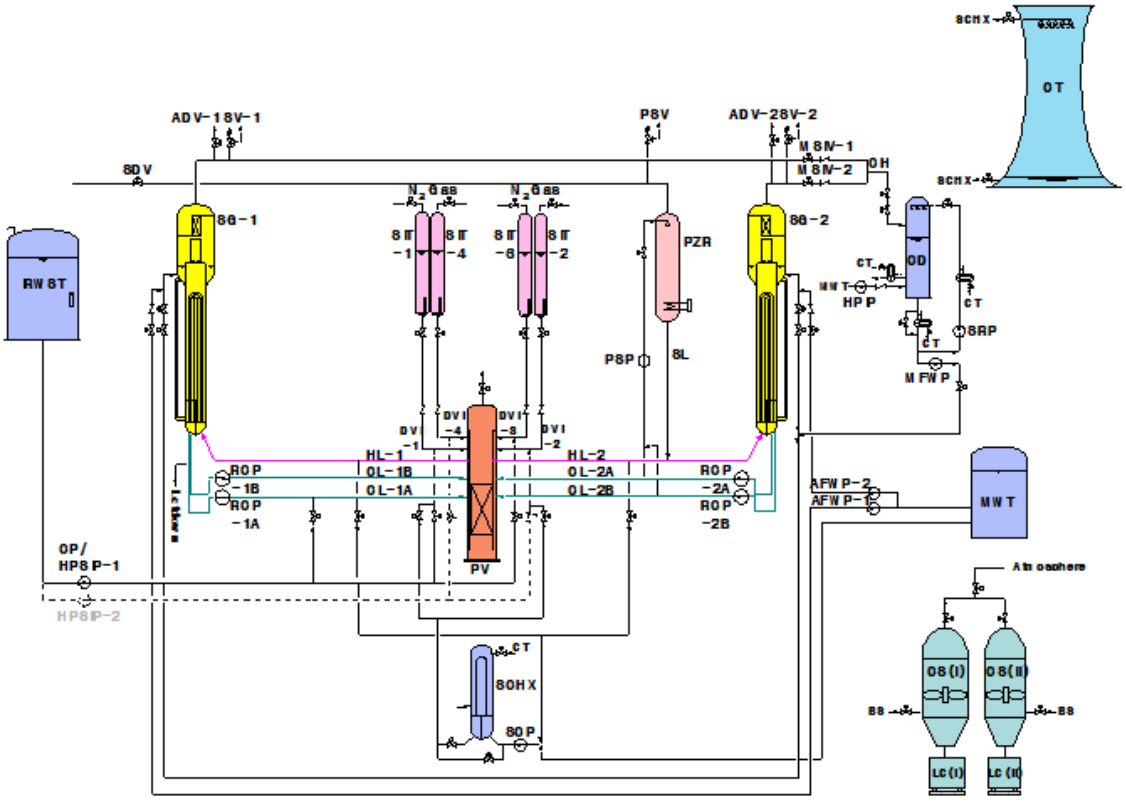


FIGURE 4.4.49. Schematic of the ATLAS Facility

ATLAS major scaling parameters are shown in the following table. The velocity and time is reduced to $1/\sqrt{2}$ because the height is reduced to half. Fuel rod diameter is preserved to keep fuel temperature behavior. All the physical parameters are scaled down according to valid scaling law as shown in this table.

TABLE 4.4.6. ATLAS major scaling parameters

Parameters	Scaling Law	ATLAS Design
Length (Height)	l_{OR}	1/2
Diameter	d_{OR}	1/12
Area	d_{OR}^2	1/144
Volume	$l_{OR}d_{OR}^2$	1/288
Core □T	ΔT_{OR}	1
Velocity	$l_{OR}^{1/2}$	$1/\sqrt{2}$
Time	$l_{OR}^{1/2}$	$1/\sqrt{2}$
Power/Volume	$l_{OR}^{-1/2}$	$\sqrt{2}$
Heat flux	$l_{OR}^{-1/2}$	$\sqrt{2}$
Core power	$d_{OR}^2 l_{OR}^{1/2}$	1/203.6
Rod diameter (core)	RD_{OR}	1
No. of rods (core)	d_{OR}^2	1/144
Tube diameter (SG)	TD_R	$1/\sqrt{2}$
No. of tube (SG)	$d_{OR}^2 TD_R^{-2}$	1/72
Flow rate	$d_{OR}^2 l_{OR}^{1/2}$	1/203.6
Frication factor	F_{OR}	1
Pressure drop	l_{OR}	1/2

The ATLAS LBLOCA test has been performed automatically by control logic. The entire reflood phase of APR1400 is simulated directly. The test has been performed to achieve following two objectives:

- To provide reliable data to help in validating the LBLOCA analysis methodology for APR1400.
  - Licensing issues for APR1400 are to be solved.
  - Quantitative data with uncertainty will be provided with well-defined initial and boundary conditions.
- To understand and identity the major thermal hydraulic characteristics during the reflood phase of LBLOCA for APR1400.

A total of 15 LBLOCA tests were performed using ATLAS facility. The first seven tests (Test No.1-7) are parametric tests during reflood period to understand the effect on core cooling by major thermal hydraulics parameters. The other 8 tests (Test No. 8-15) are LBLOCA reflood tests for the DVI design. Validation results for one test, i.e. Test No. 9, were presented. This is typical APR1400 LBLOCA test from early reflood to late reflood. Conservative initial and boundary conditions are used to demonstrate the design.

TABLE 4.4.7. LBLOCA Reflood Test matrix

Phase Classification	Objectives	Test No.
Phase 1 : Parametric test during reflood	Effect on core cooling by major T/H parameters	Test No.1 ~ No.7
Phase 2 : LBLOCA reflood test	Reflood test for DVI design	Test No.8 ~ No.15

The table below summarizes sequence of events of Test No. 9. The second column contains actual data acquisition time and the third column shows converted time from the start of reflood phase. When heater power is turned on, fuel temperature begins to increase. When fuel surface temperature is higher than 450°C, SIT injection is started. SIT injection flow rate is divided into two stages, high flow stage and low flow stage because of the use of fluidic device to regulate SIT flow rate. From 2 sec after SIT injection, reflood is started. Heater power is started to decay down.

TABLE 4.4.8. Results of events for Test No. 9: Reflood Period

Test ID	Experimental conditions
Test No.9 (IET)	Design Condition From Rx trip to late reflood

Events	Time (DAS)	Time (from reflood)	Description
Power Restart	1855	-57	After achievement of ICs
			20s linear increase
SIT Injection	1910	-2	Max. T > 450°C (target: 456)
	~1930	~18	SIT-High Flow (94% ~ 72%)
	~2033	~121	SIT-Low Flow (72% ~ 47%)
Reflood Start	1912	0	2.0 s after SIT Injection
			1.2*ANS-73 curve
HPSI Injection	1927	15	12.7 s after Reflood Start
Test End	2778.5	866.5	TW-DC-AVG < 115°C; DAS stop

Note: DAS = Data Acquisition System

The graph shows system pressure and reactor vessel wall temperature during tests. Reflood test is started at 1800 sec. Before the test, test facility is maintained at full pressure and full temperature. And pressure and temperature are controlled to make typical LBLOCA reflood condition.

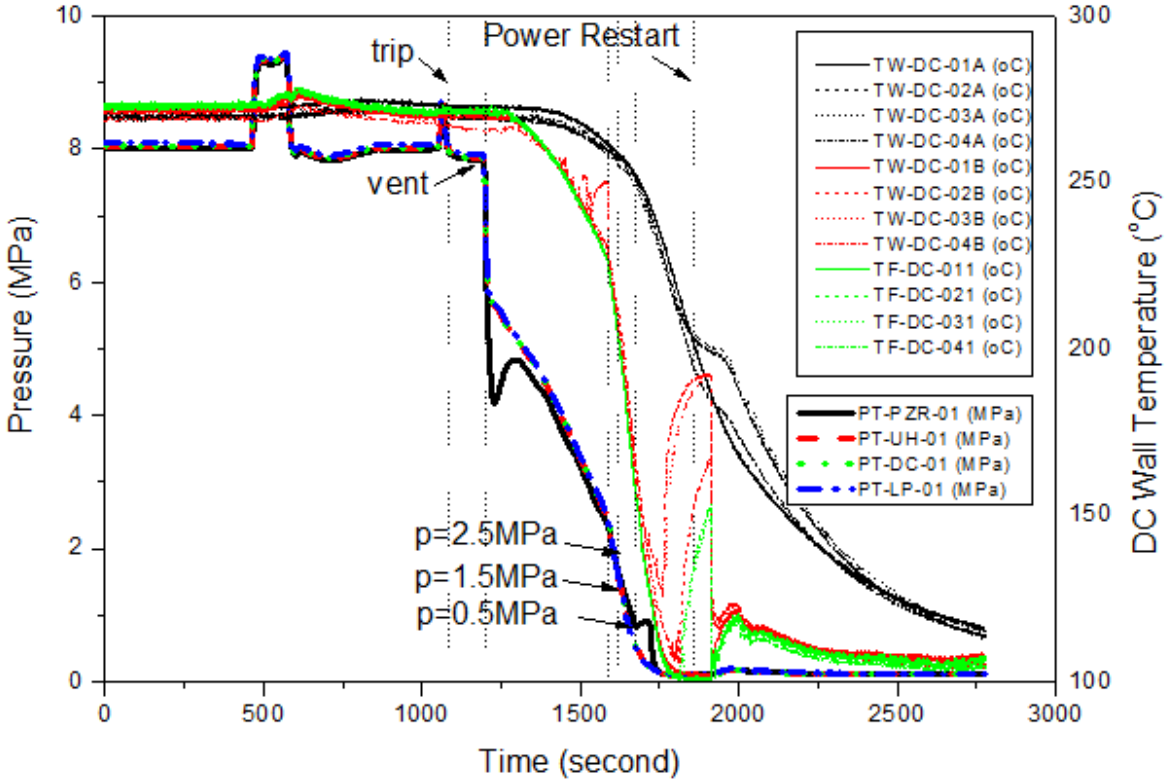


FIGURE 4.4.50. System pressure and DC wall temperature

The following graphs show major parameter's behavior. Reactor power is turned on to achieve full pressure and temperature, and then the power is shut off to reduce RCS pressure, temperature and water level. Before the reflood test, the heater power is turned on to simulate decay power. The reactor power starts to decrease when reflood test is started. Some of the graphs noted were SIT water injection flow rates, core and downcomer water levels, and peak cladding temperature behavior.

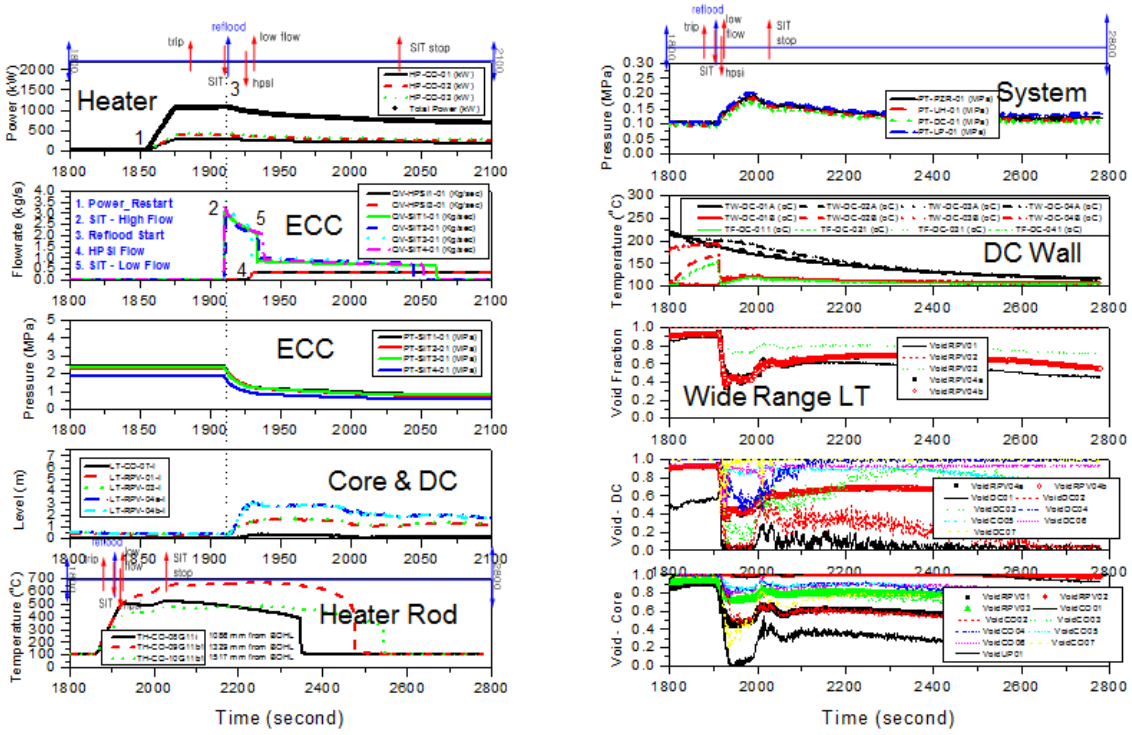


FIGURE 4.4.51. Variation of selected major parameters

Selected test results were discussed:

- Primary System Levels
- The system water level behavior during reflood period was depicted in the figure below. The graph in general illustrates (i) core level decrease after FCV-BS-02 & OV-BS-01 open, (ii) fluctuation at SIT & HPSI injection, and (iii) higher levels in the downcomer than the core. The colors represent the following: the blue line shows downcomer water level, the red and green lines show total collapsed core water level, and the black line is net collapsed core water level, i.e., water level in the active core. The downcomer water level is always higher than the core water level. The difference level between downcomer and core is the driving force for core cooling. When SIT injects water into the core, the core and downcomer water level reach the maximum level. But, just after SIT water injection is stopped, the water level exhibits some transient phenomena, but eventually the core and downcomer water level are increased continuously.

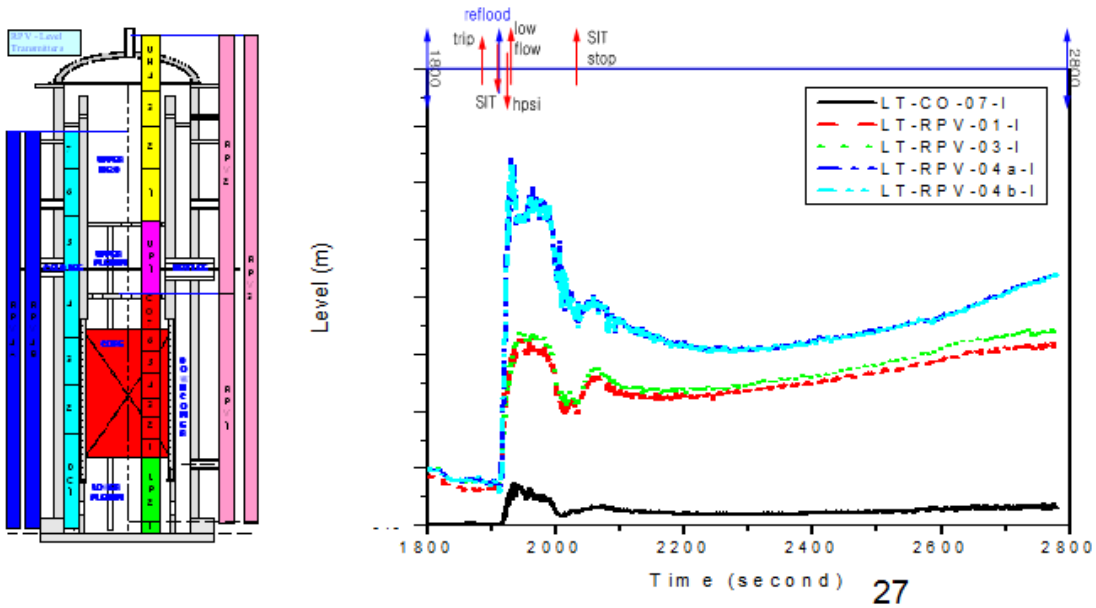


FIGURE 4.4.52. Primary system levels

- Heater rod surface temperature

  The heater rod surface temperature behavior is shown in the attached figure. The colored curves show the behavior at different elevations from the fuel rod bottom to the fuel rod top. The fuel rod elevation is 1.816 m because the height is scaled down to half. The black dots show peak cladding temperature, at an elevation of 1.086 m, slightly higher than that of the center line. The peak cladding temperature is par lower than licensing criteria, which is 2,200 °F. As shown in the graph, the fuel rods experienced gradual bottom-up quenching in the outer zone (Group 3).

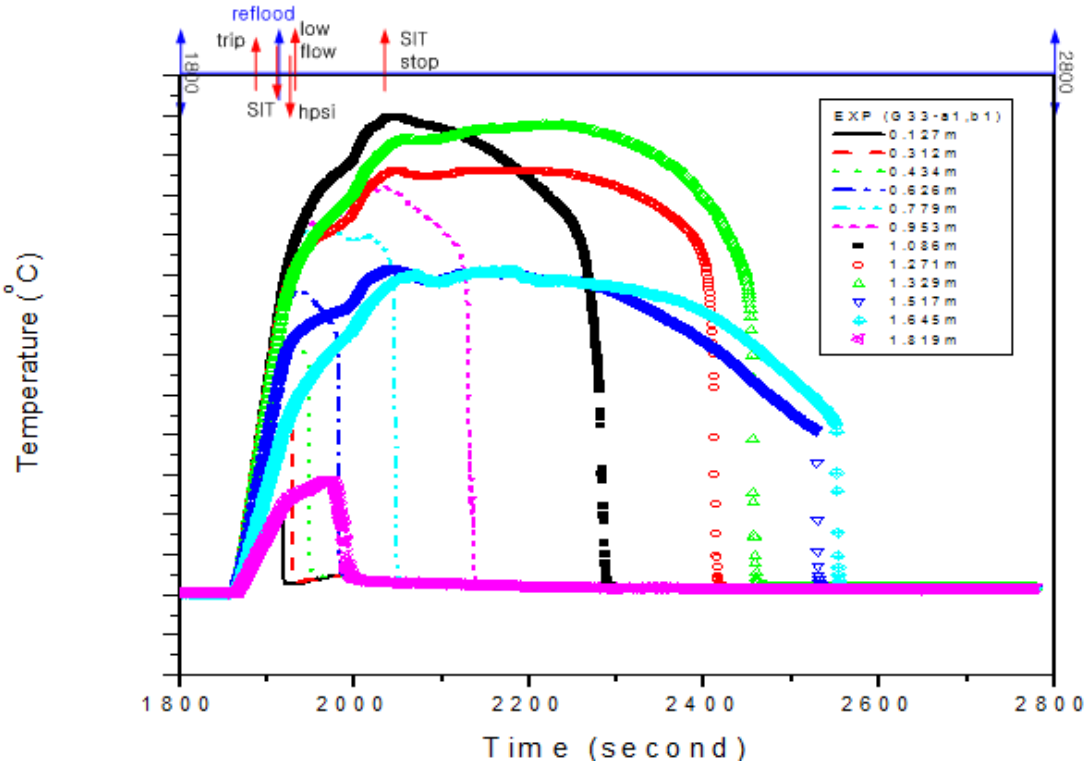


FIGURE 4.4.53. Heater rod surface temperature

In conclusion, Mr Kim (KHNP) concluded on the assessment of the DVI design to develop robust DVI design in APR1400 Safety Injection System. He noted that technical issues are well defined, and RD&D scope and necessity is defined. Various scaling methods are developed and understood. The SETs/IETs have been performed and database has been developed and used for plant design and licensing. Therefore, the APR1400 DVI design satisfies CR4.2.1 on RD&D and CR4.2.3 on Scaling.

Evaluation results for Passive Fluidic Device

Conventional PWR has SIT and large capacity LPSIP to cool the core at early reflood phase which has relative high decay power. The APR1400 has passive fluidic device to regulate SIT flow rate to eliminate LPSIP. The Fluidic Device in SIT is a passive design without electricity and moving part. Because performance should be demonstrated for design and licensing, there is a Full Scale Fluidic Device Test Facility to evaluate it.

Various feasibility tests were performed to decide fluidic device shape and dimension. After that, we discussed how to demonstrate fluidic device performance and we decided that full scale test is required because vortex resistance is not so easy to scale down. This is full scaling test facility for fluidic device, VAPER. This is actual fluidic device. This is installed at the bottom of SIT.

SIT has an internal ∅ of 2.74 m, height of 12 m, and total volume of 68.13 m^2. The SIT is pressurized up to 50 bar using air compressor. The discharged flow rate, water level and pressure are measured after quick opening valve is opened.

- Safety Injection Tank (SIT): Full-Scale (ID: 2.74 m, H: 11.99 m, V: 68.13 m^3)
- Stock Tank : 97 m^3
- Compressed Air Supply : Max. 50 bar
- Discharging Pipe : 12 inch

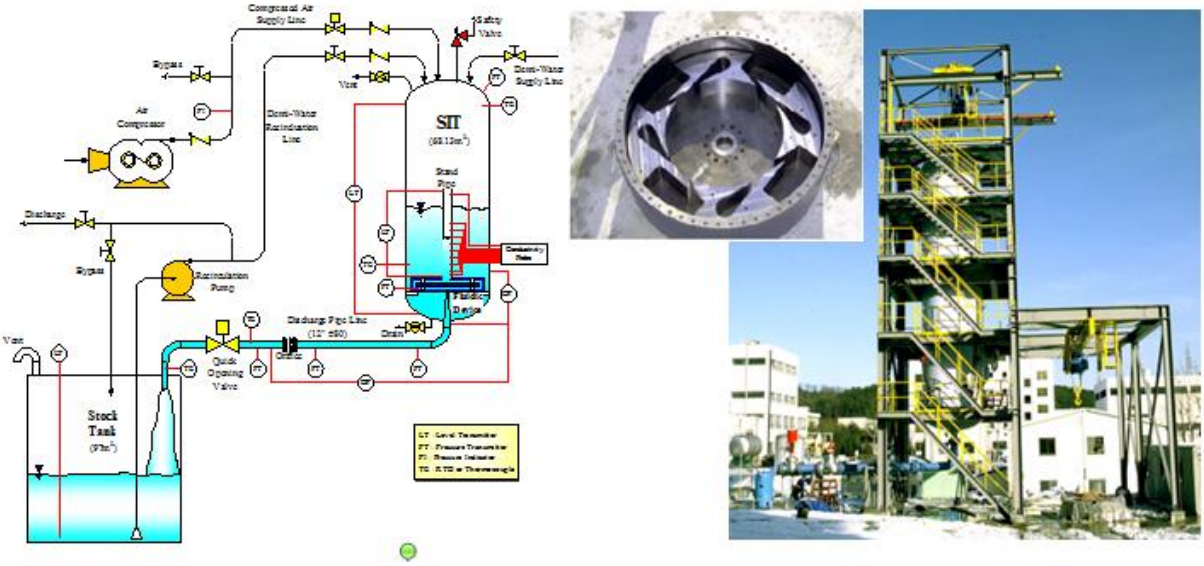


FIGURE 4.4.54. Schematic of the VAPER Facility

Three kinds of tests are conducted: (i) basic tests to confirm performance parameters, (ii) low pressure tests to check initial pressure effect on the performance, and (iii) sensitivity tests for manufacturing uncertainty. The typical test conditions are as follows: system pressure is 4,000 kPag, water temperature is room temperature, and SIT water level is 8.9 m.

The basic principle of fluidic device is vortex resistance. The typical parameters needed to characterize vortex resistance are flow resistance (K-value) at high flow period and low flow period. The K value is defined by the following equation:

$$K = \Delta P \frac{2\rho_w A_{pipe}^2}{W_{SIT}^2} \qquad W_{SIT}(t) \approx \rho_w A_{SIT} \frac{h_{SIT}(t) - h_{SIT}(t+\Delta t)}{\Delta t}$$

Mr Kim (KHNP) provided sample basic test results, as shown in the table below. By various preliminary tests, one fluidic device design was selected (see first row) and the others are repeat test results. On average, the high flow K-value is 16 and the low flow K-value is 160, i.e. 10 times higher than the high flow K-value. Therefore injection flow rate is abruptly reduced to $1/\sqrt{10}$. These graphs show K-values and flow rates. It is shown that Fluidic device has repeatability and good performance.

Test ID	Peak Flow (kg/s)	Mission Time (sec)	FD K-Factor (High/Low)
Test-II(b)-C-H-1	1000	165	18.8/159.7
FD-II(b)-C-H-1	1040	160	16.2/156.3
FD-II(b)-C-H-2	1010	160	15.6/159.8 16.4/158.6
FD-II(b)-C-H-3	1005	165	16.2/159.2 17.0/158.7
FD-II(b)-C-H-4	1005	165	15.7/162.6 16.6/162.6
FD-II(b)-C-H-5	780	210	15.2/169.4 16.1/169.4

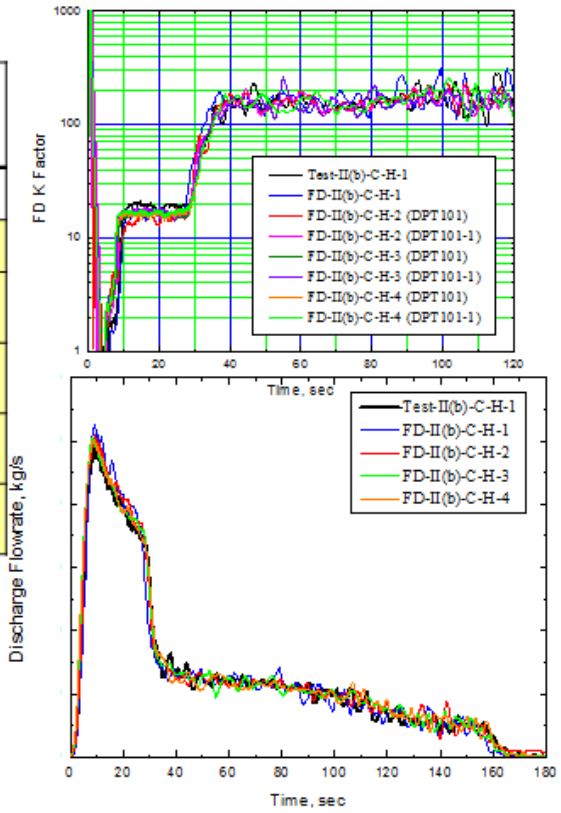


FIGURE 4.4.55. Sample test results

In conclusion, scope and necessity of the RD&D to develop robust Fluidic Device in SIT is defined. Full scale test has been performed and database has been developed and used for plant design and licensing. Therefore, APR1400 Fluidic Device satisfies CR4.2.1 on RD&D and CR4.2.3 on Scaling.

4.4.2.7. CR 4.2.2 Computer Codes

Mr Kim (KHNP) presented code validation for SIS performance (LBLOCA Analysis). He noted various system codes are used for APR1400 design and safety analysis, most of which are the same as those for OPR1000. A new code is used for APR1400 SIS performance analysis i.e. LBLOCA, because APR1400 safety injection system is different from that of OPR1000. The code is RELAP5/Mod3.3-KREM which is a modification of the original RELAP5/Mod3.3 code. This code has been validated using international test facility, ECC Bypass (MIDAS), IET for APR1400 LBLOCA (ATLAS) and Fluidic Device Performance.

Below is the phenomena identification and ranking table (PIRT) to validate system code. Reference accident is double-ended cold leg break. The accident scenario is divided into 4 phases: blowdown, refill, early reflood and late reflood. As many as 15 components and 73 thermal hydraulic phenomena related to APR1400 safety injection system are identified.

TABLE 4.4.9. PIRT for APR1400 LBLOCA

Component	Process/Phenomena	Rank				KREM 반영여부
		1	2	3	4	
Containment	Pressure History	-	-	4	4	Y
	Temperature History	-	-	4	4	Y
SIP	Time dependent delivery	-	-	5	5	Y
SIT	Stored energy release	3	3	4	-	
	Pressure	5	5	5	-	Y
	Liquid temperature	3	3	4	-	Y
	Gas temperature	5	5	5	-	Y
	Level	5	5	5	-	Y
	Gas discharge to piping	-	-	5	-	Y
	FD high flow K-factor	4	4	3	-	Y
	FD low flow K-factor	-	-	4	5	Y
IRWST	Temperature	-	-	4	4	Y
Pressurizer	Flashing	4	3	3	3	Y
	Level	4	3	3	3	Y
RCP	Pump performance	4	3	5	4	Y
Hot Leg	Entrainment	3	3	5	5	Y
	De-entrainment	3	3	5	5	Y
SG	Vapor superheating	3	3	5	5	Y
	U-tube heat transfer	3	3	5	5	Y
Break	Critical flow	5	-	-	-	Y
	Delta-P	3	3	4	4	Y
	Cold leg- Containment	-	5	5	5	Y
Reactor upper head	Flashing	5	2	2	2	
Reactor upper plenum	Flashing	4	2	-	-	
	Level	3	2	4	4	Y
	De-entrainment on structures	3	2	3	3	Y
	Multi-dimensional flow	3	2	4	3	Y
	Entrainment	3	2	3	3	Y

Component	Process/Phenomena	Rank				KREM 반영여부
		1	2	3	4	
Core/Fuel	Stored energy release	5	2	-	-	Y
	Pellet heat transfer	5	5	5	5	Y
	Gap conductance	5	4	5	2	Y
	Nucleate boiling	3	2	5	5	Y
	Critical heat flux	5	2	5	5	Y
	Revert	5	2	5	5	Y
	Film boiling	5	2	5	5	Y
	Transition boiling	3	2	5	5	Y
	Vapor forced convection	4	5	3	4	Y
	Radiation	3	3	4	2	Y
	Void reactivity	5	5	4	4	Y
	Boron reactivity effect	2	2	5	5	Y
	Decay power	5	5	5	5	Y
	Clad oxidation	3	3	4	2	Y
	Clad ballooning	4	4	4	-	Y
	Flow blockage	2	2	4	4	Y
	Fuel relocation	4	4	4	-	
	Clad burst	2	2	4	-	Y
	Guide thimble radiation	3	3	4	2	Y
	Spacer grid heat transfer	5	3	5	5	Y
	Flashing	5	-	-	-	
	Level	-	-	5	5	
	Entrainment	5	-	5	5	Y
	Multi-dimensional flow	4	2	4	3	Y
	Vapor superheat	3	4	5	5	Y
	Interfacial heat transfer	3	2	5	5	Y

Component	Process/Phenomena	Rank				KREM 반영여부
		1	2	3	4	
Reactor Vessel lower plenum	Stored energy release	3	3	3	4	Y
	Level	5	5	3	-	Y
	Multi-dimensional flow	4	2	3	-	
	Entrainment (sweep-out)	5	3	3	-	Y
	Bulk mixing	3	4	4	4	
	Boiling	-	-	-	3	
Reactor Vessel downcomer	Vessel stored energy release	3	3	3	5	Y
	Direct vessel injection jet flow	5	5	5	5	Y
	Flashing	5	-	-	-	
	Level	2	4	5	5	Y
	Entrainment	5	3	5	5	Y
	De-entrainment	4	3	5	5	Y
	Multi-dimensional flow	5	4	5	5	Y
	Counter-current flow	5	3	4	4	Y
	Condensation	5	5	5	5	Y
	Delta-P	4	3	4	4	Y
	Bulk mixing	3	4	5	5	Y
	Non-condensable gas effect	-	-	5	-	Y
	Downcomer boiling	-	-	3	5	Y

Below is code validation matrix derived from the PIRT. 27 SET facilities and 4 IET facilities performed by international societies during last 40 years are used for general PWR test data. Specific test data for APR1400 are used, such as MIDAS, UPTF-21D for ECC bypass phenomena, DOBO for downcomer boiling, ATLAS for overall APR1400 reflood behavior and VAPER for fluidic device performance.

TABLE 4.4.10. Code validation test matrix

	Test Facility	Components						
		Fuel Rod	Core	Pump	Downcomer	Break	Upper plenum/SG	FD-SIT
SET	UPTF Test 21D				•			
	MIDAS				•			
	DOBO				•			
	ATLAS		•		•		•	

	Test Facility	Phenomena	Components					
			Fuel Rod	Core	Pump	Down comer	Break	Upper plenum /SG
SET	UPTF	Lower plenum flashing, entrainment, liquid-vapor mixing with condensation, ECC bypass and DC penetration				●		●
	CE (pump)	Two-phase pump behavior			●			
	LOFT-Wyle	Break, blowdown					●	
	GE	Entrainment and de-entrainment		●				
	Dukler Air-Water	CCFL				●		
	LOFT Accu. Blowdown	Accumulator behavior				●		
	Chistensen subcooled boiling	Subcooled boiling		●				
	MIT Pressurizer test	Condensation in stratified conditions				●		
	ORNL	Interfacial drag, Core heat transfer		●				
	RIT Tube Test	Critical heat flux		●				
	Bennet	Transition boiling		●				
	MINIZWOK	Metal-water reaction	●					
	Power Burst Facility	Gap conductance	●					
	FRIGG-2	Interfacial drag		●				
	THETIS	Interfacial drag		●				
	ECN	Core liquid level		●				
	MB-2	Entrainment de-entrainment in steam generator						●
	CREARE	Downcomer countercurrent			●	●		
	FLECHT-SEASET	Core heat transfer		●				●
	Marviken	Critical flow					●	
	THTF	Blowdown heat transfer		●				
	NEPTUN	Core heat transfer		●				
	Semiscale Pump	Two-phase pump behavior			●			
	W Pump (1/3)	Two-phase pump behavior			●			
	CCTF	Steam binding, Core heat transfer, Quench front propagation		●				●
	SCTF	Core level, entrainment and de-entrainment		●				
	PKL	Steam Binding		●				
IET	LOFT	Core wide void and flow distribution, Steam binding, Core heat transfer, Noncondensible gas effect		●	●	●		
	LOBI	Core wide void and flow distribution, Core heat transfer		●				
	SEMISCALE	Core wide void and flow distribution, Core heat transfer		●				
	POWER BURST	Gap conductance		●				

The following graphs show code accuracy: y-axis represents calculated peak cladding temperature (PCT) and x-axis represents measured PCT. The left graph is PCT prediction accuracy for blowdown period by 25 SET, with a total of 616 test data from 17 FLECH-SEASET, 7 NEPTUN and 1 ATLAS. The right graph is PCT prediction accuracy for reflood period by various SET and IET, with a total of 688 test data from SET + LOFT L2-2, L2-3, LP-02-6, CCTF C2-4, PKL-IIb5, and Semiscale S-06-3. As shown in the graphs, the RELAP/KREM code is slightly over-predicted and sufficiently accurate.

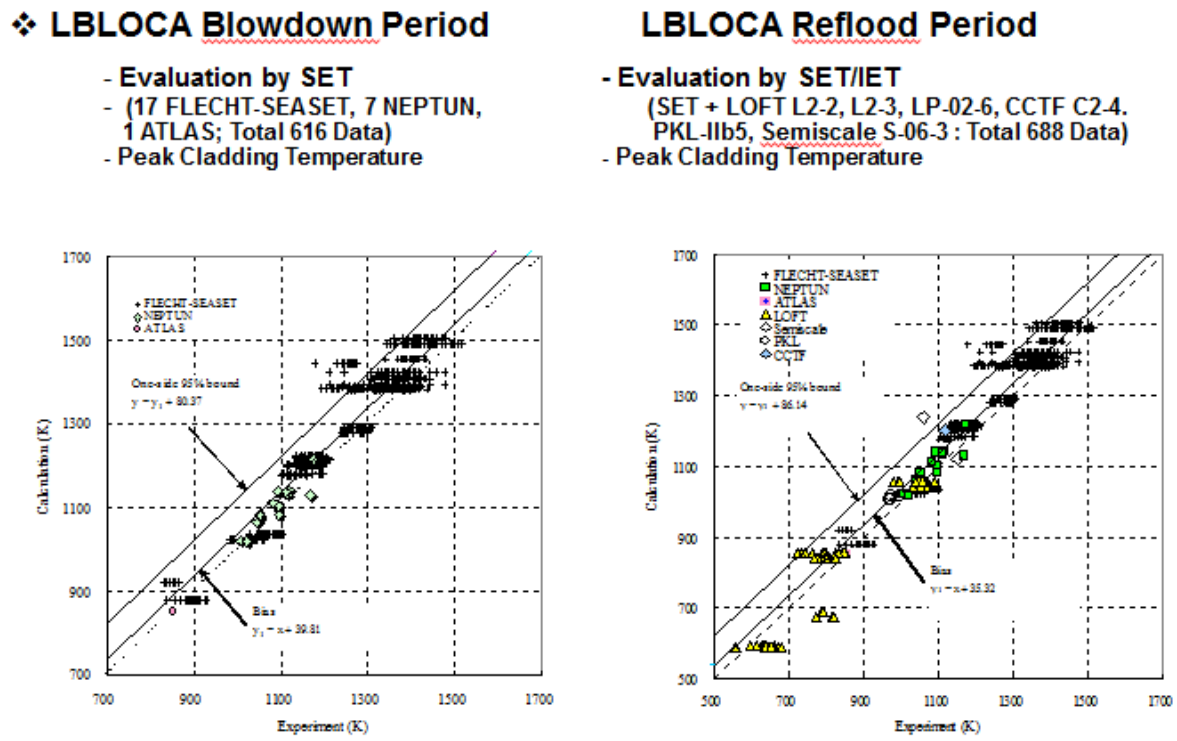


FIGURE 4.4.56. Code accuracy

- Code validation – ECC Bypass (MIDAS)
  Code validation result for ECC water bypass performed by MIDAS test facility along with the nodding diagram is presented below. RELAP5/KREM code predicts ECC bypass rate with consistent conservatism.

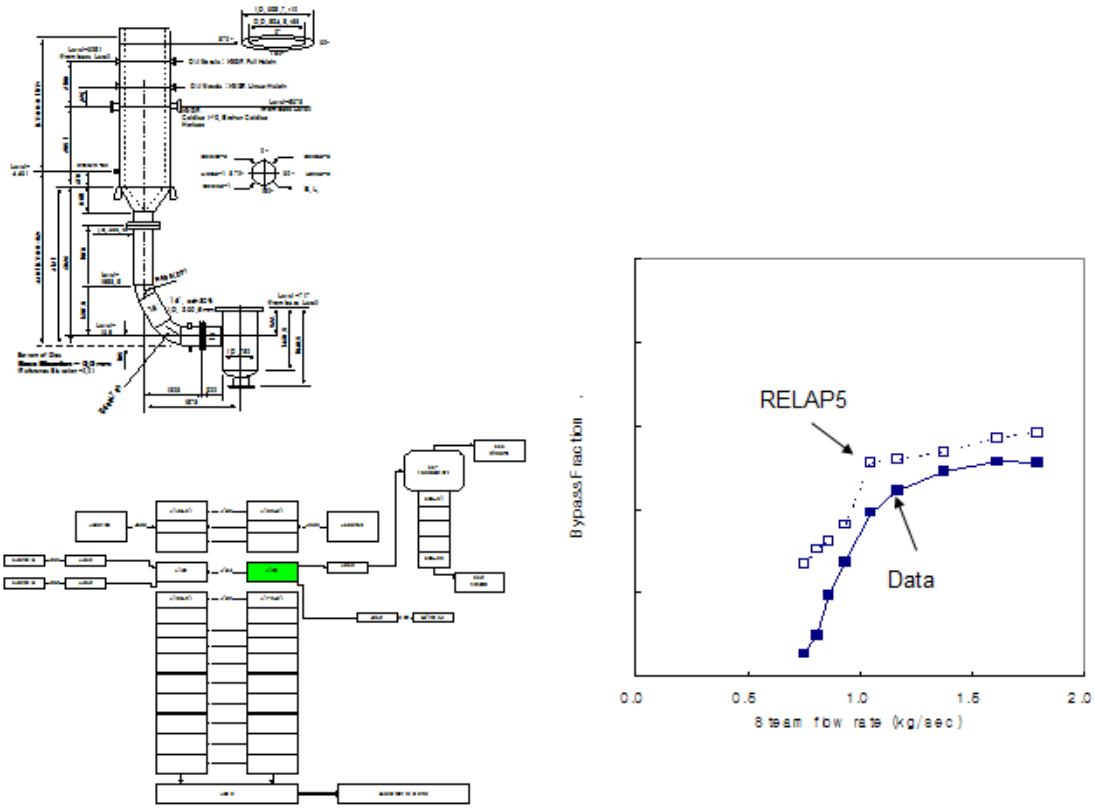


FIGURE 4.4.57. Code validation – ECC Bypass (MIDAS)

- Code validation – ECC Bypass (VAPER)

Code validation result for Fluidic Device performance by VAPER test facility is shown below. Two K-values were measured through the tests. Vortex flow cannot be simulated using the system code. Therefore RELAP/KREM code models fluidic device simply using measured two K-values. Using the simple nodding diagram, SIT injection flow rate can be predicted.

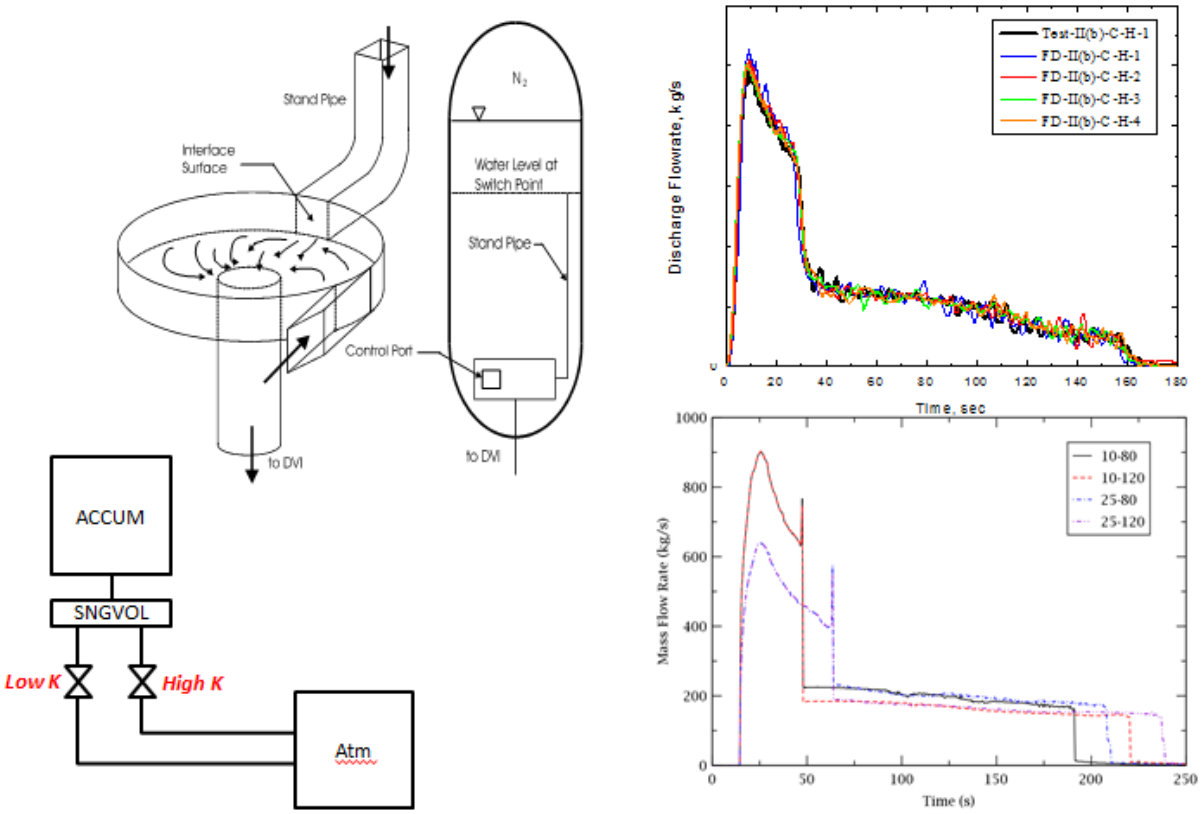


FIGURE 4.4.58. Code validation – ECC Bypass (VAPER)

- Code validation – Overall LBLOCA Behavior (ATLAS)
  Results of peak cladding temperature (PCT) prediction are presented in the figure below. At the elevation which shows maximum PCT at the test, test data and RELAP calculation data are well matched as shown in the center graph. When all the uncertainties are considered, prediction results well covered test data. The lowest and the highest prediction results can be seen.

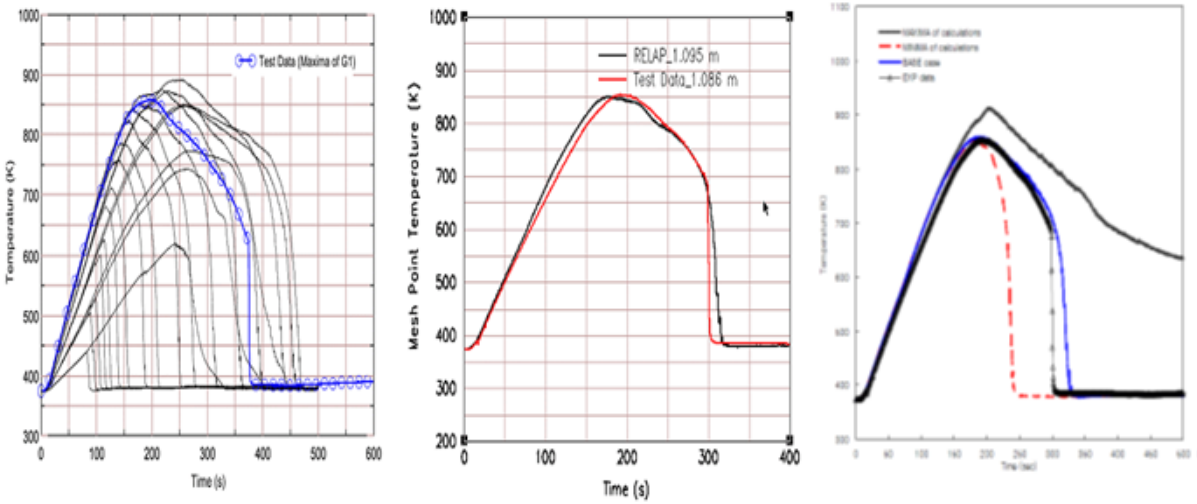


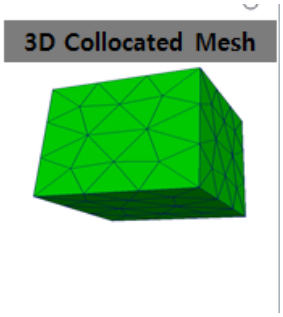
FIGURE 4.4.59. Code validation – Overall LBLOCA Behavior (ATLAS): Validation results of peak cladding temperature (PCT) behaviour

In parallel to modification of RELAP5/Mod3.3 code and validation and verification (V&V) using various test facilities, new thermal hydraulic code named SPACE (Safety and Performance Analysis CodE) are being

developed. The new system code has its specific advanced features. It is developed by C++ language and it can handle 3 fields, i.e. liquid, vapor and droplet. SPACE can use model reactor vessel by 3-D coordinates and handle 3D unstructured mesh. The developmental V&V of SPACE code has been completed in June 2012. SPACE version 2.0 was published at Dec. 2012 and is currently under licensing process (2013-2015). SPACE code will be used to design NPPs after 2015.

TABLE 4.4.11. RELAP5 vs SPACE

	RELAP5	SPACE
Language	Fortran	C++
# of fields	2 (Liq., Vap.)	3(Liq., Vap., Drop.)
Dimensions	1	3
Mesh	Structured	Structured/ Unstructured



Sample of SPACE code validation results are presented below. The test is ATLAS SBLOCA, a 6" cold-leg break in APR1400. As shown in these graphs, prediction data by SPACE are well matched with the test data.

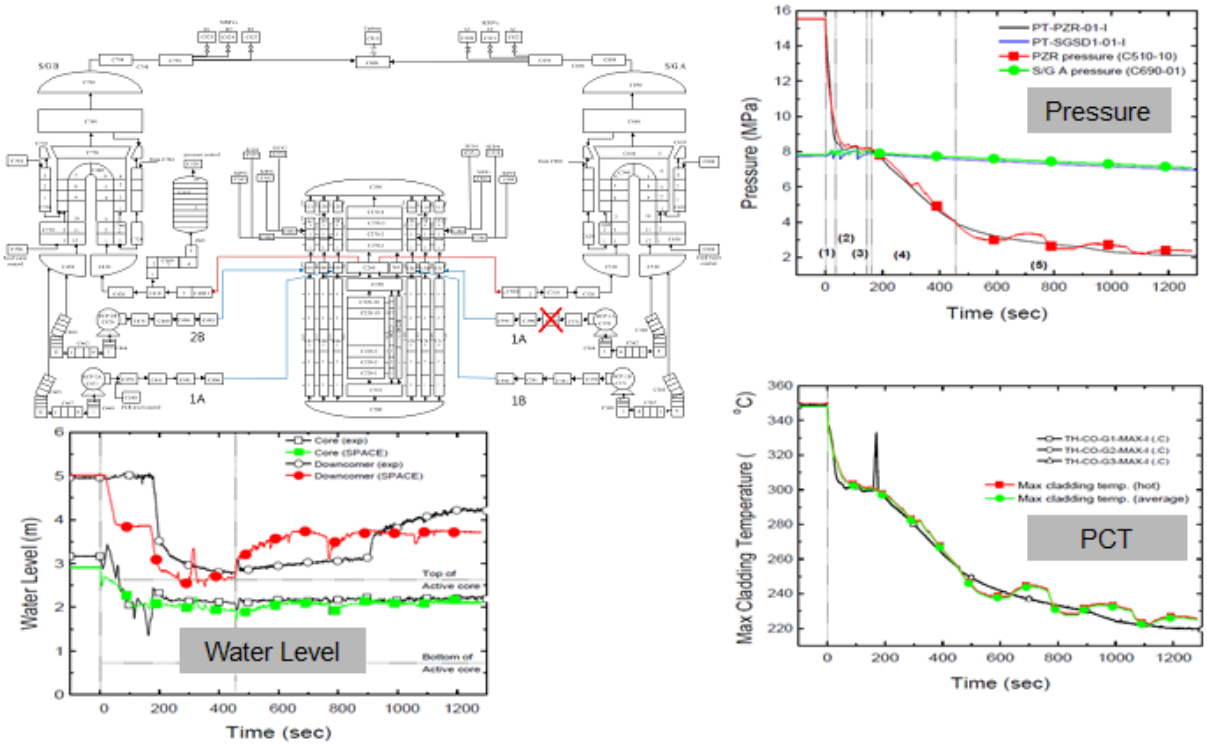


FIGURE 4.4.60. SPACE code validation results: ATLAS SBLOCA 6″ in APR1400

Plant Analysis – APR1400 noding

Regarding the current design RELAP5/KREM code, it has been fully validated using all available SETs and IETs. A noding diagram of APR1400 is developed. 2 hot-legs and 4-cold legs are modeled independently and downcomer is modeled using 6 channels. The core is modeled using two thermal hydraulic channels (hot and average).

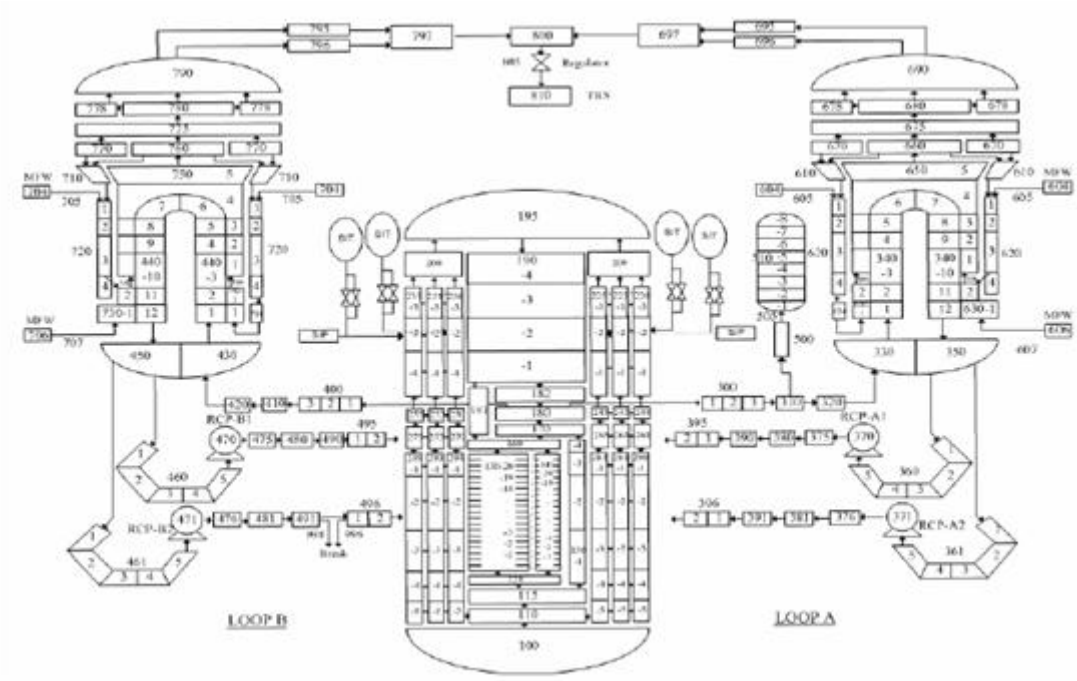


FIGURE 4.4.61. Plant analysis – APR1400 noding

The figure below depicts analysis results with uncertainty quantification for PCT behavior, oxidation rate and hydrogen generation. All these analysis values with uncertainty have sufficient margin in to be compared with the licensing criteria.

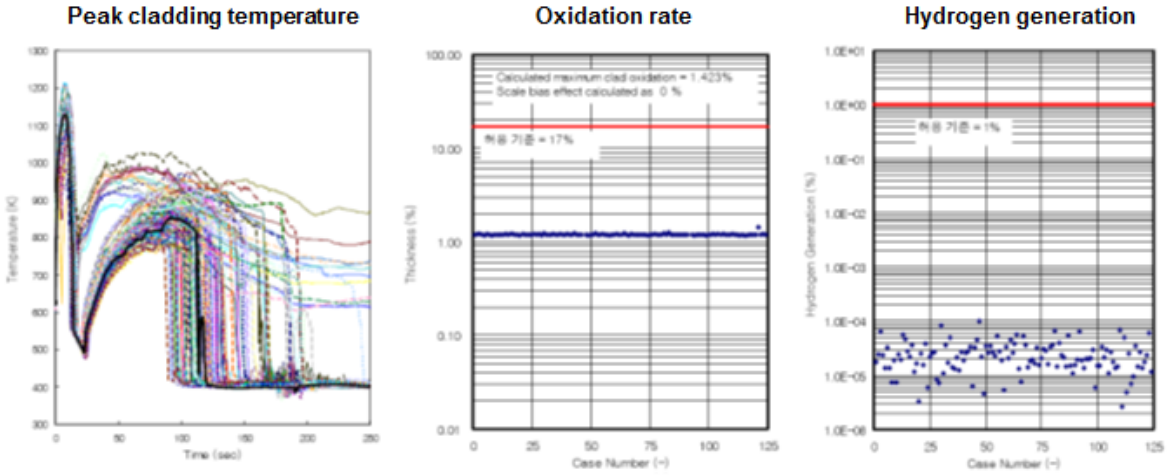


FIGURE 4.4.62. Plant analysis – Uncertainty quantification

In conclusion, RELAP5/KREM code for APR1400 LBLOCA analysis has been developed. This code is validated using various test facilities. Code uncertainty quantification and best estimate safety analysis have been performed. Therefore, APR1400 SIS design satisfies CR 4.2.3 on 'Computer Codes'.

4.4.3. Availability of References

The Design Control Document of APR1400 was submitted to the US NRC at the end of September 2013 and currently it is under review. So once accepted, the licensing review will be started. The document is expected to be available on the US NRC website in time. It was further informed that vendor cannot provide the design data on one to one basis from interested country without commercial or legal agreement. Otherwise, the approach is for an assessor country to go the open data.

4.4.4. Discussion

Questions	Answers
Mr Silva (Thailand) asked why KEPCO	**Mr Seo (KEPCO-E&C)** responds saying both methods are

decided to increase the size of the steam generator rather than to increase the number of loops in order to increase the power.	valid but increasing the size of the steam generators would require less equipment and keep the design simpler (two loops).
Mr Tjahjono (Indonesia) was interested in the experimental hydraulic loop of the APR1400. He asked the kind of experiment that had already been done with this loop.	**Mr Seo (KEPCO-E&C)** specified that the most severe transient is large break LOCA and the experimental loop facility has been used to test a large break LOCA and other DBAs.
Mr Zhou (China) asked how many designs for verification and experiments done for APR1400	**Mr Seo (KEPCO-E&C)** indicated that not all tested done are presented, but all new design features and concepts also new equipment with different philosophy function had to be tested. There are a number integral tests and separate effect tests performed to verify. Based on test results, the Korean regulator approved the APR1400 as a safe design.
Ms Yang (China) asked that the design can load up to $^1/_3$ MOX fuel. Since MOX fuel will change response of the reactor system, she asked about the safety system design for the uranium and mixed core. The use of MOX fuel would result in some changes in the safety margins. She also wanted to know whether the assessment was done for uranium or mixed core.	**Mr Seo (KEPCO-E&C)** indicated that APR1400 can accommodate up to $^1/_3$ MOX fuel without any design changes, and MOX loading has been evaluated. The assessment was not based on MOX fuel. He agreed that with and without MOX fuel, the plant parameters are different. However, currently no consideration is given for loading MOX fuel. **Mr Phillips (IAEA)** pointed out that when doing assessment, many are better compared to reference plant. So in doing INPRO assessment for MOX fuel, then it should be compared with a reference plant using MOX too, thus comparing similar things showing the evolution of reactor, within the same technical lineage, e.g. PWR in the same series, so not the specific behaviour of the fuel. Despite this, one can use the INPRO Methodology for any purpose.
Mr Reyes (Chile) asked about the construction period of the plant in Korea and in another foreign country.	**Mr Seo (KEPCO-E&C)** said that from the first concrete until fuel loading, the construction period for a FOAK is 51 months but the follow-up unit will take less maybe down to 47 months. But the goal is high which is to reach 36 months.
Mr Koshy (IAEA) asked on plant protection system and ECCS, whether the actuation system for reactor trip and ECCS are fully independent or shared – if so, which parts are shared.	**Mr Seo (KEPCO-E&C)** referred to presentation on CR1.3.3.
Mr Tjahjono (Indonesia) asked about how the APR1400 copes with the station blackout (SBO) without passive safety systems.	**Mr Seo (KEPCO E&C)** replied saying that the APR1400 is not a passive design plant but a hybrid design having active and passive features. Active components cannot operate in a station blackout but there are capabilities of operating them with use of batteries and external mobile units to provide power. These satisfy current regulations in Korea as well as the USA. Larger sized batteries and more powerful backup diesel generators could provide backup power for a longer period of time but this would cost money.
Mr Koshy (IAEA) wanted to confirm that the turbine-driven secondary auxiliary cooling system that could operate without AC, as an approach to deal with SBO. He then asked whether the PORSV also functions as a cold safety valve, i.e. use the same valve.	**Mr Seo (KEPCO E&C)** explained that turbine driven pump can operate without AC power, it only requires control power from DC batteries. Regarding POSRV, during pressure transient it automatically opens to release pressure. When operators want to reduce system pressure intentionally, they can push the button and the valve will open. So the same valve works as a safety valve and a relief valve. The POSRV is certified by ASME, so it is a semi-qualified valve.
Mr Aziz (Indonesia) asked that the design	**Mr Seo (KEPCO E&C)** explained it is called AAC

also uses diesel generator which should be suitably located. He also wanted to know if the APR1400 uses gas turbine generator.	(Alternative AC) power. APR1400 does not turbine generator now, but type of generator can be chosen when the construction period starts as requested.
Mr Bykov (Rosatom) wanted clarification regarding the fact that the same relief valve for overpressure protection of pressurizer system and depressurization. He asked about the lowest level of pressure that can be reached because the cross section of the valves, the difference for overpressure protection and depressurization. For lower pressure condition, the cross section of the valve should be larger.	**Mr Seo (KEPCO E&C)** explained that POSRV is adequately sized to function overpressure protection during power operation. During low pressure condition, there is a separate relief valve (which was not discussed in the presentation).
Mr Popov (Belarus) asked that in the case of core melting, the melt gets in touch with water, how to guarantee a critical mass state is not reached.	**Mr Seo (KEPCO E&C)** responded saying that when the molten core flows down the cavity floor, it will spread out. Physically it cannot reach criticality even though there is water on top of the corium. The water comes from the IRWST which contains 4,400 ppm of boron, i.e. borated water.
Mr Susyadi (NPTDS) asked whether lessons learned from the Fukushima accident were adopted into the SAMG.	**Mr Seo (KEPCO E&C)** said that what was presented was the standard design of SAMG and after the Fukushima accident the Korean government evaluated all the operating plants and all plants under construction to assess the capability of coping with a Fukushima-like accident. As a result of the evaluation, the government enforced a total of 50 improvements. Many are for the older plants in operation but some are applicable for plants under construction, so severe accident management and emergency planning have to be upgraded according to the Korean government requirements. What was presented did not include Fukushima requirements but it is an ongoing activity in Korea. As explained, external hook up design has been in effect during the construction of the new plants so Shin-Kori 3 and 4 (the first APR1400) have this feature.
Mr Grudev (Bulgaria) asked about what code is used for the validation of APR1400 since each country may use a different internationally recognised computer code from the one used by KEPCO.	**Mr Kim (KHNP)** responded that they are planning on using their own computer code in the future but this will be validated with the standards of international codes.
Mr Bykov (Rosatom) stated that KEPCO is using an old code for validating the mixing vane which is based on a 1 dimensional experimental data base. He asked how the code is validated.	**Mr Kim (KHNP)** said that the mixing vane in the fuel assembly cannot be simulated by the code. It is still more conservative and better to consider the mixing vane on a 1 dimensional data base than not to at all.
Mr Bykov (Rosatom) responded saying that this however can influence other things such as the flow rate regime map, the reflux conditions and maybe also the supercritical heat exchange, so it is not very conservative.	**Mr Kim (KHNP)** specified that KEPCO is developing a special model for the mixing vane and maybe also the relay pipe to use in the future.
Mr Kupitz (Chairman) asked how non-vendor can actually assess IN4.2.1 that specifies RD&D needs to be defined and performed and database developed. Will the assessment require help from designer?	**Mr Kim (KHNP)** replied that if a country wants to evaluate RD&D for other reactor design, then designer's help is needed. If the country wants to develop their own reactor, it helps to know what can be done.
Ms Yang (China) asked about definition of validation in the INPRO Methodology,	**Mr Phillips (IAEA)** explained that for the best common practice, it is a determination made between vendor and

whether the V&V is a self-developed code SPACE for APR1400 calculations that matches the test results, no matter commercial or not.	regulator when they go in the process. For example, RELAP code which has been around for long time and has been subjected to many regulatory validations depending on specific problems. So generally if something is done for the first time, and depending on the importance, regulator may ask considerable amount of information to make certain the code can be applied. So, there is not a simple answer. To know validation has occurred is to see how far reactor and its licensing process are and identify actual examples of validation process. This involves data that should be proved to the regulator. So it is not just academic question but also a regulatory question. During the discussions later in the day, **Mr Beard (GE Hitachi NE Intl.)** expanded the definition of validation, as review and verification by a second independent party of the acceptance of a code or something else. This is because vendors use the term V&V (verification and validation) also as a commandment for the development of software that is not necessarily reviewed by regulator but by some expert body. **Mr Phillips (IAEA)** noted that it may vary from country to country depending upon how the TSO works. For instance, the national laboratory system in the US performs a lot of work on independent reviews of the code, experimental data, generation of additional of experimental data, etc. So it is related to regulatory process. However, he considered it is not the best practice to use a code when not certain it applies to a particular problem.
Mr Grudev (Bulgaria) asked about the two codes used for validation, SPACE and RELAP. If the deviation is more than 20%, which code provides more confidence. He added if vendor tries to sell the reactor to another country that uses some international recognized code. If the deviation is significant, it is difficult to believe, i.e. the licensing code for the national authority. So sometimes it helps but it may create an additional problem.	**Mr Kim (KHNP)** replied that currently RELAP5 is used because it is a well validated code, but the plan is to transfer to own code after several years. On the second question, he explained that Korean vendor has to validate in own country, then validate by international society such as IAEA to make more validated code.
Mr Bykov (Rosatom) wanted to highlight the presentation on fuel assembly with spacer including the mixing vanes. He indicated that RELAP5.3.3 is used, which is an old code that was created before the use of mixing vanes. The code was developed based on one dimensional experimental data base, e.g. for critical flux path a granulary table is used and there is nothing on spacer grade mixing vanes in the table. So he wanted explanation how the code is validated against design decision. Following the explanation, he disagreed that it is conservative because the mixing vanes can have influence on flow rate regime map, reflux condition or supercritical heat exchangers and not to the conservative direction.	**Mr Kim (KHNP)** explained that RELAP cannot validate mixing vanes in the fuel assembly. But if mixing vanes are considered, then the results are better without the mixing vanes. So it is in the opinion it is conservative so mixing vanes are not modelled. Following the second comment, Mr Kim indicated that a special model (maybe a RELAP too) is being developed for the mixing vanes, and the mixing vanes should be able to be modelled in the next several years.

4.5. Enhanced Candu 6 (EC6)

Ms Sabina Boyle and Mr Albert G Lee from Candu Energy Inc. presented the application of INPRO comparative assessment on safety of Enhanced Candu 6 (EC6) pertaining to lessons learned from the Fukushima Dai-ichi nuclear event. Ms Boyle provided the outline of the presentation, and introduced the purpose and objective, Fukushima Dai-ichi event and lessons learned, and selection of INPRO Criteria through comparison of EC6 design with CANDU 6 plants. There are four consequences of earthquake and subsequent tsunami to Fukushima Da-ichi Units 1-3 pointed out in the presentation:

- Prolonged loss of all AC power
- Prolonged loss of ultimate heat sinks
- Prolonged loss of accident monitoring and response capability
- Large scale societal disruption due to large radioactive releases

The key findings from the Fukushima lessons learned based on the nuclear industry including the different regulatory reviews include:

- Cliff edges in defence in depth for extreme external hazards
- Insufficient consideration for diversity for extreme external hazards
- Loss of reliable safety related parameters and communications
- External hazards can cause simultaneous events in multiple units at a site

There are two major lessons learned pointed out in the presentation:

- Plant safety functions need to be restored and maintained after extreme natural events which have caused prolonged loss of all AC power, ultimate heat sinks and accident monitoring and response capability.
- Global nuclear energy sustainability is challenged by Fukushima Dai-ichi nuclear accident. There is a need to demonstrate existing NPPs have sufficient levels of defence in depth and robustness of the plant beyond its design basis, and also to develop evolutionary NPPs with higher levels of safety

In light of extreme natural events and to address the Fukushima lessons learned, safety margins for CANDU 6 plants and EC6 design were reassessed. The assessment of preventive and mitigative measures was done by following a defence in depth logic where sequential lines of defence are assumed to be lost. The results identified potential safety improvements to the lines of defence. To address the challenge to nuclear energy sustainability, the Canadian nuclear utilities adopted the following principle for beyond design basis events: *"Practically eliminate the potential for societal disruption due to a nuclear incident by maintaining multiple and flexible barriers to severe event progression".*

It was noted that the implications of the Fukushima event on nuclear sustainability should be considered within the context of INPRO. With this respect, a comparative assessment of EC6 to CANDU 6 plants with respect to the lines of defence to withstand external hazards that can cause loss of safety functions (prolonged loss of all AC power, ultimate heat sinks and accident monitoring and response capability) was performed. Table below summarizes the selection of INPRO criteria pertaining to the Fukushima event corresponding to the DID level.

TABLE 4.5.1. INPRO Criteria Selection Pertaining to Fukushima Lessons Learned – Defence in Depth Principle

DID Level	INSAG Objective	INPRO UR	Criteria Selection	Fukushima Lessons Learned
1	Prevention of abnormal occurrences and accidents	UR1.1 - Robustness: Installations of an INS should be more robust relative to existing designs regarding system and component failures as well as operation	CR1.1.1 robustness CR1.1.2 operation	*Ensure robustness of safety systems against common cause failures* *Review basis for design against common cause failures , e.g., seismic*
3	Control of accidents within the design basis.	UR 1.3 -Design basis accidents: The frequency of occurrence of accidents should be reduced, consistent with the	CR1.3.3 safety features CR1.3.4 barriers CR1.3.5	*Review Design Basis and plant layout for infrequent and complex combinations of external events: confirm or amend Design Basis Accident*

		overall safety objectives. If an accident occurs, engineered safety features should be able to restore an installation of an INS to a controlled state, and subsequently (where relevant) to a safe shutdown state, and ensure the confinement of radioactive material. Reliance on human intervention should be minimal, and should only be required after some grace period.	controlled state	*List* *Confirm long-term cooling of spent fuel pools, and adequate monitoring provisions, in case of loss of power*
4	Control of severe plant conditions, including prevention and mitigation of the consequences of severe accidents.	UR1.4 - Release into containment: The frequency of a major release of radioactivity into the containment / confinement of an INS due to internal events should be reduced. Should a release occur, the consequences should be mitigated	CR1.4.1 frequency of release into containment	*Provide simple diverse alternative sources of power and water, in a safe location* *Effective use of probabilistic safety assessment (PSA) in risk management*
5	Mitigation of radiological consequences of significant releases of radioactive materials.	UR1.5 -Release into the environment : A major release of radioactivity from an installation of an INS should be prevented for all practical purposes, so that INS installations would not need relocation or evacuation measures outside the plant site, apart from those generic emergency measures developed for any industrial facility used for similar purpose.	CR1.5.1 frequency of release to environment	*Evaluate need for containment venting provisions to prevent uncontrolled, unfiltered and unmonitored radioactive releases*

4.5.1. Overview of EC6 Design

Mr Lee (Candu Energy Inc.) presented the evolution of EC6 (740 MWe) from CANDU 6 which has over 150 years of safe operation in five countries: Canada (1 CANDU 6 reactor in operation), Romania (2), Argentina (1), Korea (4) and China (2). It has a lifetime capability factors of approximately 90%. The EC6 design evolved from well-proven CANDU 6 with incremental changes to meet modern regulatory requirements. It combines proven and state-of-the-art technologies, has improved safety margins, operability and maintainability and increased design life from 40 years to 60 years. There are incremental changes in the technology, largely in the digital I&C.

The safety principles of CANDU 6 are based on defence in depth outlined in INSAG-3 Objectives, with 3 fundamental safety functions which are controlling the power (reactivity), cooling the fuel and confining the radioactive material (containment). There was attention given to the capability to monitor the plant to guide the operator to respond to accident conditions. With EC6, there have been improvements made, among others, in monitoring capability, and ability of operator to monitor the plant conditions and to respond to upset conditions.

The original CANDU 6 design was based on single and dual failures: single failure being a process system failure, whereas dual failure being that along with the single failure there is simultaneous failure of safety

system. For example, CANDU 6 containment was designed on the basis of LOCA plus simultaneous unrelated loss of emergency core cooling, i.e. the design should be able to cope with failure of ECCS and still maintain releases to the environment at an acceptable level.

The introduction of PSA was made very early to define and set performance target for the reliability of the safety systems, for identifying the need for redundancy and backup. One other significance difference between CANDU 6 & EC6 versus PWR is that CANDU 6 and EC6 employ two-group philosophy for redundancy of safety functions for controlling reactivity and cooling fuel. The design relies on engineered safety systems to complement the inherent design capability to cope with transients.

Mr Lee described briefly the EC6 plant flow diagram, indicating the reactor core containing low pressure heavy water moderated calandria vessel, with 380 parallel channels penetrating the vessel, each containing 12 fuel bundles. The vessel is housed in a steel-lined concrete vault, filled with light water which acts as radiation shielding during normal operating condition and heat sink under severe condition. There are 4 SGs organized such that each two SGs are connected to each half of the core. Hence in a postulated LOCA, the reactor will not simultaneous lose coolant or depressurize all of the core, i.e. only half of the core. The other benefit of having the pressure boundary at the location of the fuel is to limit the amount of cross top between fuel channel and fuel assembly, such that failure in one channel do not necessarily cascade of fuel from adjacent location. The heavy moderator provides a large heat sink that can be used in plant transients to remove heat from the core, maintain cooling of the core, and prevent the progression of DBAs to severe accident. The containment building is fairly conventional. The original CANDU 6 has been updated following the 9/11 taking into account malevolent acts.

Comparison of the design characteristics of EC6 versus CANDU 6 was provided, and also the fact that in comparison to light water reactors CANDU can maintain small excess reactivity due to online refueling. In terms of natural external events, design basis takes into account seismic, flood and tornado. On seismicity, as shown below, the design basis earthquake has been increased and in addition EC6 is using CANDU 6's conventional hard rock ground response spectrum and superimposed it with a rather unique universal hazard spectrum to represent the eastern North American spectrum which at high frequency shows the higher g-force and g-level. For other countries, standard spectrum will be used and the seismic loading will be adjusted accordingly.

TABLE 4.5.2. Design Basis of CANDU 6 versus EC6 for Natural External Events

Event	CANDU 6	EC6
Earthquake	• Original design basis earthquake applied a return frequency of 10^{-3} years; reassessments used 10^{-4} where the HCLPF exceeds 0.3 g PGA • Design Basis Earthquake (DBE): seismic level up to 0.2 g Peak Ground Acceleration (generic design)	• Return frequency of 10^{-4} years, as per CSA standard N289.1-08; SMA measured in terms of HCLPF which exceeds 0.5 g PGA (1.67×DBE) • DBE seismic level up to 0.3 Peak Ground Acceleration
External Flood	• Return frequency of 10^{-4} years if sufficient data; otherwise use maximum observed flood plus margin • Design Basis Flood: Dry site concept	• Return frequency of 10^{-4} years if sufficient data; otherwise use maximum observed flood plus margin • Design Basis Flood: Dry site concept: • 0.60 m above max. ground water level

Tornado	• Return frequency up to 10^{-6} per year.
	• Design Basis Tornado (F4 on Fujita Scale Classification of Tornado Wind Intensity):
	  • Maximum wind speed: 420 km/h;
	  • Translational wind speed: 92 km/h
		• Return frequency of 10^{-7} per year.
		• Design Basis Tornado (F5 on Fujita Scale Classification of Tornado Wind Intensity):
		  • Maximum wind speed: 483 km/h
		  • Translational wind speed: 97 km/h

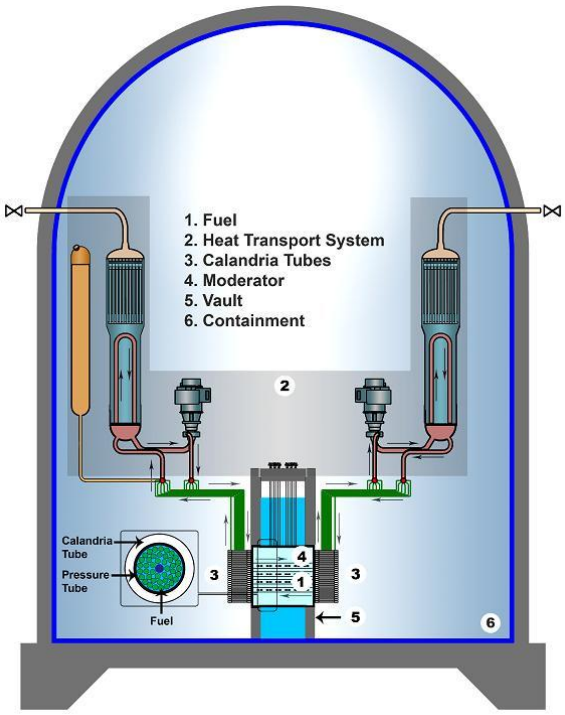


FIGURE 4.5.1. Multiple Diverse Barriers in EC6

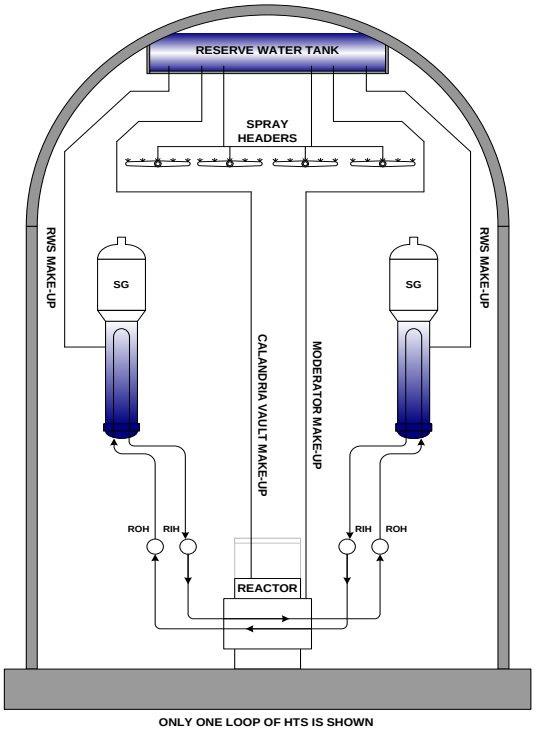


FIGURE 4.5.2. EC6 Passive Safety Features

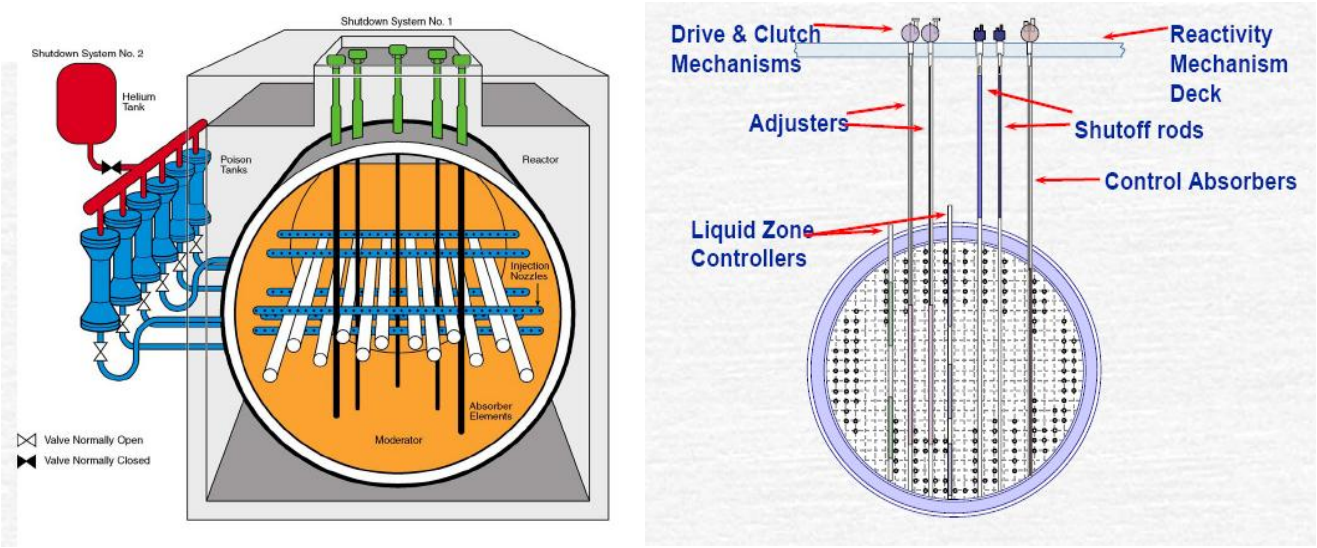


FIGURE 4.5.3. Reactivity Control Devices

Important safety features of EC6 in comparison with CANDU 6 were discussed and summarized in the table below. The containment design pressure has been increased with a lower leakage rate, containment wall has been increased to withstand conventional commercial airline crashes, and there are more safety systems by upgrading emergency heat removal system (for non LOCA) from safety support system to a full safety system. In addition, severe accident management engineering features were noted, among others:

- Increased cooling capability: Water is located in several places to provide heat sink to give time in case of SBO, estimated for around 5 days without resulting in onset core damage - without bringing reserve water supplies.
- Means of monitoring the status of plant can be done in multiple locations, including Emergency Support Centre which has been hardened against natural occurring external event.
- Increase number of DGs, i.e. dedicated DG for heat removal during severe accident recovery heat removal system

- The installed AC power supply is backed up by on site mobile power supplies stored in a safe location, also recommended backed up with an off-site depo containing mobile power and water sources. This is consistent with FLEX strategy employed in the US.

TABLE 4.5.3. Important Safety Features of CANDU 6 versus EC6

Feature	CANDU 6	EC6
Containment Design Pressure	124 kPa (g) design pressure with dousing spray from Dousing Tank during LOCA	400 kPa (g) design pressure based on main steam line break

Containment Design	- 1.07 m thick containment wall
	- Epoxy liner with 0.5vol%/d leakage rate 124 kPa(g)
	  - Controlled leakage up to 235 kPa(g)
		- 1.5 m thick containment wall
		- Steel liner with 0.2vol%/d leakage rate at 400 kPA(g)
		  - Controlled leakage up to 600 kPa(g)

Safety Systems	- Shutdown System 1 - Shutdown System 2 - Emergency Core Cooling System - Containment System	- Shutdown System 1 - Shutdown System 2 - Emergency Core Cooling System - Containment System - Emergency Heat Removal System
Severe accident management engineering features		
Water inventory in reactor	3,000 Mg of water available for passive heat removal: - 2,056 Mg in Dousing Tank - 215 Mg HPECC - 240 Mg D_2O in calandria vessel - 520 Mg H_2O in calandria vault - 190 Mg D_2O in HTS	3,000 Mg of water available for passive heat removal: - 2,056 Mg in RWT - 215 Mg HPECC - 240 Mg D_2O in calandria vessel - 520 Mg H_2O in calandria vault - 190 Mg D_20 in HTS
Control of reactivity	H_2O make-up to calandria vault	H_2O make-up to calandria vessel
Extended Cooling Capability	Make-up to the calandria vault via portable water source for some CANDU 6 plants	Severe Accident Recovery and Heat Removal System (SARHRS) to extend cooling makeup capability
Shielding against radiation	Containment wall (1.07 m)	Containment wall (1.5 m)
Control of planned radioactive releases, as well as limitation of accidental radioactive releases	- Emergency Filtered Containment Venting Systems added to some CANDU 6 plants, initiated at 200 kPa(g) - 33 Passive Autocatalytic Recombiners added to some plants	- Emergency Filtered Containment Venting System, initiated above 400 kPa(g) - 33 Passive Autocatalytic Recombiners - RWT low flow spray for containment heat removal and pressure suppression
Means of monitoring the status of the plant	Safety Monitoring System in MCR and SCA	Safety Monitoring System in MCR, SCR and Emergency Support Centre
Seismic Design	Up to 0.2 g peak ground acceleration	Up to 0.3 g peak ground acceleration
Control Room Availability during Seismic Design Basis Event (DBE)	- Secondary Control Room (SCR)	- Main Control Room (MCR) - Secondary Control Room (SCR) - Hardened Emergency Support Centre
Diesel Generators (DGs)	- 2 Standby DGs per unit - 2 Emergency Power System (EPS), common for 2 units	- 2 Standby DGs per unit - 2 EPS per unit - Dedicated DG for SARHRS
Battery Back-up	- 1 hour Group 1 batteries (battery life for CANDU 6 plants has been extended by re-distribution of loads)	- 1 hour Group 1 batteries - 24 hour UPS batteries - 8 hour batteries for Safety Monitoring
On-site Portable Power Supplies	Provisions for portable power added to CANDU 6 plants to address Fukushima	One portable power supply/unit

	lessons learned	

For multiple diverse barriers for prevention of releases, there are fuel sheath, HTS (including pressure tubes), calandria tubes, cool low pressure moderator, cool low pressure reactor vault and steel-lined concrete containment structure. For the passive safety features, there are:

- Elevated reserve water tank (top of containment), which can be gravity fed to (i) SGs to transfer heat to the ultimate heat sink in case of loss of cooling to SGs, (ii) moderator to arrest progression of severe accident, (iii) calandria vault to maintain corium within the calandria system in case severe accident progresses beyond the point where the fuel channel can be preserved in the primary vessel, and (iv) low-flow spray system in the containment for pressure and heat suppression in the containment building.
- All buildings where safety system equipment are housed in containment which is robust, seismically qualified and aircraft crash-proof. They are located on site such that it is not possible for a single aircraft crash to cause both MCR and SCR to be simultaneously rendered unavailable. The same applies for the location of pumps and power supplies for the safety system, i.e. located in diverse locations.

There are 3 measures for controlling hydrogen produced following accident sequences where zirconium-steam reactions occur such as LOCA plus loss of emergency core cooling - as summarized below.

	CANDU 6	**EC6**
Mixing	- Local air coolers with Class IV/III power or EPS - Natural convection with loss of Class IV/III power and EPS - Level 3: DBAs	- Local air coolers with Class IV/III power or EPS - Natural convection with loss of Class IV/III power and EPS - Level 3: DBAs
Igniters	44 with Class IV/III power (manual connection to EPS) - Level 3: DBAs with Class III power or EPS - Level 4: BDBAs, including limited core damage accidents and severe accidents	44 with Class IV/III power (manual connection to EPS) - Level 3: DBAs with Class III power or EPS - Level 4: BDBAs, including limited core damage accidents and severe accidents
Passive autocatalytic recombiners (PARs)	Up to 33 added to some CANDU 6 plants - Level 3: DBAs with loss of Class III power and EPS - Level 4: BDBAs, including limited core damage accidents and severe accidents	33 - Level 4: BDBAs, including limited core damage accidents and severe accidents

There is also emergency filtered containment venting system used to protect the structural integrity and leak tightness of the containment building for severe accident events where containment pressure significantly exceed the design pressure of the containment.

The concepts of core damage accident for a PHWR were discussed as follows:

- Limited core damage accident

  It is defined as accident that occurs within one pressure tube or one fuel channel. Loss of heat sinks can result in the pressure tubes deforming into contact with the calandria tubes. A heat rejection pathway to the moderator is established, as a result of the inherent separation of the primary coolant from the moderator. Fuel continues to be retained within the fuel channels and the fuel channel lattice geometry is essentially unaffected.

- Moderator System

  Moderator provides an additional heat sink to remove heat especially from damaged single channel. The system normally circulates the heavy water moderator to remove the heat generated by the fission reaction. Event progression may be halted at a limited core damage state using the moderator system. Heat is transferred via heat exchangers to the recirculated cooling water system and to the ultimate heat sink. If the active moderator heat sink is unavailable, the large volume of heavy water in the calandria vessel (~240 tonnes) is heated up and boiled off as a passive heat sink (this lasts about 5 hours)

- Severe core damage accident

It is defined as 2 or more fuel channel failures. If the moderator heat sink fails (or is exhausted), the fuel channels will fail and the core channel debris (corium) will settle to the bottom of the calandria vessel. With the corium retained inside the calandria vessel, this is an in-vessel core damage state, i.e. the final stage of accident progression. In this state the vault inventory can be cooled actively using the end shield cooling system, which circulates the inventory with pumps and transfers heat to the recirculated cooling water system via heat exchangers. Without active cooling, the large calandria vault inventory (~520 tonnes of light water) starts boiling at 10+ hours (for worst case event) and will gradually boil off over the course of ~20 hours. It was noted that beyond in vessel retention strategy, there is also a refractory layer of (consumable) material on reactor vault floor which will react with the corium and arrest the progression of corium to a molten corium-concrete interaction.

4.5.2. INPRO assessment in the area of safety of EC6

4.5.2.1. EP1.1.1.1: Margins of Design

- Improved sub criticality margins

  There are 3 systems for managing reactivity:

  - Reactor Regulating System. This includes 6 liquid zone control units with 14 compartments containing mixed light water or helium; mechanical absorber rods, adjuster rods, moderator poison system. The reactor rating system can maintain shutdown indefinitely.
  - Shut-Off Rods. There are additional 4 shut-off rods to achieve an improved shutdown margin. To maintain reactor's subcritical in long term after cool down and decay of transient poison, in addition to shutoff rods there are additional 10 supplemental absorber rods to achieve improved subcritical margin after design basis earthquake (DBE). The supplemental rods are to be inserted manually 36 hours after SDS1 action has taken place. In comparison CANDU 6 requires addition boron or gadolinium poison from the moderator poison system to achieve long term sub criticality.
  - SDS2 which injects of gadolinium nitrate into the moderator to maintain reactor sub criticality in the long term.

  It was noted that the 3 systems are independent of each other, addressing redundancy and diversity.

Shutdown Designs	**CANDU 6**	**EC6**
Reactor Regulating System	Liquid Zone Control Units, CARs, Adjusters, Moderator Poison System	Liquid Zone Control Units, CARs, Adjusters, Moderator Poison System, Supplemental Absorber Rods
Liquid Zone Control Units	6 units with 14 zones	6 units with 14 zones
# of Control Absorber Rods (CARs)	4	4
# of Adjusters	21	11
Moderator Poison System	Boric acid addition to Moderator System	Boric acid addition to Moderator System
# of Supplemental Absorber Rods	0	10
# of Shut-off Rods	28	32
# of Poison Injection Nozzles	6	6

- Improved LOCA margins

  Both EC6 and CANDU 6 have a positive coolant void reactivity (CVR) coefficient as a consequence of using natural uranium fuel and the geometric arrangement of the lattice pitch. Improvements are made to reduce the total amount of CVR and peak positive reactivity during LOCA. There is a 12% pressure tube thickness increase for EC6 design to reduce creep during 30 year operation.

Design Characteristics	**Unit**	**CANDU 6**	**EC6**
Coolant Void Reactivity (Limit of operating envelope)	$	~3.3	~3
Peak positive reactivity inserted during LOCA	$	~0.9	~0.61

- Improved containment design margins

The EC6 containment design pressure is based on main steam line break peak pressure plus design margin, in comparison with the CANDU 6 plant containment design pressure which is based on LOCA peak pressure plus design margin. Improvements of containment design for EC6 include increase of design pressure, containment wall thickness and type and pressure of the airlock seals. The EC6 containment structure meets post 9/11 requirements for aircraft crash. The leak rate has also been reduced.

Design Features	CANDU 6	EC6
Containment Design	- 124 kPa (g) design pressure with dousing tank - 1.07 m containment wall thickness - Inflatable airlock seals, good to 235 kPa(g)	- 400 kPa (g) design pressure with Reserve Water Tank (RWT) - 1.5 m containment wall thickness - Solid face type airlock seals, good to 600 kPa(g)
Leak Rate	Epoxy liner with 0.5%/d leakage rate	Steel liner with 0.2%/d leakage rate

- Improved seismic margins

The EC6 DBE level is above the CANDU 6 level, with the reduction of return frequency from 10^{-3} years to 10^{-4} years; also enough seismic margin is available for the seismically qualified systems and components to withstand a 0.5 g horizontal peak ground acceleration (PGA).

The EC6 provides additional provisions for seismic function capability:

- Control: 10 seismically qualified supplemental absorber rods for reactor regulating system (RRS) to prevent re-criticality after transient fission product poisons have decayed when using SDS1
- Remove decay heat: Seismically qualified emergency heat removal system (EHRS), 2 seismically qualified emergency diesel generators per unit
- Contain: Seismically qualified connection for the spent fuel bay and the reception bay
- Monitor: Seismically qualified main control room

Event	CANDU 6	EC6
Earthquake	Original DBE applied a return frequency of 10^{-3} years; reassessments used 10^{-4} where the HCLPF (High Confidence of Low Probability of Failure) exceeds 0.3 g PGA (Peak Ground Acceleration). DBE: seismic level up to 0.2 g PGA (generic design)	Return frequency of 10^{-4} years, as per CSA standard N289.1-08; SMA measured in terms of HCLPF which exceeds 0.5g PGA. DBE seismic level up to 0.3 PGA (Horizontal PGA at DBE level is 0.21 g for CSA based spectra and 0.3 g for Uniform Hazard Spectra (UHS))

- Improved flood margin

The EC6 applies CANDU 6 'dry site' condition philosophy for flood protection. The EC6 flood protection: *The building grade level is set at an elevation which provides significant margin against the maximum external flood.* The flood margin for the location of the system, structure and component (SSC) has been increased from 0.50 m above the heavy rain coincident to 0.60 m.

Event	CANDU 6	EC6
External Flood	- Return frequency of 10^{-4} years if sufficient data; otherwise use maximum observed flood plus margin - Design Basis Flood: Dry site concept (0.50 m margin for heavy rain coincident with design basis flood)	- Return frequency of 10^{-4} years if sufficient data; otherwise use maximum observed flood plus margin - Design Basis Flood: Dry site concept (0.60 m above max. ground water level)

- Improved heat sink margin

CANDU has the following features: (i) heat distributed over large fuel volume, (ii) many cooling sources, and (iii) both active and passive systems and multiple barriers to prevent severe accidents.

The EC6 design further improved the heat sink margin by incorporating the following features:

- Addition of a SARHRS to extend cooling make-up capability to calandria vessel and calandria vault. There is a dedicated SARHRS diesel generator to draw water for cooling of heat exchangers and/or for make-up to the Reserve Water Tank from the local water supply (lake or river)
- Improved heat sink capability by seismically qualifying shield cooling system piping inside the reactor building for use by SARHRS
- For a severe core damage accident, SARHRS (or SCS) is used to remove heat from corium to maintain in-vessel retention strategy

On SBO or loss of all AC Power Event (Loss of Class IV (off-site and on-site) + Class III Standby diesel generators + Emergency Power Supply diesel generators), the diesel generator for SARHRS would be assumingly unavailable and there is up to 5 day time for the operator to restore either power to SARHRS to use the water as a heat sink, or to substitute by supplying power from mobile diesel generator or gas turbines or add a fire truck.

System Availability & Consequence	**CANDU 6**		**EC6**
SDS1/SDS2	×	×	×
SARHRS	-	-	No
HPECC	×	No	×
Batteries	×	No	×
RWT to SG	× (Dousing)	No	×
Consequence			
Containment integrity	Containment failure after ~6 days	Containment failure after ~5 days*	Containment integrity maintained – prevention of severe core damage

* If calandria vault make-up is available, as for some CANDU 6 stations, containment integrity is maintained

It was concluded that EC6 design provides added safety margins for reactor shutdown, sub criticality, containment design, seismic margin, flood margin and heat sink margin. Therefore, evaluation parameter EP1.1.1.1 has been demonstrated by the EC6 design.

4.5.2.2. EP1.1.1.5 Redundancy of System

- Reactivity control

The EC6 design provides additional means for reactivity control in comparison to the CANDU 6.

	CANDU 6	**EC6**
Level 1 and 2 DID	- 14 liquid control zones - 4 mechanical Control Absorber Rods - 21 Adjuster Rods - Gadolinium or boron addition via Moderator Poison System	- 14 liquid control zones - 4 mechanical Control Absorber Rods - 11 Adjuster Rods - Gadolinium or boron addition via Moderator Poison System
Level 3 DID	SDS 1: 28 SS-clad cadmium rods Moderator Poison System after Xe-135 poison decayed	SDS 1: *32 SS-clad cadmium rods* Moderator Poison System after Xe-135 poison decayed or *10 Supplemental Absorber Rods after Xe-135 poison decayed*
Level 3 DID	SDS 2 : 6 injection nozzles of gadolinium	SDS 2: 6 injection nozzles of gadolinium
Overpoison GSS (Guaranteed Shutdown State)	Adding gadolinium poison to moderator	Adding gadolinium poison to moderator
Rod-Based GSS (Guaranteed Shutdown State)	Stable sub-critical state achieved by - inserting 28 shut-off rods - locking 21 adjuster rods - inserting 4 control absorber rods	Stable sub-critical state achieved by - inserting 32 shut-off rods - locking 11 adjuster rods - inserting 4 control absorber rods

	- adding small amount of B/Gadolinium	- *inserting 10 Supplemental Absorber Rods*

- Heat sink capability

The CANDU 6 and EC6 designs have multiple heat transfer paths to remove decay heat from fuel to the ultimate heat sink. It was noted that for DBA and BDBA there is an upgrade of Emergency Heat Removal System (EHRS) and SARHRS as compared to CANDU 6. There is also emergency filter containment system as a last resort to reduce pressure in the containment. Also for both CANDU 6 and EC6, there is provision for portable water and power during BDBA.

Plant State	CANDU 6	EC6
Normal Operation and AOO	Main Feed water System including - Auxiliary Feed water - Shutdown Cooling System - Recirculated Water /Raw Service Water System (RCW/RSW) - LACs	Main Feed water System including - Auxiliary Feed water - Shutdown Cooling System - RCW/RSW - LACs
DBA	ECCS *Emergency Water Supply System Dousing* Shutdown Cooling System RCW/RSW LACs	ECCS *EHRS* Shutdown Cooling System RCW/RSW System LACs
BDBA	- Moderator System - Shield Cooling System - RCW/RSW - HPECC - Emergency Water Supply System - EFCVS (added to some CANDU 6 plants) - Emergency Mitigating Equipment (portable power and water)	- Moderator System - Shield Cooling System - RCS/RSW - HPECC - *EHRS* - *SARHRS* - EFCVS - Emergency Mitigating Equipment

The independence of the SARHRS for providing heat sink capability was briefly discussed, noted it has separate diesel generator and pump. The SARHRS can also provide water to calandria vault and vessel from the RWT.

Improved capability of the EC6 design for heat sinks is summarized below.

Design Feature	CANDU 6	EC6
Safety Systems	- Shutdown System 1 - Shutdown System 2 - Emergency Core Cooling System - Containment System	- Shutdown System 1 - Shutdown System 2 - Emergency Core Cooling System - Containment System - Emergency Heat Removal System
Water inventory in reactor	3 000 Mg of water available for passive heat removal: - 2056 Mg in Dousing Tank - 215 Mg HPECC - 240 Mg D_20 in calandria vessel - 520 Mg H_2O in calandria vault - 190 Mg D_2O in HTS	3 000 Mg of water available for passive heat removal: - 2056 Mg in RWT - 215 Mg HPECC - 240 Mg D_20 in calandria vessel - 520 Mg H_2O in calandria vault - 190 Mg D_2O in HTS
Extended cooling capability	- Make-up to the calandria vault via portable water source for some CANDU 6 plants	- Severe Accident Recovery and Heat Removal System for cooling water make-up to RWT

		- Make-up to the calandria vessel from the RWT - Make-up to the calandria vault from the RWT
Spent Fuel Bay (SFB) – Long term cooling	Firewater make-up provisions upgraded to seismically qualified in some CANDU 6 plants to address Fukushima Lessons Learned	- Make-up to SFB from seismically qualified firewater - SFB walls qualified to 100°C for long-term - Seismically qualified monitoring provisions for SFB level, temperature and radioactivity detection

It was concluded that the EC6 design provides increased redundancy by means of: (i) Reactivity control during normal operation, (ii) Increased shutdown margin for SDS1 during DBAs, and (iii) Increased heat sink capability during severe accidents, as compared to CANDU 6. Therefore, evaluation parameter EP1.1.1.5 has been demonstrated by the EC6 design.

4.5.2.3. EP1.1.2.7 Increased availability / capability

The EC6 design builds on CANDU 6 lessons learned to achieve higher availability and capability of the plant. This accomplished by implementation of extensive Equipment Health Monitoring System (EHM) and reduction in maintenance induced equipment failures. The capacity factor has been increased from 85% to 92%, lifetime from 40 to 60 years, and outage interval has been extended to 3 years.

Availability / Capability	CANDU 6	EC6
Capability Factor	Lifetime capacity factor greater than 85%	Lifetime capability factor greater than 92%, with a year to year plant capability factor greater than 94%.
Design Life	40 years	60 years
Outage Interval/ Duration	Between 18 and 30 months/~30 days	3 years/30 days

It was concluded that the EC6 design provides higher availability and capacity factor resulting with a longer design life, as compared to CANDU 6. Therefore, evaluation parameter EP1.1.2.7 has been demonstrated by the EC6 design.

4.5.2.4. CR1.3.3 Safety Features

The EC6 design has additional protection against single failures for safety groups. Redundancy is added to address single failures identified by failure modes and effects analyses for all systems performing fundamental safety functions. Each safety system in EC6 and CANDU 6 is designed to satisfy regulatory requirements for less than 1E-3 failure on demand. It was noted that the requirement is still met for EC6 despite the increase in the number of components in the safety system, e.g. increase in trip parameters and coverage of safety systems. One challenge to fulfil this requirement is during longer outage there are always components that cannot be tested while reactor is in operation hence they must be procured to reliability standards to meet 1E-3.

It was concluded that the EC6 is designed with similar reliability targets as the CANDU 6 plants, which satisfies the latest regulatory requirements. The EC6 design has increased the protection against single failures for safety groups relative to the CANDU 6 plants*.

(**Mr Lee noted the one universal exception not peculiar to CANDU is it is not practical to introduce redundancy for check valve on the HP ECCS. For CANDU, it has been demonstrated that with four injection points for HP ECCS into pressure tubes, adequate performance is obtained if only one check valve fails, i.e. there is no need to have four to meet the objectives in terms of protecting the fuel and removing heat from the core.*)

4.5.2.5. CR1.3.4 Barriers

The EC6 design provides ‘extended’ defence in depth provisions for DID Level 1-5.

DID Level	CANDU 6	EC6 Additions
1 – Prevention of abnormal operation and	- Conservatism and consists of safety features for	- Additional redundancy in process systems for increased outage

failures	prevention of abnormal operation and failures	intervals*
2 - Control of abnormal operation and detection of failures.	- Reliable reactor regulating systems (RRS)	- Additional reactor regulating systems (RRS) parameters to respond to AOOs**
3- Control of accidents within design basis	- Engineered Safety features - SDS1, SDS2, ECCS, Containment	- Added safety margins (Sped up SDS) - Lower leakage rate from containment
4 – Control of severe plant conditions	- SAMGs - Portable sources of power and water - EFCVS at 200 kPa(g)	- Added SARHRS - Portable sources of power and water to multiple connections and systems - EFCVS at ~400 kPa(g) - Added severe accident monitoring parameters
5 – Mitigation of radiological consequences	- On-site and off-site emergency response centers	- Hardened on-site emergency response center***

Added explanation during presentation:
* and also lower forced outage rate. Hence greater number in balance of plant (BOP) can be tolerated without shutting down the plant, and more online maintenance can be done.
** Adding setback and stepback parameters to RRS
*** Safety parameter monitoring system display will provide plant information to plant emergency response personnel, and control operator will have access in either the main control room, secondary control area or technical support center. They are set up so as parallel and redundant in term of communication pathways.

For both EC6 and CANDU 6, at least two confinement barriers remain secured during DBA, namely (i) fuel matrix – no fuel melting, and (ii) containment. The safety system is designed to be sufficiently capable and fast acting. In case of large LOCA, the barriers include (i) contained fuel matrix, and (ii) intact containment boundary, and (iii) fuel sheath and pressure boundary be maintained in unbroken loop.

It was concluded that depending on the accident severity, a minimum of two confinement barriers can be maintained by CANDU 6 plants and the same is maintained by the EC6 design. The fuel channels are expected to remain intact, as long as they are surrounded by water. The EC6 improves safety margins for containment design by increasing containment design pressure from 124 kPa(g) to 400 kPa(g) with a conservative design margin. The EC6 is designed with 'enhanced' defence in depth provisions in comparison to the CANDU 6 plants, resulting in improved barriers to protect the plant, public and environment.

4.5.2.6. CR1.3.5 Controlled State

There is improvement in shutdown margin for EC6.

Design Features	CANDU 6	EC6 Additions
Level 2 (AOO function) DID	Setback: 9 parameters Stepback: 8 parameters	Setback: 14 parameters Stepback: 10 parameters
Level 3 (DBA reactivity devices) DID	- 28 shut-off rods and manual addition of liquid poison - Poison moderator	- 32 shut-off rods and manual addition of liquid poison or 10 seismically qualified Supplement Rods - Improved safety margin for SDS1 - Poison moderator
Level 3 DID	SDS1: 10 trip parameters SDS2: 9 trip parameters	SDS1: 15 trip parameters SDS2: 14 trip parameters

Level 3, 4 and 5 DID	- 1 hour Group 1 UPS - Battery life for CANDU 6 plants has been extended based on distribution of loads
	- 38 post accident monitoring (PAM) parameters for Operator Action
		- 1 hour Group 1 UPS
		- 24 hour Group 2 UPS
		- Safety computers available for 8 hours
		- Increased post accident monitoring (PAM) for Operator Action
		  - 22 existing parameters SQed
		  - 7 parameters added for DBA (Level 3)
		  - 36 parameters added for BDBA (Level 4 & 5)

It was concluded that EC6 design has improved margins and capability for shutdown in reaching and maintaining controlled state.

4.5.2.7. CR1.4.1 Major release into containment

In terms of CDF, the EC6 is designed to meet CDF safety goal of 1E-5/year with margin, as per regulatory requirements, i.e. more rigorous safety goals than CANDU 6 plants with 1E-4/year. It was noted the contribution to the CDF for internal events is of an order magnitude lower. In terms of fire, there is a substantial reduction to the CDF related to internal fire as a direct result of implementing the latest nuclear standards and building standards for fire protection. Meanwhile contribution for shutdown, flood and seismic are not much different compared to the current CANDU 6.

Therefore CR 1.4.1 has been demonstrated by the EC6 design.

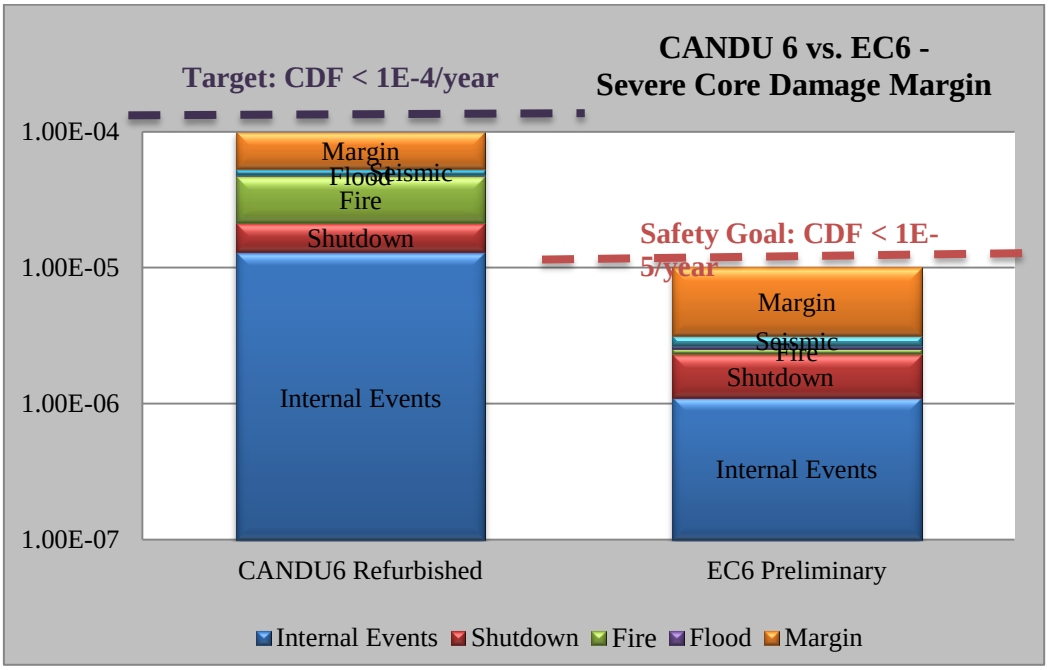


FIGURE 4.5.4. Comparison of core damage frequency (CDF) for EC6 vs CANDU 6

4.5.2.8. CR1.5.1 Major release to environment

It was noted that although the AL asked for improved large release frequency (LRF), the EC6 is designed to meet not only large release frequency (LRF) but also small release frequency (SRF) safety goals with margin, as per regulatory requirements (10 times lower than that for the existing CANDU 6 plants). The reason is the indicators for the SRF are the contribution of radionuclides that could be discharged to the environment that would affect the decision for temporary evacuation, with I-131 as a benchmark measure. For long term evacuation and possible relocation of the populous, the benchmark is Cs-137 simply because of the difference in half lives.

- The SRF is defined as the sum of frequencies of all event sequences that can lead to release to the environment of more than 10^{15} Bq of iodine-131 is less than 10^{-5} per reactor year. A greater release may require temporary evacuation of the local population.
- The LRF is defined as the sum of frequencies of all events sequences that can lead to release to the environment of more than 10^{14} Bq of cesium-137 is less than 10^{-6} per reactor year. A greater release may require long term evacuation of the local population.

The EC6 design also includes an emergency filtered containment venting system (EFCVS) as a complementary design feature to prevent uncontrolled, unmonitored and unfiltered releases. Some CANDU 6 plants have or are in the process of installing an EFCVS.

Comparison of CANDU 6 versus EC6 LRF margin is provided in the figure below but without breakdown for shutdown, flood, and seismic for EC6. However, a smaller margin is resulted for requirement of being an order of magnitude lower, reflecting the practicality to achieve the improvement.

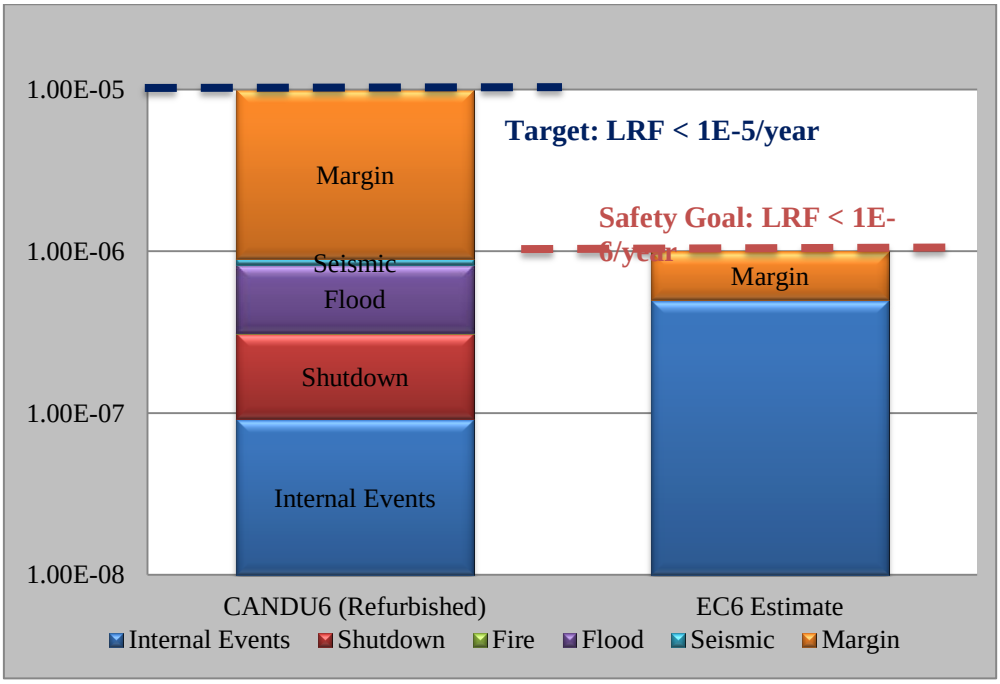


FIGURE 4.5.5. Comparison of large release frequency (LRF) for EC6 vs CANDU 6

It was concluded that EC6 is designed to meet more rigorous safety goals than CANDU 6 plants. Therefore CR 1.5.1 has been demonstrated by the EC6 design.

4.5.2.9. CR1.6.1 Independence of DID levels

The EC6 has added improvements relative to CANDU 6 plants to provide more independence between Levels 1 through 5 for the following safety functions:

- Control of reactivity
  With respect to control reactivity, in Level 1 and 2 there are RRS to control reactivity and shutdown the plant for AOOs, and in Level 3 there are 2 independent SDS to maintain sub criticality in the long term.
  It was noted that for control of reactivity for fresh fuel and spent fuel storage, since EC6 and CANDU 6 use natural uranium there is no need to control reactivity in the fresh fuel storage as long as there is no heavy water. In the spent fuel bay, no poisoning or spacing of fuel bundles, or criticality control is required.
- Removal of heat from core
  In Level 1 and 2, there is HTS. In Level 3, in case of LOCA there is 3-staged ECCS: high pressure ECCS with water tank in an aircraft crash protected located outside the containment building, medium pressure ECCS using water in the reserve water tank at the top of the containment building, and low pressure ECCS i.e. a set of pumps taking water from sumps inside the containment building and recirculating by pumping it up into the reserve water tank and injecting by gravity back into the core. For non-LOCA events where heat transport pressure boundary remains intact, the EHRS will inject water into the secondary side of SG for DBA. Also in the event of seismic occurring after a LOCA, the EHRS will provide water to the secondary side of ECC heat exchanger to remove heat from the core. For BDBA, if the EHRS is impaired, there is recovery heat removal system which will transfer heat to the ultimate heat sink. For limited and severe core damaged accidents, the water inventory in the calandria vessel will transfer heat from the core to the ultimate heat sink. This can be done if the moderator system is still intact and running, if not then injecting water from the severe accident heat recovery removal system, or resorting to mobile water sources like fire truck. If all fail and the corium forms, the water inventory in the calandria vault transfers heat from the calandria vessel to the ultimate heat sink and it can be replenished by SARHRS or mobile water.
- Confinement of radioactive materials, e.g. primary function of containment
- Shielding against radiation
  Mr Lee noted one of the side benefits of increasing thickness of containment wall from 0.7 m to 1.5 m, which is to lower on-site radiation.
- Control of planned radioactive releases and limitation of accidental radioactive releases

In the case of low containment pressure, the leakage rate is made sufficient low up to 600 kPA(g). In the case of severe accident, the EFCVS will be used.

- Monitoring to guide operator actions.
  There is addition of large number of parameters for severe accident monitoring and DBAs. The monitoring system is provided in the emergency response system, main control room secondary control room and technical support center. The system in each location system has its own dedicated UPS sufficient for 8 hours, and a separate provision to connect to portable generator to recharge the battery.

Overall there are more barriers, hence larger safety margins are achieved in each level, as demonstrated by lower values for CDF and LRF safety goals.

It was concluded that the EC6 is designed to meet more stringent safety goals and provides complementary design features to cope with system challenges, that strengthen DID levels. Therefore CR 1.6.1 has been demonstrated.

In summarizing the assessment presentation, Mr Lee indicated that comparative assessment of EC6 design to CANDU 6 plants take into account Fukushima lessons learned based on the DID principle in the INPRO Methodology. The EC6 design has improved safety margins for DID Level 1 to 3, and also has additional capability in Level 4 and 5. For Level 4 and 5, the EC6 design and CANDU 6 plants can rely on similar provisions for flexible emergency mitigating equipment. The more robust and lower leakage containment means more time to implement off-site emergency procedures in Level 5. The EC6 design improvements have practically eliminated the potential for societal disruption due to a nuclear incident by maintaining multiple and flexible barriers to severe event progression.

4.5.3. Availability of References

The availability of data for INPRO assessor depends on the level of data needed. Some technical information is available in the technical summary while other can be made available upon request or requires a commercial arrangement. It is possible for INPRO assessor to obtain safety analysis report that was submitted to the Canadian Regulator, already made available to the public - with detailed sensitive confidential information already removed. Data such as on SSC and DID approach to the design are not readily available, e.g. reliability data for EC6 for SSC is a proprietary database because it includes information from operating CANDU 6 plants.

For the purpose of the Dialogue Forum, Candu Energy Inc. has prepared a document on INPRO comparative assessment for the EC6 pertaining to lessons learned from the Fukushima Dai-ichi nuclear event. An EC6 technical summary document is also made available for reference.

Overall, involvement of reactor vendor is considered important to provide technical information and interpretation of the design intent to have a high quality assessment.

4.5.4. Discussion

Questions	Answers
Mr Zhou (China) asked whether EC6 can be used for mixed fuel	**Mr Lee (Candu Energy Inc.)** replied that EC6 can use a variety of fuel mixes, including recycle of spent PWR fuel down blended to enrichment equal to natural U without the need any further changes to the reactor or a safety case. It is also possible to achieve higher burn-up in each of the individual fuel bundles with recovered uranium maintaining enrichment above natural at 1.21 wt% U-235 range, double the exit burn-up of the bundle, reduced number of fuel bundles discharged per year, and consume stockpile of recovered U from spent PWR and BWR fuel. EC6 can also able to fuel reactor with other recovered fuel such as MOX fuel. A whole core MOX will require some changes to the reactivity control devices.
Ms Bayazitova (Kazakhstan) asked whether D_2O is also used for SG or only as a coolant. She also asked about the economics of heavy water.	**Mr Lee (Candu Energy Inc.)** explained that the coolant in HTS which removes heat directly from the fuel to SG is heavy water and the secondary side uses light water. The heavy water economics was not covered in the presentation. The plant comes with initial inventory of heavy water to fill the HTS and calandria vessel, i.e. 190 Mg in HTS and 420 Mg in the moderator.

	There is built-in small continuous heavy water upgrader, so heavy water lost / spilled over time from refuelling can be collected and upgraded – for the entire 60 years period.
Mr Dulera (India) questioned about the lifetime of pressure tubes and improvements of pressure tubes. He later asked about the life limiting factor of the pressure tube - in comparison with hydriding and oxidation for zircaloy tube.	**Mr Lee (Candu Energy Inc.)** said that pressure tube in EC6 has a lifetime of 30 effective full power years (EFPY), compared to 25 EFPY for CANDU 6. So after 30 EFPY, it is necessary to do midlife replacement of the pressure tubes. He noted elongation and creep as the life limiting factors for the pressure tubes.
Mr Benkovskyi (Ukraine) asked comparison of HWR and PWR on radioactive waste flow (whether much more) He also asked about the price comparison per kWe installed capacity for CANDU reactors.	**Mr Lee (Candu Energy Inc.)** explained that in terms of radioactive wastes for EC6, there are several categories. For 4560 bundles per year (~one core load), for other types there are similar amounts and types of liquid waste and also solid waste similar to that for PWR or BWR, e.g. resulting from maintenance of SGs such as cleaning and of turbines. Similar volumes and types. But there will be small amount of liquid waste containing tritium given the inability to recover 100% of spilled heavy water from routine maintenance. The historical liquid discharge of tritium is 18 Bq/l which is somewhat higher than that from PWR, but not an order of magnitude higher. On price comparison, Mr Kupitz (Chairman) stated that commercial matter be discussed separately.
Mr Aziz (Indonesia) asked about the load following capability and capacity factor of EC6 compared to CANDU 6.	**Mr Lee (Candu Energy Inc.)** explained the existing CANDU 6 fleet has an average capacity factor of ~90%. For EC6, improvements were made to equipment on the BOP side such that on a year to year basis, the capacity factor is expected to be greater than 94%. Taking into account maintenance outages and midlife outage for retubing the reactor, it is expected the capacity factor for a 60 year lifetime to be around 92%. For the load following, all CANDU 6 historically have the capability to do 2 types of load-following: (i) frequency control load following, where reactor power is allowed to increase and decrease by ±2.5% to respond to grid fluctuations, and (ii) reduce power from 100% full power to 50% net electrical output to the grid in 2 hours, maintain the operation at lower power for days or weeks, and return to full power within 2.5 hours. For EC6 design, there are some additional changes including daily load cycle, i.e. down from 100 to 50% net electrical output to the grid in 2 hours, maintain for 4-10 hours before returning to 100% in a period of 2 hours, day after day for 60 year life.
Mr Kupitz (Chairman) commented on recommendation on INPRO principles but would return after the assessment presentation.	-
Mr Korinny (IAEA) wanted to clarify the redundancy of shutdown systems: one system using mechanical control rods and the other system using liquid poison. For system with mechanical control rod, is it enough to maintain reactor subcritical long enough after shutdown or cooled down to 50°C?	**Mr Lee (Candu Energy Inc.)** noted that it is one of the changes made to EC6. There is a set of mechanical absorber rods to maintain reactor subcritical indefinitely after cool down and after decay of transient poisons.
Mr Popov (Belarus) wanted to confirm whether in several states CANDU has positive power	**Mr Lee (Candu Energy Inc.)** explained CANDU 6 has positive power coefficient of reactivity in the power range from 94% upwards. The value is sufficiently small that it does not create any control

reactivity coefficient and asked measures that are implemented in these states.	concern in terms of stability of control. The contribution during any postulated accidents has been taken into account, e.g. during LOCA because the main cause of the positive power coefficient of reactivity is the positive coolant void reactivity. At normal operating condition near 100% power, there is ~2% steam quality in the coolant at the exit which contributes to positive reactivity. This has been taken into account in the safety analysis, design of control system and there is large margin in the stability diagram for reactor power control as well as impacts on any postulated accidents.
Mr Silva (Thailand) asked what CANDU would suggest to newcomer countries about finding a dry site.	Mr Lee (Candu Energy Inc.) mentioned there are several approaches, one of which taken in several CANDU location was to perform engineering landscape and terrain to achieve the dry site condition. For examples, by adding backfilled soil and rock to build up the base, or in China, mountain was levelled and the rock was used to produce the dry site.
Mr Benkovskyi (Ukraine) asked reason to shut down CANDU reactor in South Korea.	**Mr Lee (Candu Energy Inc.)** indicated that he was not in the position to answer the question. But he indicated retubing of Wolsung-1 was successfully completed and it was restarted in June 2011, briefly shutdown to install passive autocatalytic recombiner and complete the commissioning and operated at full power. However, the licensing operation for Wolsung-1 was expired in November 2011 and the Korean regulator asked KHNP to perform EU stress test to identify improvements to be made to the reactor before renewed operation license can be granted. KHNP has been working toward completing those actions and is expecting to restart the reactor in a near term. **Mr Seo (KEPCO-E&C)** confirmed the current status and added that following Fukushima accident, decision has been delayed. Also, the new Korean president promised all plants that are not safe will not be allowed to operate. So, before restart, safety must be verified, i.e. following the licensing review.
Mr Ibitoye (Nigeria) asked the cost of unit electricity produced for CANDU 6 in comparison to a typical PWR.	**Mr Lee (Candu Energy Inc.)** pointed at the experience with CANDU 6 reactors in Korea, Argentina, Romania and China. Along with building the reactor, there has also been localized manufacturing of fuel, i.e. easy transfer of the technology and training of knowledge to fabricate the fuel.
Mr Grudev (Bulgaria) asked about the sub criticality safety function, and pointed that the presentation did not show the anticipated transient without scram (ATWS). Given the shutdown is improved but without reactor scram, he asked whether it is possible that in case of something happens, the power continues to increase without operator action.	**Mr Lee (Candu Energy Inc.)** said that within the control system, there is setback and stepback to monitor reactor power independent of the shutdown system - unless there is a rapid power transient - to bring power back under control. The setback and stepback function, if transient power occurs, either step the power back to 60% or set it back to 1 %. The design of control system is intended to minimize the use of the shutdown systems. For ATWS, CANDU has analysed the consequences of LOCA assuming SDS1 and SDS2 do not function. The reactor is rendered subcritical by injection of light water using the ECCS.
Ms Yang (China) asked about the purpose of supplemental absorber rods which she believed are not shutdown system, but perhaps for reactivity control redundancy or something else.	**Mr Lee (Candu Energy Inc.)** noted there are 10 supplemental absorber rods that do not exist in CANDU 6. The rods are not automatically inserted in response to trip action, and the operator needs to push a button to drop the rods into the core. It is designed not automatic to give operator the time to make decision whether the shutdown happens because of a real transient or a spurious transient. He noted resetting 32 rods takes time, and resetting 42 rods takes longer time. The 10 rods need not be inserted until 36 hours later giving sufficient time for operator to decide. The supplemental rods have the same drive mechanism as the shut-off rods avoids immediate

	shutdown, but are not connected to the automatic trip.
Mr Reyes (Chile) asked how to assure or confirm that EC6 exceeds 0.5 g.	**Mr Lee (Candu Energy Inc.)** explained that testing was conducted to confirm components having sufficient robustness, margin and capacity to withstand 1.6 times the 0.3 g of DBE. The majority of the components are tested on the shaker table, with the exception for the structure. There are standard civil engineering practices and code that identify the margin to account stress loading from seismic event. The equations are all empirically derived from historical small scale being for containment building back in the late 1960s / early 1970s, subjected to a range of transients until the wall structurally failed. The civil engineers code established multiplying factors to provide the margin. So input loading in the design is adjusted so as to have 1.6 margin to withstand the 0.5 g.
Ms Yang (China) pointed out there are more parameter for reactor protection, and asked whether it is possible to resolve some unplanned shutdown during the operation. She further asked the experiment to test the reliability of I&C system.	**Mr Lee (Candu Energy Inc.)** indicated the designer has looked at the likelihood of unplanned shutdown in forced outage rate, one of which is whether the noise signal received from additional parameters is sufficiently filtered so as not to increase forced outage rate. The design target for the forced outage rate is still less than 1%. The design takes into account adding additional parameters such as stepback parameters to intercept more transient before SDS parameters will be reached. This requires more reliability in the I&C system. Modern I&C are less likely to be analogue and more to be digital with fewer types of failure modes, hence quality of instruments can be improved. However, within the context of safety, new postulated accidents must be considered, such as resulting from internally generated cyber terrorism.
Mr Korinny (IAEA) asked whether Candu Energy Inc. considers an accident with control rod rejection at full power condition within the design basis.	**Mr Lee (Candu Energy Inc.)** indicated it is not the case because the adjuster rods are the only rods in the core during full power operation, they are all in moderator at low pressure and temperature so there is no driving force to drive them out. The only way to pressurize the moderator to drive the adjuster rods out to the core would be failures of pressure tubes and calandria tube. These failures will cause pressurization of moderator and calandria vessel. The overpressure protection disc will rupture very early, ejecting moderator from calandria vessel. The loss of moderator causes large amount of negative reactivity to be inserted. If ECCS initiates, the injection of light water into the moderator through the broken pressure tubes shuts the reactor down and maintains in a subcritical condition regardless whether or not the shutoff rods and poison injection system operate.
Mr Grudev (Bulgaria) asked about the filtering system that is used to manage severe accident, with regard to level of temperature that can be reached on the filtering system because up to 30% of fission products will go to the filtering system and the fact decay heat will be put in a small area. He asked whether Candu considers replacing the filtering systems or melt them. He also wanted clarification on the fact that Candu does not use core catcher to manage corium in the cavity, meaning in-vessel retention is used.	**Mr Lee (Candu Energy Inc.)** pointed out that the filter venting system used to discharge the containment atmosphere, is a liquid system. It can be regenerated and recharged when the liquid gets saturated by using compressed air to force the liquid back through the duct into the containment building and add new charge solution to the filter. It is allowed to boil generating steam, most of which will be trapped in the baffles and condense. It is a well proven system with R&D done in the 1980s. With regard to in-vessel retention, Candu's strategy is first to protect the calandria vessel and retain the corium in vessel. Candu has done extensive heat transfer experiments to examine conditions under which the vessel will fail if cooling is lost. There is a back-up strategy in the case in-vessel retention fails, i.e. the use of refractory materials on the bottom floor which will react with the corium. Furthermore, below the refractory layer there is a steel liner, a special concrete mix and conventional concrete. So multiple steps have been taken.

Mr. Kupitz (Chairman) asked Ms. Boyle (Candu) to explain the recommendation made during her presentation in which she indicated that INPRO should have a new principle; disruption in social life due to accidents.	**Ms. Boyle (Candu Energy Inc.)** explained that the Fukushima event made an impact on the public and disturbed the social aspect, therefore it would be prudent for INPRO methodology to look at the Fukushima lessons learnt and incorporate them. **Mr Phillips (IAEA)** responded that lessons learned from Fukushima are being considered in the current revision of the INPRO manual, being driven directly by safety action plans. In addition, a Dialogue Forum **Mr Drace (IAEA)** added that a Dialogue Forum was held in Seoul in 2012 which has a similar idea to better understand the lessons learnt from Fukushima. **Mr Lee (Candu Energy Inc.)** stated that within the nuclear industry there are a number of discussions on how to treat extreme external events, and what has been emerging from the discussions is that one cannot simply use a probabilistic approach to treat the problem. In North America (certainly in Canada and USA), the term used to consider such extreme external events is to look at 'black swan' events, i.e. events in which regardless of the likelihood of the occurrence and how low it may be, the ultimate consequences should be considered if such an event were to happen. When looking at these events from a sustainability perspective is important to think about these 'black swan' events.

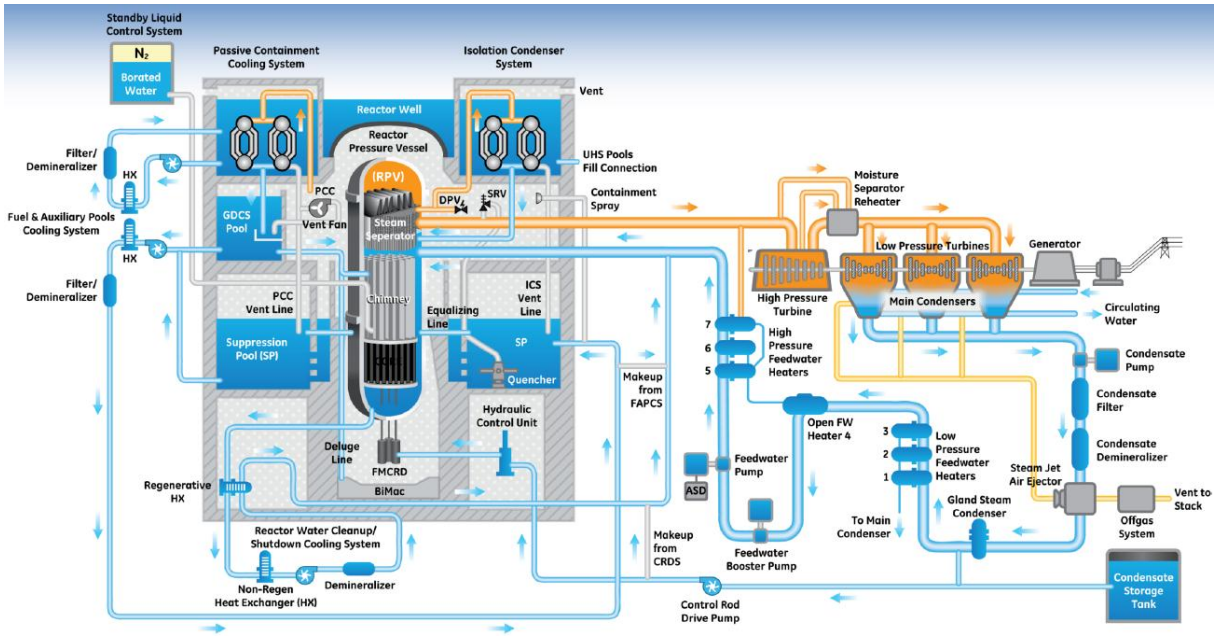
4.6. ESBWR (Economic Simplified Boiling Water Reactor)

Mr James A Beard and Mr Gary L Miller from GE Hitachi Nuclear Energy International presented INPRO assessment in the area of safety of the US Economic Simplified Boiling Water Reactor (ESBWR) design.

4.6.1. Overview of ESBWR Design

Mr Beard (GE Hitachi NE Intl.) explained that the ESBWR builds on the successful operational history of the BWR technology from GE, Hitachi, Toshiba, ABB and others, including the ABWR design. It is designed to incorporate proven state-of-the-art technology, simplification, passive safety features, less dependence on operator actions, and no dependence on safety-related diesel generators. The key EBSWR design is a passive power plant, i.e. primary means to address DBE relies upon passive principles to achieve safety shutdown and maintain safe shutdown. The design still has active system like conventional BWR to provide core cooling, high pressure make up to reactor vessel and removal of decay heat. So the design features diversity and DID. In addition, significantly more margins are available to operators during normal events to response to transient conditions. One of the key drivers of the design mentioned is no reliance on emergency diesel generator as part of the safety case, thus avoiding having to spent cost for maintenance program to ensure availability of the emergency diesel generator.

The ESBWR has a thermal power output of 4,500 MW and an electric output of 1,600 MW. The design eliminates forced recirculation (no recirculation pumps), hence employing natural circulation only. This is an optimization or a trade-off; one downside is no capability for rapid power maneuverability but the reactor can still perform load-following. The design fulfils the US NRC 72 hours passive capability requirement, and in addition it has passive capability on site up to 7 days (instead of the US NRC 4 days requirement). The ESBWR operating cycle length without refuelling is of 12 to 24 months.



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FIGURE 4.6.1. ESBWR overall flowchart

Two changes made following some optimization in the designs. First is the use of open/direct contact feedwater heater which has significance during transient, disruption in the condensate, or potential transient initiation. The size of direct contact can provide for 15 minutes of full power operation even with condensate pump trip. Second is the additional high pressure feedwater heater to adjust power by changing the temperature of feed water going into the reactor vessel without moving the control rods.

Optimization was done to achieve reduced (overnight) capital cost and at the same time higher level safety. There are 11 fewer major systems equivalent to 7% reduction of the systems) and 25% fewer components. With 60-year design life, there is no need to replace or make major refurbishments to nuclear steam supply system (NSSS), containment, or safety related structures. Whenever cost effective, safety-related components are qualified for 60 year lifetime. Some overhaul/replacement work like battery banks can be performed online (batteries are significant on passive power plant, providing 72 hour electricity, requiring significant maintenance and testing – which can be performed online). Design of I&C based on non-proprietary standards with large margins is adopted allowing for inevitable obsolescence and subsequent increases in performance.

TABLE 4.6.1. Optimised parameters for ESBWR

Parameter	**BWR/4-Mark I (Browns Ferry 3)**	**BWR/6-Mark III (Grand Gulf 1)**	**ABWR**	**ESBWR**
Power (MWt/MWe)	3293/1098	3900/1360	3926/1350	4500/1590
Vessel height/diameter (m)	21.9/6.4	21.8/6.4	21.1/7.1	27.6/7.1
Fuel Bundles (number)	764	800	872	1132
Active Fuel height (m)	3.7	3.7	3.7	3.0
Power density (kW/l)	50	54.2	51	54
Recirculation pumps	2 (large)	2 (large)	10	zero
Number of CRDs / type	185/Locked Piston	193/Locked Piston	205/Fine Motion	269/Fine Motion
Safety system pumps	9	9	18	zero
Safety Diesel Generator	2	3	3	zero
Core damage frequency/yr	1E-5	1E-6	1E-7	1E-8

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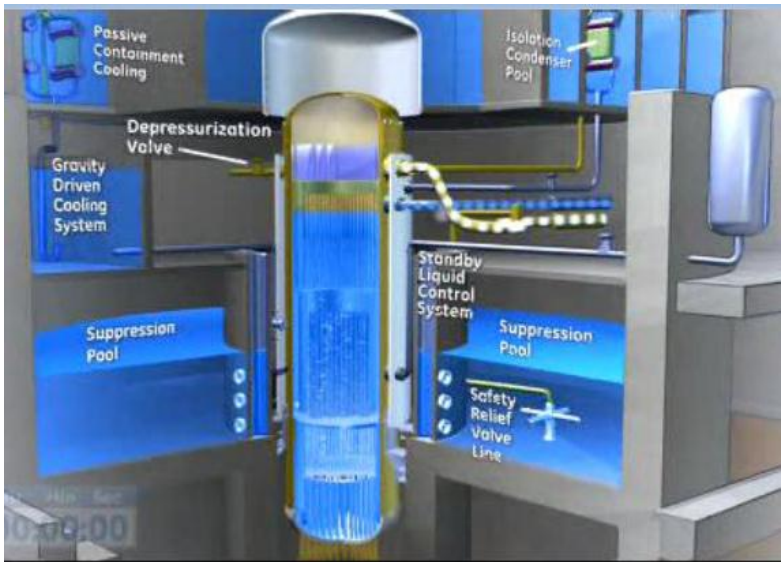
Assessment of EBSWR using the INPRO methodology uses BWR6 as the reference plant. It was noted BWR6 uses pressure suppression containment (Mark III containment) with large wetwell airspace so lesser need of high design pressure. Comparison of their key safety functions are presented below.

TABLE 4.6.2. Comparison of key safety functions of ESBWR versus BWR6

Key safety functions	BWR6	ESBWR
Coolant injection	- High Pressure Core Spray: 1 Train - Motor Driven Pump with dedicated Diesel Generator - Reactor Core Isolation Cooling: 1 Train -Turbine Driven Pump - Control Rod Drive Bypass Flow - Automatic Depressurization System: Safety/Relief Valves - Low Pressure Core Spray: 1 Train - Motor Driven Pump - Low Pressure Coolant Injection: 3 Trains - Motor Driven Pumps - 2 Diesel Driven Fire Water Pumps	- Gravity-Driven Cooling System: 4 Trains No Pumps - Automatic Depressurization System: Safety/Relief Valves and Squib-actuated Depressurization Valves - High Pressure Control Rod Drive System Flow: 1 Train Motor Driven Pump - Low Pressure Injection: 2 Trains Motor Driven Pumps - Isolation Condensers: 4 Trains Natural Circulation, Passive actuation - 1 Diesel Driven Fire Water Pump
Containment cooling	- 2 Trains – Motor Driven Pumps with Heat Exchangers - Wetwell venting	- Passive Containment Cooling Heat Exchangers: 6 Trains Natural Circulation - Wetwell venting
Reactivity control	- Hydraulic Control Drive System - Standby Boron Injection System with Charging Pumps	- Hydraulic control rod drive with mechanical drive back-up - Passive standby boron injection system
Key supporting systems	- AC Power – 2 Emergency Diesel Generators + 1 HPCS Emergency Diesel Generator - DC Power – 12 hour batteries - Service Water for component cooling and decay heat removal	- Cooling Pools for Isolation Condenser and Passive Containment Cooling - DC power – 72 hour batteries - Diesel Generator and Service Water available for alternative cooling paths, but not required.
Loss of coolant accident (LOCA)	- High Pressure Core Spray, Reactor Core Isolation Cooling, Low Pressure Core Spray, Low Pressure Coolant Injection, Firewater - Core becomes partially uncovered during design basis LOCA	- Gravity-driven Cooling System, High Pressure Control Rod Drive Injection, Low Pressure Injection, Firewater - Core remains covered during design basis LOCA
Loss of heat sink (LOHS)	- Reactor Shutdown - Containment Isolation - Safety / Relief Valves discharge to Containment Suppression Pool - Cope for 12 hours using non-AC injection until heat sink is restored	- Reactor Shutdown - Containment Isolation - Isolation Condensers actuates, removing decay heat to atmosphere - Restore water to Cooling Pools after 72 hours

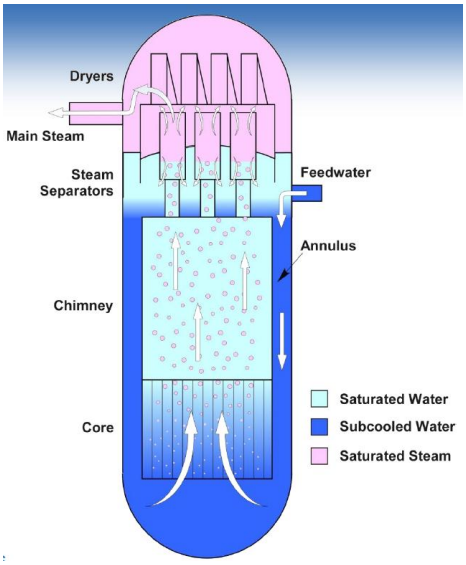
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Mr Beard had the opinion that one should look failure mechanism beyond Fukushima that could lead to disruption and disable capability to react. He showed response of EBSWR to LOCA, and noted the time elapse from initiation of event to occurrence of certain activity (*See also figure in CR1.3.2*). It was noted that the water level of the reactor vessel never drops below the top of the fuel, at least 1 m over top of the fuel, and in vast majority of DBAs it is around 3-4 m.



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FIGURE 4.6.2. ESBWR LOCA response



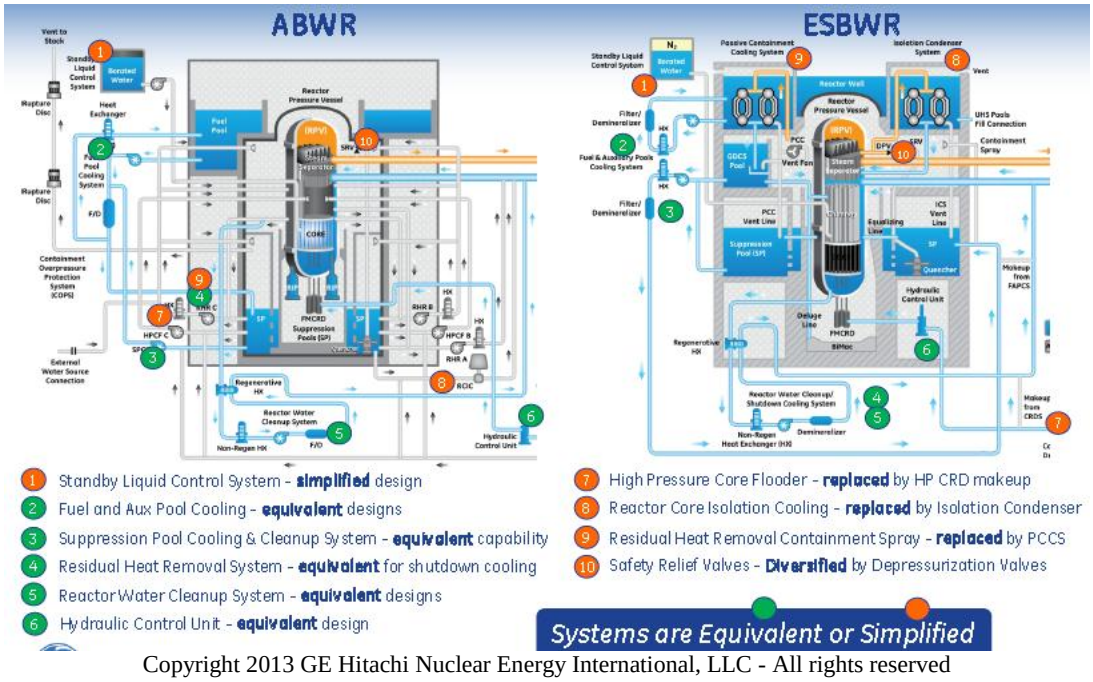
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FIGURE 4.6.3. Natural circulation of ESBWR

4.6.2. Assessment of INPRO Criteria

4.6.2.1. EP 1.1.1.2 Simplicity of Design

Mr Beard (GE Hitachi NE Intl.) discussed one of the options to simplify the reactor design, i.e. reduction of the number of components of the cooling system. The approaches taken for the ESBWR are elimination of systems or combining system functions, and simplification of systems. The Reactor Recirculation System including support systems (e.g. pumps, heat exchangers, powers supply, adjustable speed drive) and the Relief Valve mode of Safety Relief Valves (SRV) are eliminated. Elimination of relief valve is due the availability large volume of subcooled water and vapour in the vessel that during pressure transient no relief valve is required. Passive safety requires a lot of water to transfer heat to the ultimate heat sink, requiring continuous cooling and cleaning. The ESBWR combines system functions for Reactor Water Cleanup and Shutdown Cooling, and also Water Pool Cooling and Cleanup system (Spent Fuel Pool, Suppression Pool, Reactor Cavity and Equipment Pool). The Standby Liquid Control has been simplified, i.e. active system converted to passive system.



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FIGURE 4.6.4. Evolution of the ESBWR nuclear island showing simplification with respect to the ABWR

The ESBWR adopts passive natural circulation by increasing the differential water level by ~8.2 m (27 ft), by increasing the length of pressure vessel by 6.5 m. The fuel assemblies are also shortened (3 m for ESBWR vs 3.7 m for ABWR) to overcome the dominant two-phase flow restriction (in natural circulation) in the upper part of the fuel assemblies. Simplification in the ESBWR removes many risks of forced circulating plants, e.g. pump failure. Components, e.g. pumps, motors, controls and heat exchangers, are significantly reduced. Altogether there are 11 fewer major systems and 25% fewer components in the ESBWR compared to that in the ABWR design.

In conclusion, simplicity of the system with respect to reference plant is achieved.

4.6.2.2. EP 1.1.1.5 Redundancy of Systems

Mr Beard (GE Hitachi NE Intl.) discussed design features focusing on elimination or minimization of transients. In the turbine island, the design adopts triplicated fault tolerant digital I&C (turbine control via steam by pass and pressure control, and 3-element feed water level control) and 100% load rejection and transition to island mode without scram. There are redundancies in both condensate and feed water systems (4×35% for pumps) and within filters and demineralizers within the condensate system (N+1). The feedwater heater #4 (direct contact) has a capability of ~15 minute volume. Standby condensate and feedwater pumps auto start if needed.

In the nuclear island, there are isolation condensor system (ICS), high pressure CRD makeup and RWCU shutdown cooling. It was noted that the ICS although credited and qualified as passive system, have components that need to be repositioned to put the system in operation (according to the regulator, passivity constitutes reliance on stored energy although a change of state is expected in a minimum set of components). The use of ICS means no loss of coolant on isolation event (no opening of SRVs). The ICS comprises 4×35% heat exchangers with fail-safe initiation through the parallel control valves operable by 3 of 4 safety related divisions (Q-DCIS and DPS). Any 3 out of the 4 heat exchangers in operation will remove sufficient decay heat to ensure safe shutdown. Mr Beard explained there are 4 safety related divisions of I&C. Because the design relies on solenoid or squilb valves, multiple signals can be brought in into a configuration such that a change in state in any one of the 4 valves will give the desired action. When ranging control configuration for the valves and the ICS, signals come from 3 of the 4 different I&C systems and they are distributed out. Because common cause failure is a significant consideration, there is also diverse protection system. So the 4 different I&C systems can cause a valve to actuate and perform the desired function, although mechanically it is only N+1 because there is no requirement to postulate passive failure of the heart exchangers, rather only active failure (parallel control valve and ability to control with multiple divisions of I&C), so it can be shown that ESBWR is a N+2 plant for that particular function.

The high pressure CRD makeup consists of two trains of high pressure pumps (1.96 m^3/min per train) and each is backed by on-site standby diesel generators. Active system is available for transient; the pump should deliver enough water to RPV to handle a small break LOCA. The RCWU shutdown cooling is high pressure decay heat removal and has 100% capacity 30 minutes after shutdown. The major improvement is that shutdown decay heat prior to the ESBWR was limited to 1 MPa pressure.

In conclusion, the ESBWR has achieved the desired characteristics.

4.6.2.3. EP 1.1.2.2 High Quality of Operation

Mr Beard (GE Hitachi NE Intl.) referred to features that show enhancement to the reference plant. In the nuclear island, the ESBWR has combined Reactor Water Cleanup and Cooling Combined together into a single system; the low pressure RHR is not utilized and source of Inter-System LOCA (ISLOCA) is eliminated. Utilizing natural circulation eliminates the recirculation system (external recirculation piping and pumps). Although with ABWR, unstable operation can be avoid for a single up to 3 pump trips, the ESBWR is further claimed to take one step further, i.e. no pump trip hence it is not possible for the core to go into the region of ABWR operation and cause the core to go neutronically unstable. The favourable water to steam volume ratio eliminates opening of SRVs for transients and DBAs.

The ESBWR greatly improves the natural circulation with passive safety/natural circulation. The differential water level is increased by approximately 8.2 m (27 ft) to increase driving heat to natural circulation flow. The ESBWR can operate under natural circulation condition up to ~40% of rated thermal power.

Simplification removes many risks of forced circulation plants e.g. pump failure. Significant reduction is done for components, e.g. pumps, motors, controls, and heat exchangers.

On MSIV closure transient, the design eliminates potential transient conditions in the operating plants, e.g. in the event SRVs are open, stuck open or not fully receded, forcing unnecessary shutdown.

In conclusion, it is shown that the design features in the ESBWR has improved operability of the plant in comparison to the reference plant.

4.6.2.4. CR1.2.2 Grace Period (Normal Operation)

Mr Beard (GE Hitachi NE Intl.) indicated that focus has been given to the BOP side because the PSA shows that the transient starting of BOP tends to be developed into cascaded DBAs. However, in the nuclear island, the support systems are arranged in 2×100% train configuration (active equipment) to remove decay heat from the spent fuel pool, RWCS, and during shutdown cooling. The system in standby is available to auto-start in the event that the system running is disrupted. Each train has on-site AC power sources. This applies to electrical power, cooling (service water), component cooling water, compressed air, chilled water, heating ventilation and air conditioning (HVAC), and other support functions. The ESBWR features 100% load rejection and transition to island mode without scram (this is optional for ABWR) following experience of SBO in North American state.

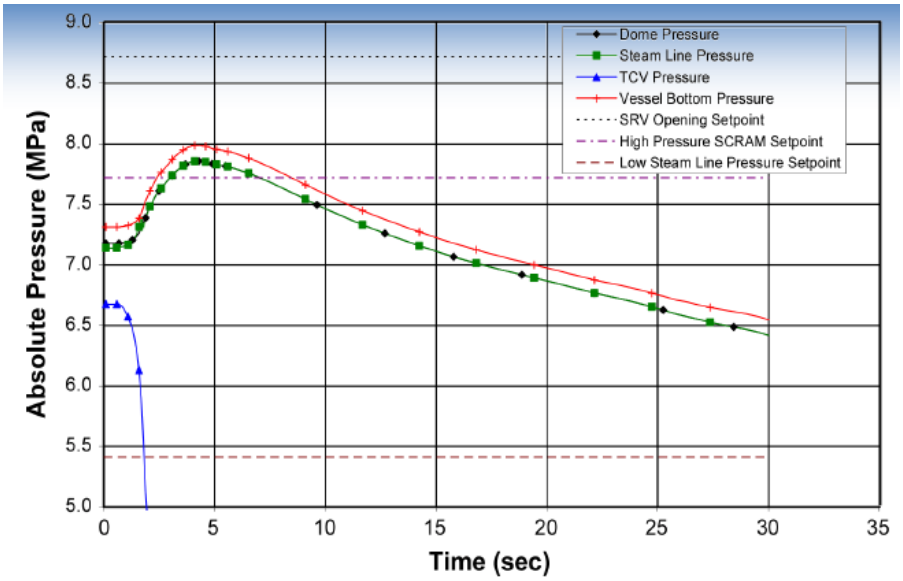
In the turbine island, there is triplicated fault tolerant digital I&C (turbine control via steam by pass and pressure control, and 3-element feed water level control). Condensate and feed water is 4×35% for pumps, and N+1 for filters and demineralizers. Feedwater heater #4 (direct contact) has ~15 minute volume. Standby condensate and feedwater pumps auto start if needed. In the nuclear island, for rapid reactor power reduction the pre-designated control rods are hydraulically scrammed. In addition, other pre-designated control rods are inserted normally. Both are sufficient to reduce the power to 50%. Reactor power is decreased to match sustained steam bypass capacity of the main condenser. The operator can then discuss to determine whether it is a short-term or long-term situation.

In conclusion, operation capability for this criterion is shown.

4.6.2.5. CR1.2.3 Inertia

Mr Beard (GE Hitachi NE Intl.) pointed at the NSSS and turbine island features to address the criterion. In the NSSS, the key feature is that pressurization transients such as main steam isolation valve (MSIV) closure or turbine trip without bypass do not lift the safety relief valves (SRVs). The reason is large volume of both sub-cooled water and vapor in the RPV with favourable ratio of water to steam. The turbine island has 100% steam bypass capability and generator load rejection without turbine trip or scram (island mode operation).

He referred to the MSIV closure transient below. Within a few seconds, the pressure in BWR6 or ABWR will exceed the pressure relief valve setpoint. With ESBWR, there is just minor increase in pressure followed by significant drop-off in pressure due to all cold water being swapped down under the core (and not due to the isolation condenser that is still not in operation within the first 30 seconds or so of transient).



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FIGURE 4.6.5. MSIV closure transient: limiting pressure transient – no SRVs open

In conclusion, natural circulation in the ESBWR provides significant amount inertia to enable operator to correctly analyse the response to the event.

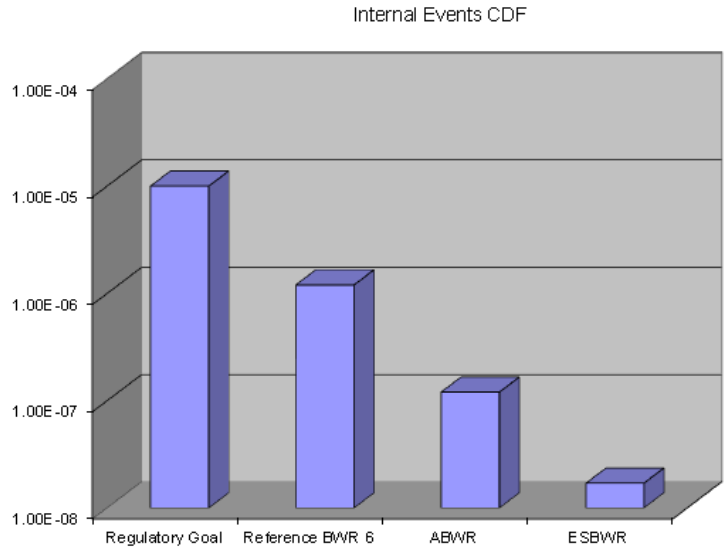
4.6.2.6. CR1.3.1 Frequency of DBA

Mr Miller (GE Hitachi NE Intl.) pointed out the use of PSA directly in the design process as a feedback mechanism, looking at safety and operational features to provide insight and analysis as the design progressed. The development and use of a PSA in the ESBWR design process allowed the designers to assess design options that reduced the risks of core damage and radiological release. Extensive use of operating experience and advanced light water reactor design principles (e.g. passive safety) has significantly reduced the plant risks. One example is the comparison of CDF. The original regulatory goal was even achieved by the reference plant by a factor of 10, as part of the evolutionary process. He also noted the BWR6 has 2 safety-related divisions at the time and the ABWR has 3, while the ESBW already has 4 safety-related divisions as well as passive safety features. So when quantifying CDF and reliability of safety system, the ESBWR is considered as very unlikely to fail.

Although the ESBWR has a very low CDF, the importance is not the value but the relative insights. For example, the reference plant BWR6 has high contribution to SBO due to the use of active safety systems which require pumps and forced recirculation for safety injection cooling system in the decay heat removal. In the current fleet of light water reactors PRA and PSA, the dominant contributor to CDF is the SBO. As noted in the table, there is slight improvement of the CDF for the ABWR given more safety-related divisions, use of reactor core isolation cooling that is independent of AC power, and use of residual steam from reactor to drive the turbine to drive the pump to supply cooling water. For the ESBWR, the calculated contribution to SBO is very small, even no actions are required for some time in the event of SBO.

The ATWS is another big contributor to CDF. Mr Miller noted the CDF due to ATWS for the ESBWR is higher than that for the ABWR. In developing the ESBWR, the designer used the realistic and some conservative approaches for a purpose and removed actions that require lots of procedural steps. In developing the PRA model, the role of operator is downplayed for dominant accident sequences that could be mitigated by more operator's action, by means of additional design features in order to increase the reliability of the design and less reliance on operator.

Transient condition is a small contributor to CDF. The ESBWR looks at more prevention and mitigation, increasing BOP capability, forward heating of feedwater safety system, and additional capacity of condensate and feedwater pumps to accommodate transient without having to trip the reactor. On LOCA, the ESBWR design has eliminated recirculation piping, resulting in progressive reduction of the CDF magnitude.



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FIGURE 4.6.6. Comparison of internal core damage frequencies (CDF)

TABLE 4.6.3. Contributions to core damage frequency. Internal events accident sequence frequency (/a)

	Reference BWR	ABWR	ESBWR
Station Blackout	3.9 E-06	1.1 E-07	<1 E-09
ATWS	1.1 E-07	3 E-10	3 E-09
Transient Condition	1.9 E-08	4.5 E-08	6 E-9
LOCA	<1 E-08	7 E-09	7 E-09
Safety Architecture	2 Divisions	3 Divisions	4 Divisions + Passive

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4.6.2.7. CR1.3.2 Grace Period (Design Basis Accident)

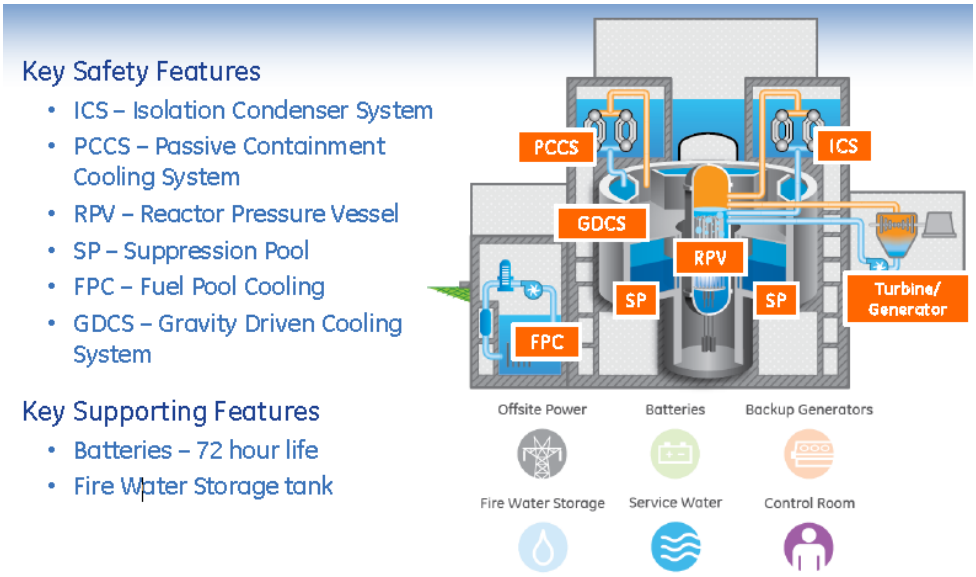
Mr Beard (GE Hitachi NE Intl.) noted that depressurization and flowing water from the gravity driven cooling system into the RPV do not start until about 8 minutes into the event. This shows very extended time for the need of additional outside water into the RPV, relative to that in other designs. The fault tolerant digital I&C is designed to execute all actions required for the first 72 hours of transients and DBAs, without operator actions. The large sub-cooled water volume in the RPV dampens immediate demands, hence providing confidence and time for the operators to assess the events.

In light Fukushima, the vendor looks back at some inherent capabilities of the ESBWR. It was shown that the 72 hour water inventory is actually sufficient for 7 days. In the 72 hour case, the water level is brought down to up about halfway down the heat exchanger tubes. Assuming continuous removal of water from the heat exchanger, the water level will continue to drop. There will be a slight increase in pressure of the RPV because a slightly larger ΔT is needed given the less tube surface immersed in water. All analyses show there is no need to replenish the water to the Isolation Condenser Passive Cooling Containment System (ICPCCS) pools for the next 7 days. It was noted the function does not rely on electricity to accomplish. The safety analysis in the case of loss of both AC and DC power, the ESBWR includes designs elements such that During beyond DBA and DEC, the fail-safe actuation of scram, MSIV closure and ICS (initiate at fail-safe mode) occurs without I&C or control power – allowing the plant to maintain safe shutdown at least for 7 days.

There is automated response to DEC events, whether ATWS or SBO. SBO is handled by isolation system condensers (ICS) with minimal heat up of the primary containment. For Anticipated Transient Without Scram (ATWS), all actions are automated, with diverse I&C system controls functions, backup actuation of hydraulic scram, alternate insertion of control rods electrically, and time delayed actuation of SLC which allows for electrical insertion to complete. For the LUHS, disruption of ultimate heat sink is very unlikely. However, a mechanism is yet to be identified because the design currently relies on water stored within the reactor building to remove heat from the ICPCCS.

ESBWR response to extended SBO: Full power

The assumptions used include automatic system responses, operator recovery actions not credited for 7 days, and auxiliary equipment not credited for 7 days. Responses from the first second to the 7 days SBO are presented below. For long-term recovery, when AC power and service water are restored, normal shutdown cooling is established.

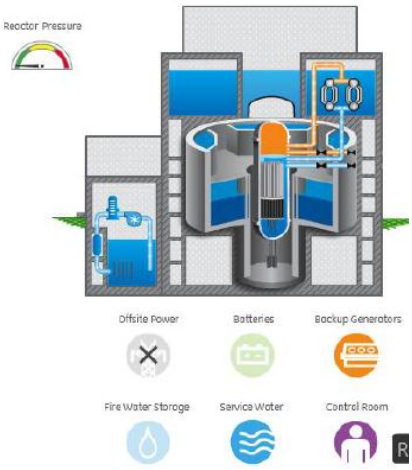


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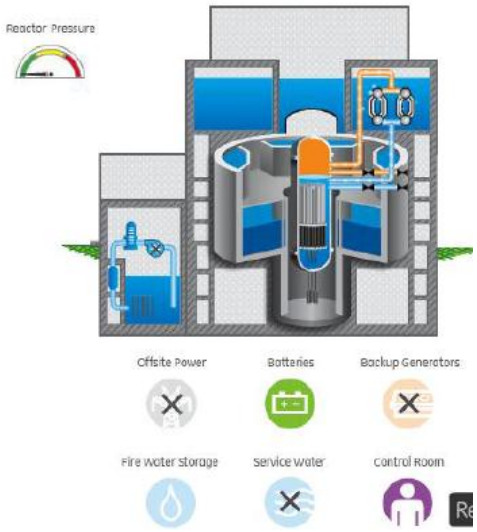
FIGURE 4.6.7. Response to extended station blackout (SBO)



(i) Normal operations:
Steam from RPV to Turbine/ Generator; Condensed steam returns to RPV; Spent fuel cooling



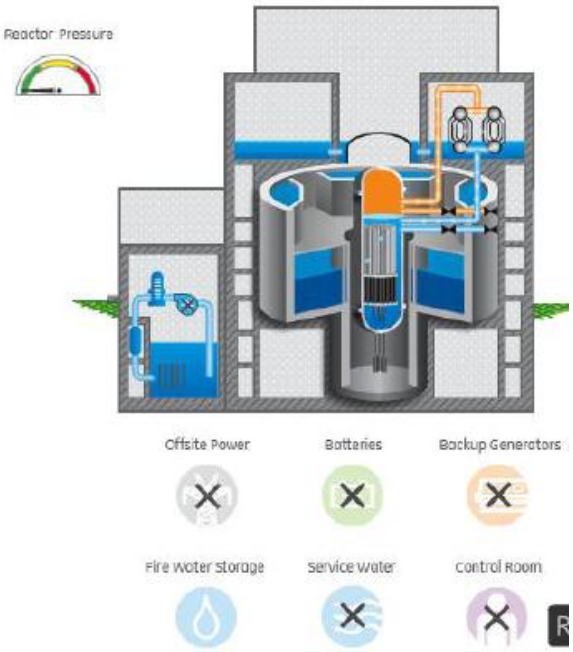
(ii) Earthquake (00:01):
Loss of Offsite Power; Turbine/Generator Trip; Reactor Scrams; Containment Valves Close; ICS automatically runs; Diesel Generators start; FPC pumps stop; Reactor cool-down is in progress.



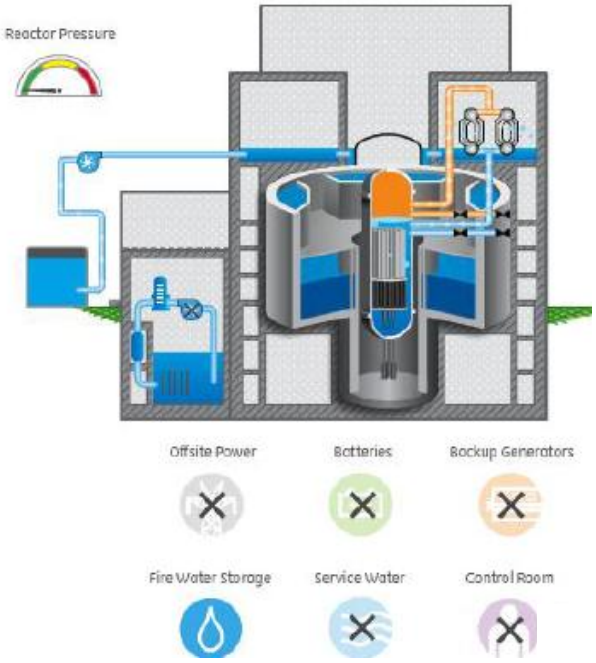
(iii) Beyond Design Basis Flooding (01:00):
- Service Water and Diesel Generators lost
- FPC pump unable to run
- ICS continues to cool Reactor
- Decay heat is removed from Reactor to IC Pools
- Control Room panels and lighting continue to function: Non-essential displays lost after 2 hours



(iv) ICS Pool heats up and boils, reducing level (7 days):
- Valves automatically open providing water from Equipment Pool and Outer Expansion Pools to ICS Pools



(v) ICS Pool heats up and boils, reducing level (7 days):
- Valves automatically open providing water from Equipment Pool and Outer Expansion Pools to ICS Pools



(vi) Long-term Response (7 days):
- ICS Pools boil; can be refilled using:
- Diesel-driven fire pumps with fire water storage tank water or
- Portable pumps/fire trucks through dedicated firewater stand pipes
- ICS continues to cool Reactor
- Decay heat continues to be removed to ICS Pools
- Spent Fuel Pool can be refilled using portable diesel-driven pumps or fire trucks

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FIGURE 4.6.8. ESBWR response to extended station blackout (SBO)

In conclusion, the ESBWR satisfies the large margin in the grace period.

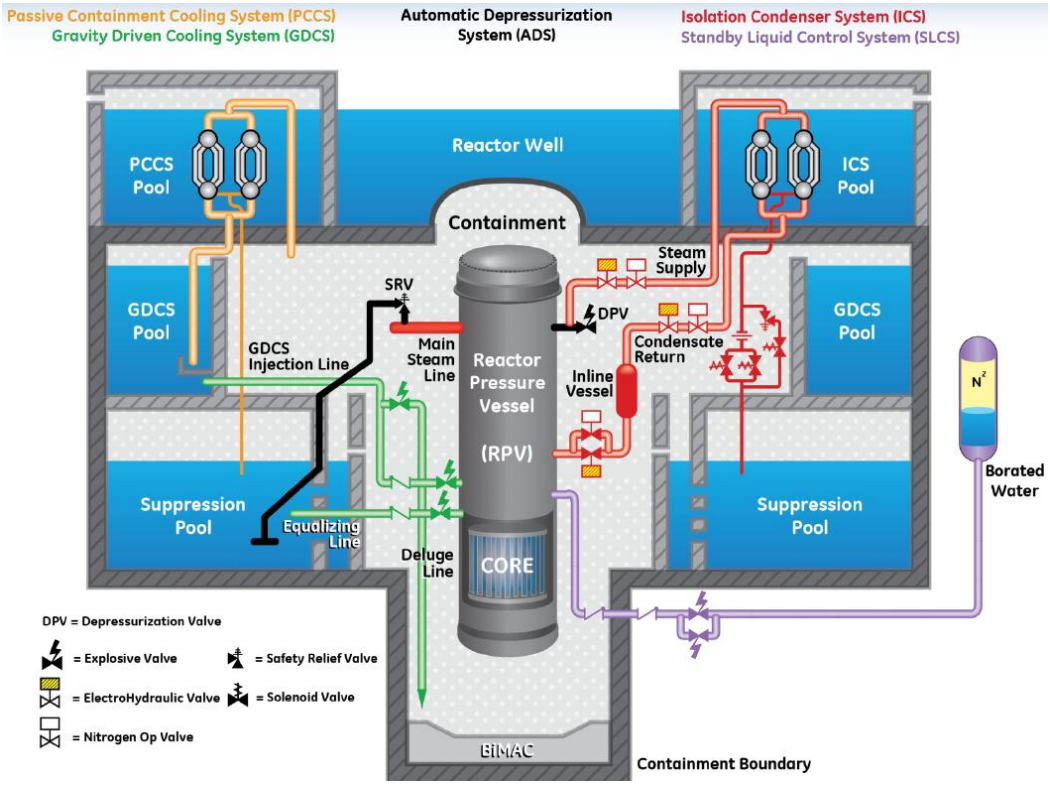
4.6.2.8. CR 1.3.3 Safety Features

Mr Beard (GE Hitachi NE Intl.) delivered the assessment on enhanced reliability of engineered safety features of the ESBWR. He particularly discussed the ICS which are available in several operating BWRs including in Fukushima. The ISC on Unit 1 was almost too good. Immediately after the earthquake, the ISC came into operation and the plant was cooling down too rapidly and the operator made a decision to limit the cool down within the technical specification limit and secure the operation of the ICS because the AC power was available. The operators did not anticipate of the tsunami that followed, which took the away the AC power and the operator had difficulty to re-establish the ISC. In the ESBWR, the size of ISC is decreased to handle decay heat immediately after shut down and not to provide rapid cooling. It was noted that in relation to the size of DC battery, once the system is actuated, the system does not actually require any more electrical power to operate. The valves do not require constant energy or pressure to maintain the position.

The Passive Containment Cooling System (PCCS) is entirely passive. The system actually varies slightly with the steam condition in primary containment. Because of the 9 condensable gases that will start to accumulate in the PCCS and degrade heat removal capacity. As the pressure rises lightly, purge cycle is on where non-condensable gases are pushed out. This restores full heat removal capacity.

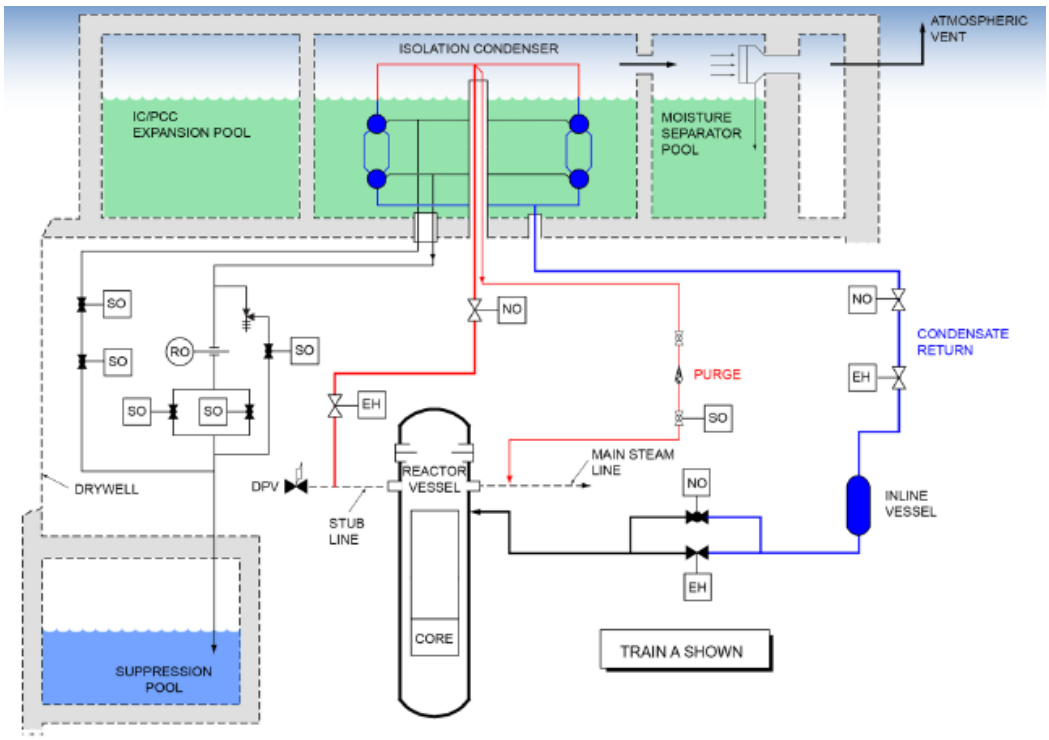
In the Gravity Driven Cooling System (GDCS), each train is controllable by 3 of 4 Q-DCIS and DPS. The squilb circuits are continuously monitored for continuity and the squilb circuit contacts can be verified without firing. The valves do not require constant energy or pressure to maintain position.

For the Automatic Depressurization System (ADS), there are two stages: first stage relies upon external actuation of SRV followed by depressurization valves using small explosive charge to actuate. 10 of 18 SRVs are controllable by 3 of 4 Q-DCIS and DPS and are only needed for a short period. For the depressurization valves (DPVs), the squilb circuits and valves work in the same way like those in the GDCS.



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FIGURE 4.6.9. ESBWR passive safety systems



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FIGURE 4.6.10. Isolation condenser system (ISC) – standby

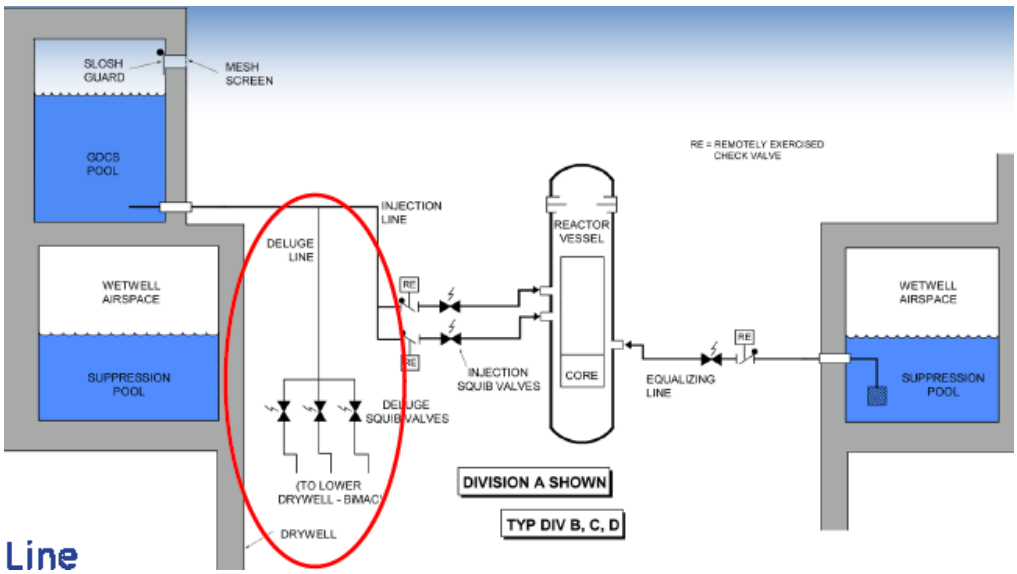
Squilb valve cannot be tested. So to verify the circuit is always available, so there are 3 contacts in series. Each has a resistor and there is always voltage applied. Because of resistor, there is not enough current passing through the squilb device to actually initiate it. Even by closing individual contact within the circuit, there is not enough current. But by measuring the change in current, the availability of the circuit across the contact can be verified when needed. To prove operability, each time shutdown for plant maintenance outage, remove certain percentage of the explosive assembly, and tested.

4.6.2.9. CR1.4.1 Major release into containment

Mr Miller (GE Hitachi NE Intl.) indicated that the calculated frequency of a highly degraded core is well below the recommended value by IAEA of 10^{-5} per unit-year. He referred to the internal CDF figure (*See CR1.3.1*) and reminded again the importance of the progress of the design, and not the absolute value of the CDF.

4.6.2.10. CR1.4.2 Processes

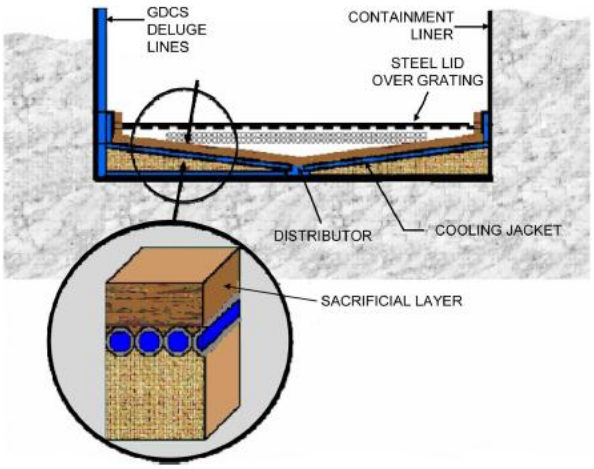
Mr Miller (GE Hitachi NE Intl.) noted that after the core has melted through the vessel, the number one of objective of the plant operator is to protect the containment as the barrier to the environment. The EBSWR has a Deluge Line which opens a flow path between the GDCS pool and floods the lower drywell & BiMAC. He discussed the BiMAC device under the core and start with the water source for BiMAC to start for cooling. Above BiMAC device, there is an array of simple thermocouples to detect very high temperature which is an indicative of core melt through. The signal will go to independent control platform that would initiate squilb valves, i.e. the Deluge squib valves open on high temperature set point in the lower drywell. The squilb valves would allow water from gravity driven cooling system to be diverted into the BiMAC pipes. The deluge line is designed to cool the core in the event it melts through the vessel. DBA analyses show that the core will always remain covered, so no melting will occur. Therefore, the deluge function is a DID feature (BDBA).



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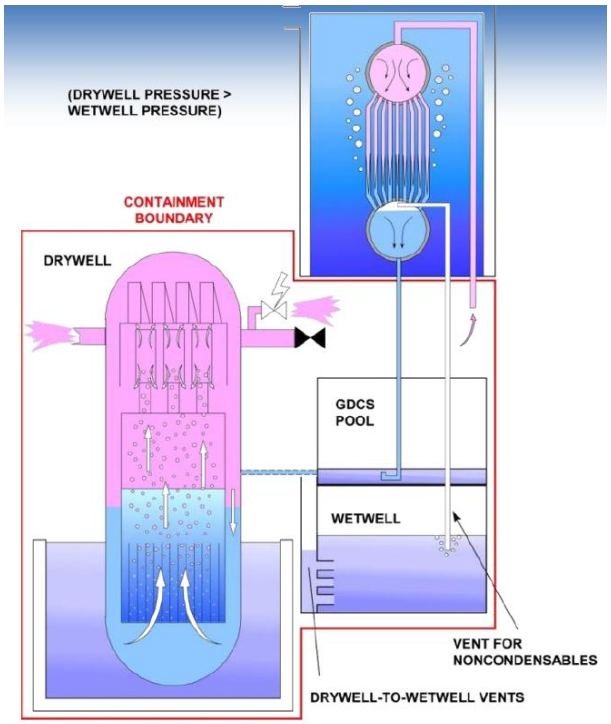
FIGURE 4.6.11. GDCS configuration

The BiMAC is designed to quench any high temperature corium and mitigate core-concrete reactions, with dedicated I&C and electrical supply. The heat is passively transferred out of the containment utilizing PCCS. Once an equilibrium is established, the PCCS only works as hard as it needs to in removing decay heat. Inside heat exchangers there is a moving boundary established between fluid rich in steam and fluid rich in non-condensables.



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FIGURE 4.6.12. Basemat internal Melt Arrest and Coolability (BiMAC)



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FIGURE 4.6.13. PCCS and GDCS operation

The ESBWR also has some passive features for cleaning the atmosphere of the containment following a severe accident. All steam in the drywell and LOCA will be directed to the passive suppression pool and scrubbed. The passive suppression pool scrubbing has a decontamination factor (DF) of 10 for gases and

100+ for particulates. The manual containment venting is done through the suppression pool airspace. Another passive feature is that when radioactive steam from the suppression pool will go to the PCCS heat exchangers and condenses, there could be some plate out of particulates to provide sufficient decontamination (and not enough to cause flow blockage). The ESBWR also has containment spray.

4.6.2.11. CR1.4.3 Accident management

Mr Miller (GE Hitachi NE Intl.) stressed the importance to have accident measures in advance in the beginning of the designs. The nuclear industry has learned that planning of severe accidents would have mitigated some of the accidents that took place in the past. The GE Hitachi NE Intl. used the PSA and Human Reliability Analysis to confirm that severe accident guidance effectively addresses preventing core damage, recovering from core damage, maintaining containment integrity, and minimizing radionuclide releases. An example was provided on risk-informed emergency operating procedures. The human actions, that would contribute greater than or equal to 10% of the NRC Safety Goals (e.g. CDF of 1E-4/year, LRF of 1E-6/year) if not completed successfully, are identified as key operator actions having the highest risk importance. The designer informed the human factors engineering group that those functions and operator actions must have sufficient procedures and training developed as well as control room indications to know exactly that for specific actions operators have all the information required to diagnose and take actions, keeping in mind the sequences are still of very low frequency.

4.6.2.12. CR1.5.1 Major release to environment

Mr Miller (GE Hitachi NE Intl.) noted that for severe accidents there are several common phenomena and the goal is to make the frequency low (AL of less than 10^{-6} per unit-year) or practically excluded by design.

- *Combustible gases* control. Problem with excess of H_2 that can cause containment over pressurization not only from explosion but also overpressure of H_2 that can result in containment breach. In the EBSWR, the containment is inerted with nitrogen during normal operations. The calculation shows that the rate of hydrogen and oxygen production by radiolysis will not yield a detonable mixture in the first 24 hours of a severe accident. In addition, Passive Autocatalytic Recombiners (PARs) are also included in the design.

- *Direct Containment Heating (DCH) (occur during vessel failure at high pressure, impacting on the containment structure integrity).* 2 of 4 Isolation Condensers must fail (passive actuation) and depressurization valves and safety relief valves must also fail. The likelihood of DCH is thus very low for the ESBWR. Reinforced concrete containment is not structurally challenged by DCH.

- *Line break outside containment - LRF ~1E-10.* Design requirements have eliminated potential high-to-low pressure piping in the Reactor Coolant Pressure Boundary (RCPB). Redundant isolation valve are added to RWCU/SDC (shutdown cooling) line based on probabilistic risk assessment (PRA) (although not imposed by the US requirement) .

- *Isolation failures - LRF ~5R-9.* All RCCB isolation valves have redundancy. Distributed digital information and control systems provide more reliable operating system for actuation signals.

- *Core - Concrete Interaction (CCI) – LRF <1E-11.* The GDCS Deluge subsystem provides flooding of lower drywell following vessel melt-through to minimize CCI. Pressure from non-condensable gas generation does not exceed containment ultimate pressure for most likely accident sequences.

- *Basemat Penetration – LRF <1E-11.* BiMAC device cools the corium, preventing basemat penetration.

- *Ex-Vessel Steam Explosion – LRF ~1E-9.* This phenomenon could occur if Lower Drywell contained a lot of water at a depth greater than 1.5 m prior to vessel melt-through, i.e. hot corium/core debris dropping into steam causing very violent explosion. This accumulation is prevented by design features, such as elimination of large breaks below the core (e.g., recirculation piping which was major contributor to LB LOCA and high drywell water level).

4.6.2.13. CR1.6.1 Independence of DID levels

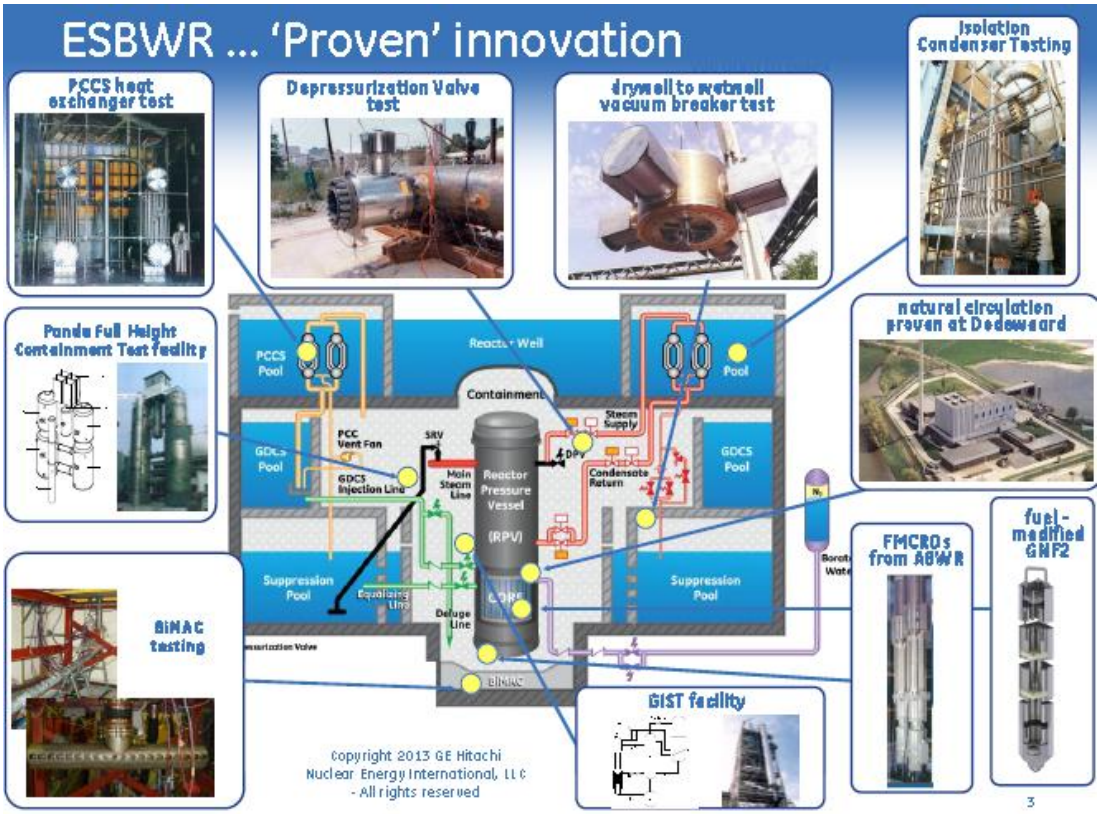
Mr Miller (GE Hitachi NE Intl.) mentioned the ESBWR design incorporates independence between accident mitigation and severe accident management. He highlighted true independence, that is, when a system is

credited for mitigating DBA and the same system is also credited for severe accident, then one should ask if the system fails during the DBA then how it cannot work in severe accident. Between DID Level 3 (design basis) and Level 4 (severe plant conditions), the designer has incorporated core catcher called BiMAC device. (*See BiMAC in CR1.4.2*) PSA severe accident insights were used to develop BiMAC is designed to passively quench high temperature corium and mitigate core-concrete reactions. It is located on the bottom of drywell floor, buried under refractory materials. It is a series of pipes, side by side, covering the whole drywell floor area at an inclination. There is a distribution header in the middle, inclined pipes on the side. Given core melt / severe accident, it allows cooling with water on top and below, thus creating natural circulation to cool the corium. The BiMAC is controlled by an independent logic platform.

4.6.2.14. CR2.1.4 Confidence in innovation

Mr Miller (GE Hitachi NE Intl.) discussed several testings and RD&D programs to validate the innovative components and approaches of the ESBWR design. He noted the major new features of the ESBWR which are passive features to replace conventional pumps and valves. Proven innovation of the design is presented in the figure below. As new features, both GDCS and PCCS received extensive testing.

- The PCCS heat exchanger test. A full scale test was done in Italy, e.g. on non-condensable gas testing to make sure non-condensable is properly vented. The main purpose is validate the codes to predict performance and characteristics of conditions thus providing high confidence. The same unit is used for for ICS heat exchanger heating (because of lower power, both halves of full-size unit could be tested).
- PCCS heat exchanger integral testing was performed in Japan and Switzerland (Paul Scherrer Institute). It is a full scale in vertical direction but with fewer tubes.
- The DPV test is performed at Wyle Laboratories. The DPV is a squilb actuated valve which will open to allow depressurization. This is a common squilb valve but of bigger size, hence testing must be done to ensure its performance under the expected accident condition.
- Drywell to wetwell vacuum breaker test. Vacuum breaker is used to prevent large pressure differential across the suppression pool and drywell that can cause backflow.
- The Isolation Condenser for ESBWR which is same in principle with those used in BWR2 and BWR3 (Dresden2/3, Oyster Creek, Nine Mile 1, tube and shell configuration), but with different geometry. The older BWRs use horizontal isolation condenser (shells and tubes) while the ESBWR uses the vertical one. A a one full-scale half unit was tested over a wide range of operating conditions.
- Natural circulation. The EBSWR used data from proven natural circulation at operating Dodewaard and also data from Canada to understand the behaviour of the void in the chimney area.
- Fuel. The ESBWR uses the same fuel as that for the ABWR, except for the difference in height, as already explained before.
- FMCRD. The ESBWR adopts the FMCRD for the ABWR given the excellent operating experience.
- A full scale vertical GDCS Integrated Systems Testing (GIST) facilitywas developed at GE site to simulate gravity cooling system, i.e. to take into account the impact of gravity, although volumetrically scaled down. No unpredicted phenomenon was found on the ability of the gravity driven cooling system to provide flow into the vessel.
- BiMAC testing. BiMAC testing includes single channel test and system effects test. Different configurations were used, looking at angle of inclination, critical flow, and two-phase flow, under different simulated heat fluxes.
- Panda Full Height Containment Test facility is used to study passive containment cooling, to understand elevation and head, and others.



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FIGURE 4.6.14. ESBWR proven innovation

4.6.2.15. CR4.4.1 Risk informed approach

Mr Miller (GE Hitachi NE Intl.) noted that insights from the ESBWR PSA have been used to implement several design enhancements, and consequently, have contributed a significant improvement to nuclear safety. Examples of the use of PSA in the design process were provided.

- Addition of redundant and physically separated flow paths in response to fire analysis. There is alternative cooling system, which is not a safety class, but has a lot of benefits in certain sequence (In PRA if a system has a safety useful function then its benefit can be quantified). When looking at the fire, the system has one incoming path to the vessel. A fire in the area would disable that, so a redundant / alternate path is provided for the cooling.
- Fail-safe actuation of ICS. The ICS accommodates the pressure, converts and condensates steam so maintain the level in the vessel and reject heat to the atmosphere. It does not go into containment, i.e. not heat up the containment and no containment cooling is needed. In PRA analysis of different accident scenarios, in case of loss of certain control power, it would be better to modify the ICS to make it fail-safe.
- Addition of parallel injection line valves for the ICS to eliminate a dependency on power supplies.
- Change of fire suppression route to reduce room flooding frequency. This is because a pipe break could cause room flooding. The example shows the benefit of using PSA tools, looking not only from internal events and fire but also from internal flooding.

Mr Miller referred to the ICS diagram (*See figure in CR1.3.3*) and noted that the only thing that will stop the ICS in normal operation is the two parallel valves. The original design had one valve that needs to open. In the modified design, there are two valves in which at least one valve should be energized to close. So if the plant lost either DC power or control power, or both, the valve would automatically open – an indicative condition during containment isolation and transient when the ISC is needed to establish natural circulation flow path.

Mr Miller provided another example on operator actions. The PSA was intentionally pessimistic on estimation Human Error Probabilities. The designer did not go into a lot of details on estimating of performance shaping factor or doing time study on simulators, and instead used conservative values. Also

the designer did not take credit for recovering fail system. In some current plants in their PRA models, when reasonable, a pump that fails may restart within a reasonable amount of time. For the ESBWR, additional design feature is used so as not to rely so much on operator considering human makes mistake. So the designer makes use of identification of PSA accident sequences where operator actions are important, and then improvements with physical design changes.

Reflection on Lessons learned from Fukushima

At the end of all presentations, Mr Beard (GE Hitachi NE Intl.) addressed two key points.

Immediately after Fukushima, the US NRC convened working group with one of the tasks was to examine two designs APWR and ESBWR whether to reopen certification. The NRC noted the ESBWR passive design would have handled the Fukushima event well. The only recommendation was to enhance the spent fuel pool and instrumentation, which is currently undertaken. The GE Hitachi NE Intl convened a small team who agreed the NRC findings on spent fuel pool instrumentation, and discussed the use of hardware connection to portable generator.

On the debate on injection of salt water into the RPV like in Fukushima, he noted the pool for the ISC in the PCCS can be filled with any kind of grade water. In the ESBWR ICPCCS concept, the operator shall not hesitate to use salt water because the worst case will be to replace the heat exchanger as salt water will not damage the NSSS at all. This will only cause temporary shutdown ~2 years to replace the 100 million dollars components, saving the multibillion dollar investment.

4.6.3. Availability of References

The following references are available on the US NRC website:

- Safety Evaluation Report (NRC): http://pbadupws.nrc.gov/docs/ML1034/ML103470210.html
- Design Certification Document (DCD) Rev. 9: http://pbadupws.nrc.gov/docs/ML1034/ML103440266.html
- NEDO-33201, Revision 6, Probabilistic Risk Assessment ESBWR Design Certification: http://pbadupws.nrc.gov/docs/ML1028/ML102880536.html
- NRC Final Design Approval: http://pbadupws.nrc.gov/docs/ML1105/ML110540310.pdf

4.6.4. Discussion

Questions	Answers
Mr Arshad (Pakistan) requested a comparison of the ABWR and ESBWR. On a follow-up question, since the ESBWR seems much more reliable compared to that ABWR, Mr Arshad asked the rationale for developing ABWR.	**Mr Beard (GE Hitachi NE Intl.)** stated that primary differences are that the ABWR uses active safety to accomplish emergency core cooling and decay heat removal, and the ESBWR has converted those functions to passive, i.e. isolation condenser and PCCS for decay heat removal, and gravity driven core cooling. The second difference is the ESBWR eliminates the recirculation system. In order to enhance the natural recirculation, the reactor vessel height is increased, giving greater driving head. The huge volume of subcooled water is able to remove a lot more decay heat. During the initial response after scram, the back pressure from the two-phase flow through the flow decreases, and the subcooled water sweeps down into the core during the first 10-15 seconds and removes a lot of decay heat. The pressure in the vessel decreases for a short period of time even before isolation condenser comes into operation. On the second question, the key driving in developing ESBWR are lower the capital cost, increase safety, and decrease operation and maintenance cost. One key attraction with the ABWR is power manoeuvrability with the ability to change 10% power in 10 seconds using flow control, and then 5% per minute afterward. So if there is a grid condition with tremendous amount of renewable or not very stable, the ABWR could be attractive. The ABWR has 40+ years of experience while the ESBWR has not been deployed yet

	(currently in the very final stage being reviewed and certified by the US NRC).
Mr Dewayatna (Indonesia) asked if the 10-12 hour DC battery capacity was increased because the systems of the ESBWR are passive systems or to raise capacity.	**Mr Beard (GE Hitachi NE Intl.)** said that it is a combination of both but with the introduction of the passive systems there is a very low demand on the batteries. The primary demand is for the I&C systems so there must be sufficient batteries to operate the I&C for 72 hours. But the motor forces for core cooling or heat removal are entirely passive so there is no requirement for electricity.
Mr Koshy (IAEA) asked how the isolation condenser comes into service, if it purely valve principle or if it can work in the absence of DC and AC power. He also asked how the other relief valves work for depressurisation.	**Mr Beard (GE Hitachi NE Intl.)** explained that for the ICs they are globe type valves which are held closed by pneumatic pressure so the solenoids that are maintaining the pneumatic pressure and are normally energised, would de-energise and vent the pneumatic pressure to force the valves open establishing the flow path for the isolation condenser to go into operation. For depressurisation, there are two different stages. In the initial stage there are 10 safety relief valves with external actuators which will be open to release the steam in the suppression pool to affect the initial part and then there are 8 depressurisation valves which are pyrotechnically actuated, relying on a very small electrical current to initiate the chemical charge to open the valves. Therefore the valves rely on DC but depressurisation is only necessary in a LOCA scenario. In a SBO there is still an intact reactor cooler pressure boundary then there is no need to go into the depressurisation phase.
Mr Koshy (SH-NPTDS) asked how the isolation condenser valves are triggered to open.	**Mr Beard (GE Hitachi NE Intl.)** stated that they are triggered on sensing low water level within the RPV. There are 4 isolation condensers and 2 valves in parallel so either of the valves opening will put the system into operation. So when the low water level is reached then both valves are commanded to open and one of the valves is also designed as the fail-open valve.
Mr Popov (Belarus) asked if the natural circulation has been experimentally confirmed.	**Mr Beard (GE Hitachi NE Intl.)** confirmed that there is experimental and practical data. The largest pressure drop occurs in the fuel assembly and there is lot of data from just fuel design process. There are additional minor changes to the spacer elements to decrease the pressure drop even further. There is a test facility at Canadian Hydro to study the chimney assembly partitions that are used to minimize possibility of horizontal eddies, to see what possible pressure drop given the two-phase flow. From the practical side, the Dodewaard BWR in the Netherlands (much smaller in thermal power) had agreed to perform some additional test, prior to closing, to obtain correlation with Dodewaard database.
Mr Grudev (Bulgaria) asked for what reason GE Hitachi NE Intl. decides to create such high power output plant with such a large reactor vessel that may be difficult to deliver.	**Mr Beard (GE Hitachi NE Intl.)** said that this was decided on the base of the economies of scale and the initial market for potential deployment by the European and US utilities which 15 years ago were all looking for large base load capacities. A smaller plant would also be too expensive on a per kW basis.
Mr Silva (Thailand) recalled the results of internal events CDF shown in the presentation and asked if the external events such as large earthquakes would change the CDF a lot, and secondly whether the engineered safety features would still be reliable in case of large earthquake.	**Mr Miller (GE Hitachi NE Intl.)** replied that the ESBWR is based on the US design but with the intention of accommodating regulations of any other host country. The requirements in the US are to perform a seismic margins assessment and not a seismic PRA. So GE did not quantify the seismic PRA however there is still a lot of research going on to estimate profiles and frequencies and other such factors. On the second question, that would dominate due to uncertainty. The plant is designed to a

	higher peak ground acceleration of 0.5g so it is very robust. But the seismic hazard probability of an earthquake even 1 in 100,000 years is still more frequent that the CDF. So the number is not important, but the insights. The important things is what ESBWR will do to survive a major upset are to scram the reactor, close the main steam isolation valves and operate the isolation condensers: few components, no rotating equipment, no relays and very robust from a seismic standpoint. There are a lot of features to mitigate such an event. **Mr Beard (GE Hitachi NE Intl.)** added that g-force described is actually slightly higher frequency content, 2-100 Hz, which as described, designed for higher frequency content, thus more realistic to get that high peak ground acceleration.
Ms Yang (China) asked if there is any experiment of the core catcher.	**Mr Miller (GE Hitachi NE Intl.)** replied there are a series of experiments on BiMAC, a semi-scale with tubes and full scale with single tube, with different arrays, to validate the phenomena and model them on computer.
Mr Randy Beatty (USA) asked about the usefulness of INPRO criteria for the vendors, i.e. it may not be useful to enhance the design (also vendor's are driven by utility requirement). But INPRO requirements representing the international standards may be useful for countries considering purchasing the reactors, to help them feel confident, that the assessment is a useful exercise for country's particular application.	**Mr Miller (GE Hitachi NE Intl.)** responded the criteria are valid reflecting key features.
Mr Tjahjono (Indonesia) asked in the case of pipe break between reactor vessel and the injection gate valve, whether the injection is still effective.	**Mr Miller (GE Hitachi NE Intl.)** indicated in the case of pipe break, it is possible a LOCA will occur. But there is still adequate capacity to flood the core. The lower drywell can flood up in fact above the level, and cover the pipe, i.e. there is water above the core too.
Mr Dulera (India) asked if in the core catcher there is a vent for gases build up in the core catcher, and if there is a chance that hydrogen gas can be produced in the core catcher.	**Mr Miller (GE Hitachi NE Intl.)** responded that in the core catcher, hydrogen generation will have occurred earlier and probably still occurring regardless of the core catcher itself but there is no reason why the non-condensable gases such as hydrogen and oxygen would accumulate in the bottom of the core catcher. They would be driven by the natural circulation of the water. They would have a tendency to migrate up towards the passive containment cooling system. There is no need to vent the gases because the pipes of the core catcher are inclined so the gas would migrate upward.
Mr Susyadi (NPTDS) asked how long it takes to bring the reactor to full power during start-up, considering there are recirculation pumps.	**Mr Beard (GE Hitachi NE Intl.)** informed that it takes generally between 24 to 28 hours to go from cold conditions to 100% power operation.

4.7. EPR (European Pressurized Reactor)

Mr Franck Lignini from AREVA (France) presented INPRO assessment in the area of safety of the European Pressurized Reactor (EPR) design.

4.7.1. Overview of EPR Design

Mr Lignini briefly noted the EPR™ currently under construction in Flamanville (France) and Taishan (China) and then discussed the following aspects of the EPR™ design:

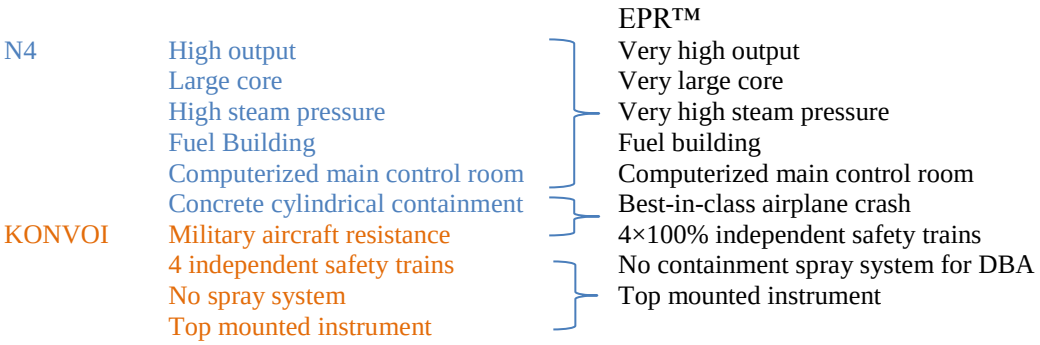
- Design key drivers

  There are four main drivers to the design:
  - Take advantage of lessons learned from decades of existing commercial plants (operating PWRs) to develop an evolutionary design
  - Reduce power generation costs by means of larger size PWR (~1600 MWe), longer plant lifetime (up to 60 years), increased availability (target>92%), decreased O&M costs (reduce typical outage to 16 days), and reduced fuel cost (7 to 15%) for enhancing the economic aspects.
  - Further increase safety levels, being reduced collective doses during normal operations, reduced probability of core meltdown by a factor of 10, reinforced deterministic DID, and reinforced resistance to external aggressions such as airplane crash contribute to further increase safety levels.
  - Further decrease volumes of ultimate waste through reduction of liquid waste (by ~10% versus EDF existing best plants), gaseous waste (by 30 to 50% versus EDF existing best plants) and irradiated material by 20 to 30% in particular due to utilization of 60 GWd/t fuel burn-up.

  Mr Lignini (AREVA) said that these key drivers are the reason why AREVA chose those criteria for the Dialogue Forum.

- Building on N4 and Konvoi achievements

  Mr Lignini stressed the fact that the EPR™ reactor, in fact, has been developed by combining the best technology based on the France and Germany reactor. France and Germany had the N4 reactor (1,475 MWe) and KONVOI reactor (1,365 MWe) respectively around the 1990s, both of which were evolutionary reactors at the time. In total there were 69 NPPs in France and Germany which was integrated into the design of EPR™ reactor. Thus EPR™ was built on N4 and KONVOI's achievements.



Type of plants	N4	EPR	KONVOI
Core thermal power (MWth)	4250	4590	3850
Electrical output (MWe)	1475	1660	1365
Number of loops	4	4	4
N° of fuel assemblies	205	241	193
Type of fuel assemblies	17x17	17x17	18x18
Active length (cm)	427	420	390
Total F.A. length (cm)	480	480	483
Rod linear heat rate (W/cm)	179	166,7	167
N° of control rods	73	89	61
Total flowrate (kg/s)	19420	22256	18800
Vessel outlet temp. (°C)	330	330	326
Vessel inlet temp. (°C)	292	295.2	292
S.G.: heat exch. Surface (m²)	7308	7960	5400
Saturation Steam Pressure (bar)	73,1	77.2	65.5

FIGURE 4.7.1. Development of EPR by Combining N4 and KONVOI

- Key technical characteristics

The EPR™ has a net output of 1600 MWe with a net efficiency of about 37%. It is expected to have target availability of 92% and operation cycle of up to 24 months with design service life of 60 years.

TABLE 4.7.1. Several Specifications of EPR Design

Thermal power	~ 4600 MWth
Gross output	> 1750 MWe
Net output	> 1600 MWe
Net efficiency	Up to 37%
Target availability	92%
Number of loops	4
Steam pressure (secondary)	~ 77 bars at 100% power
Operation cycle length	Up to 24 months
Design service life	60 years

- Reactor core characteristics

  The EPR™ has a large size core comprising 241 fuel assemblies to provide the large output. It has low linear power of 167 W/cm. Better utilization of fuel is achieved with high burn-up (60GWd/t) and the use of neutron reflector. The design has flexibility in fuel utilization with a possibility to have up to 50% MOX fuel in core without any specific provision, and up to 100% MOX fuel taken into consideration right from design. Mr Lignini indicated that for the INPRO assessment, the use of MOX fuel in EPR™ reactor was not taken into account.

- Safety systems

  Mr Lignini pointed out several main safety systems, such as large amount of cooling water inside the reactor building, emergency feed water tank in safeguards building in which there is a 400 m^3 tank for emergency feed water pool in each of the 4 divisions (see figure on installation in the safeguards and auxiliary building), and also 2600 m^3 tank for firefighting system.

  Both redundancy and diversity for the EPR™ are illustrated in the figure below, where safety functions are achieved with the use of both passive and active systems.

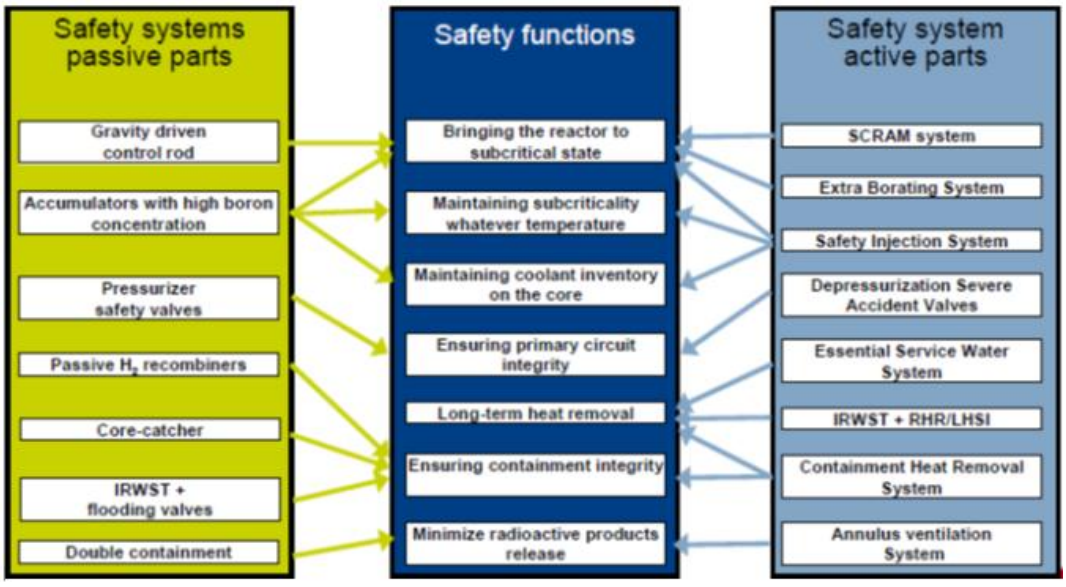


FIGURE 4.7.2. Diversity of Safety Systems of EPR™ Design

An overview of the reactor main fluid system inside the reactor was provided, starting with reactor coolant system (chemical and volumetric control system), Safety injection system in the emergency

borating system, the containment heat removal system, emergency feed water system and the ultimate heat sink.

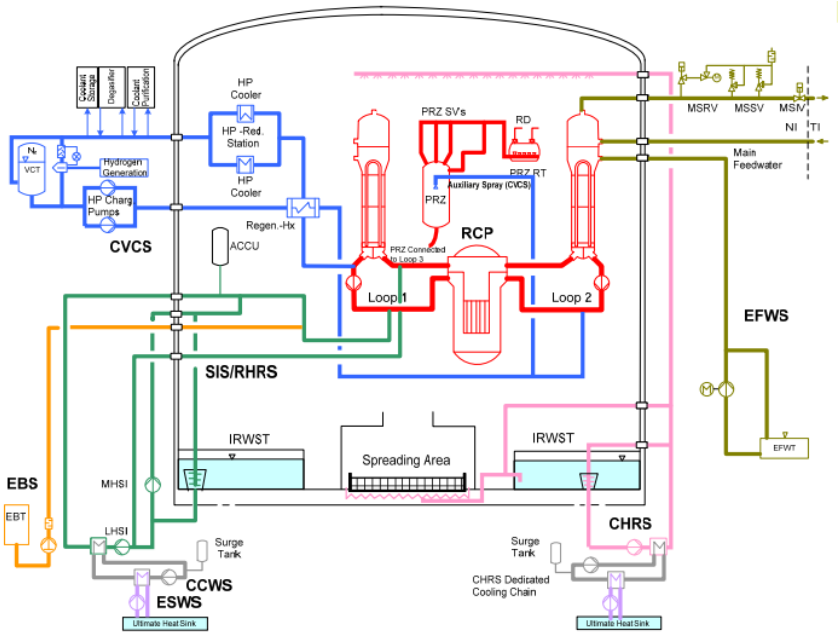


FIGURE 4.7.3. The EPR™ Reactor Main Fluid System

The safety systems in EPR™ are divided into 4 divisions, in which 2 divisions are housed inside building resistant to airplane crash while the other 2 divisions on both sides of the containment building are not. In addition, the fuel building containing the emergency borated system (EBS) is also protected against airplane crash.

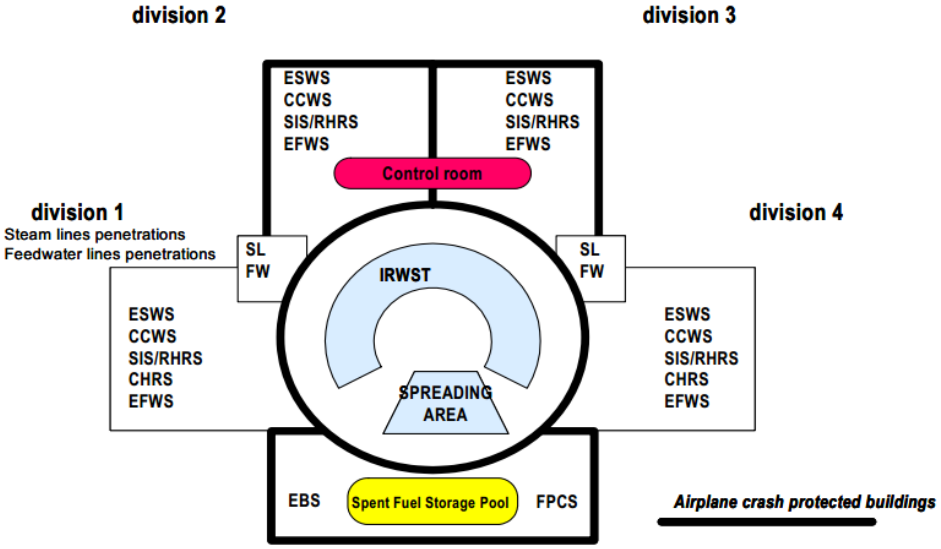


FIGURE 4.7.4. Installation in the Safeguard and Auxiliary Building

- Principles for safety classification of systems

  There are three classes of safety functions corresponding to whether the functions are needed for control state, shutdown state or final state.

  - F1 Functions:
    - F1A Functions: all safety functions which are required to reach a controlled state after a DBC event
    - F1B: all safety functions required, once the controlled state is reached, to reach and maintain shutdown state after a DBC event
  - F2 Functions

- Functions related to the monitoring and controlling of radioactivity during normal operation
- Safety functions required to reach and maintain, from 24 up to 72 hours, the safe shutdown state after a DBC event
- Safety functions needed to reach final state for DEC-A sequences or to prevent significant releases in DEC-B sequences

- Non-Classified Functions

Note that shutdown state is defined as:
- Controlled state (for DBC 2 to 4): the core is subcritical, heat removal is assured on short-term basis (max. 24 h), core water inventory is stable and radioactive releases remain acceptable.
- Safe shutdown state (for DBC events): the core is subcritical, residual heat removed on a long term basis through the cooling chain and radioactive release remain acceptable.
- Final state (for DEC-A): the core is sub critical, residual heat is reliably removed at least up to the plant autonomy limits, and radioactive releases remain acceptable.

- Allocation of main I&C functions

  The EPR™ has the following lines of defence:

  - Preventive line of defence
    The preventive line of defence is comprised of Process Automation System (PAS) for functions which do not use control rods as actuator, and Reactor Control Surveillance and Limitation system (RCSL) for functions which use control rods as actuator. The RCSL is aimed at avoiding the reactor trip in case of deviations such as reactor coolant pump failure so a partial trip is generated, not a full scram.

  - Main line of defence: Design Basis Accident (DBA)
    The main line of defence is comprised of Reactor Protection System (RPS) to reach control state by means of monitoring safety parameters and actuation of reactor protection and safety features, and Safety Automation System (SAS) to reach safe state, i.e. post accident management (PAM) to bring the plant into and maintain a safe shut-down state.

  - Risk reduction line for Design Extension Conditions (DEC)
    The risk reduction line is comprised of Diverse Actuation System (DAS) to cope with a common cause failure of I&C systems, and Severe Accident I&C (SA-I&C) to mitigate severe accidents.

- I&C Human Machine Interface

  EPR™ has two main man-machine interfaces. The first one is the Primary Human Machine Interface (Primary HMI) also known as Process Information and Control System (PICS) which is used to monitor and control the plant in all conditions. It is basically screen based, integrated and fully computerized. It is implemented in the main control room as well as in the remote shut-down station. The second one is the Secondary Human Machine Interface (Secondary HMI) known as Safety Information and Control System (SICS) which is mainly used if PICS is not available. It mainly consists of conventional monitoring and control means dedicated to the individual automation systems. It is implemented in a dedicated area in the main control room. It was noted that depending on the requirements of national regulators, the design is flexible enough particularly the layout of building to accommodate additional cubicles in particular, e.g. for non-computerized safety system as required by some European regulator.

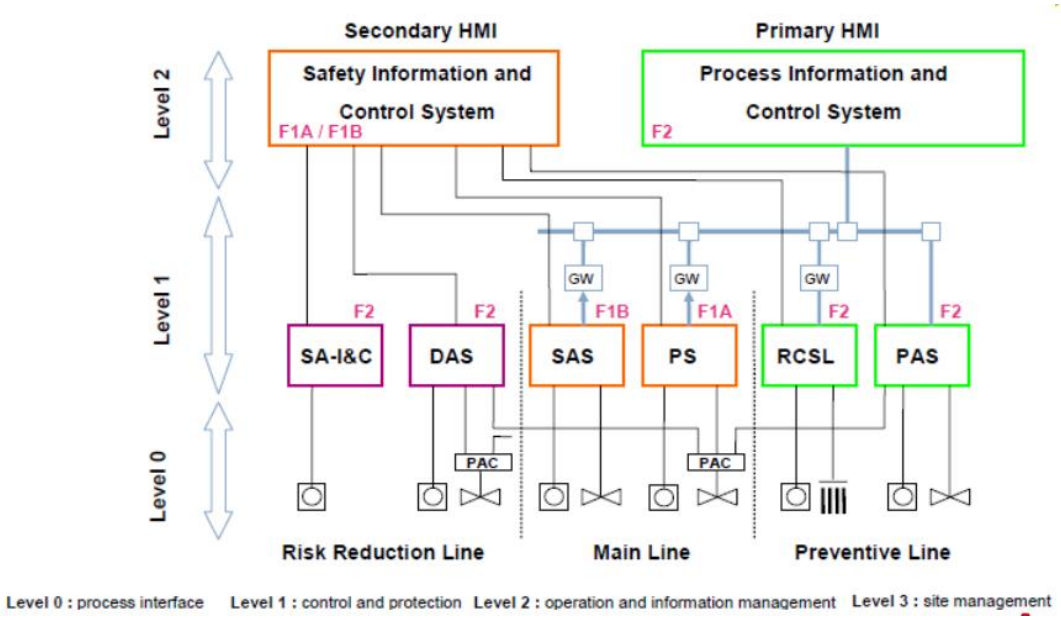


FIGURE 4.7.5. The I&C Overall Architecture

- EPR™ reactor certified by EUR and AREVA Worldwide engagement in EPR licensing

  99.9% of EPR™ features comply with EUR requirements and the certificate attesting to the EPR™ reactor's compliance with European utility requirements was officially awarded to AREVA on July 15th 2009. The EPR™ reactor has outstripped the EUR specifications in the following areas:
  - Protection against the airplane crash (the risk of the large commercial aircraft is covered)
  - Spent fuel storage capacity
  - Life expectancy of 60 years against 40 required by the EUR

  AREVA has worldwide engagement in EPR licensing with several countries, i.e. France, Finland, China, UK and the US. Mr Lignini pointed out that the EPR™ reactor is the only Gen3 model under construction in the EU, the only reactor designed within the EU, and the only reactor licensed by 3 independent European regulators. Furthermore, AREVA continues to incorporate lessons learned not only from operation but also from engineering process, standardizing early engineering / construction activities. One concrete milestone achieved by AREVA is the reduction of ERP™ construction duration from 47 months in Olkiluoto (Finland) to 24 months in Taishan (China).

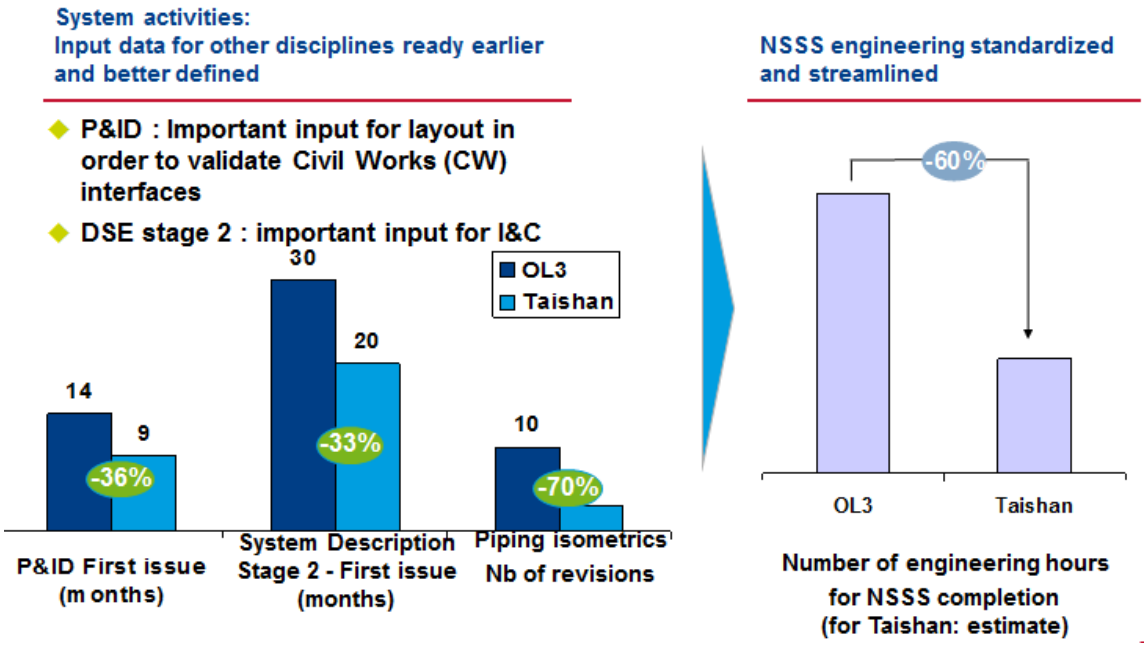


FIGURE 4.7.6. Standardization of Early Engineering Activities

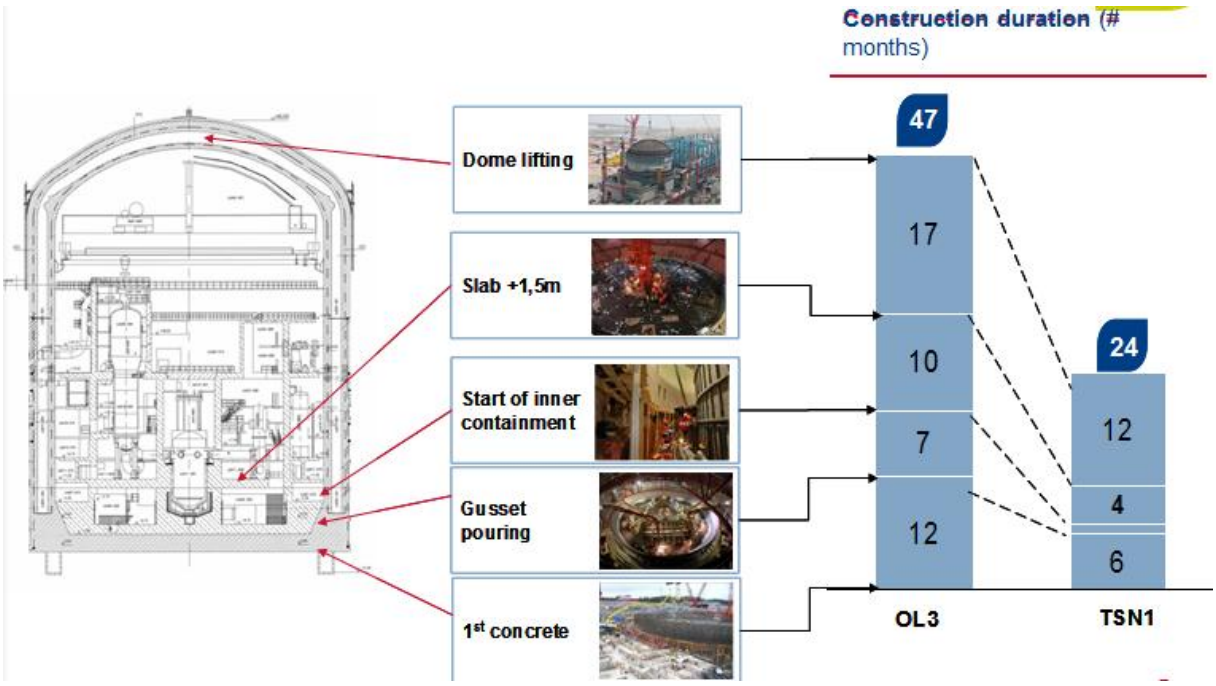


FIGURE 4.7.7. Reduction of Construction Duration as the First Main Milestones

In conclusion, Mr Lignini made the following summary:

- The EPR™ reactor is an evolution of mature French and German reactors, incorporating proven technologies as well as innovations validated through R&D programmes.
- The Safety features of the EPR™ reactor integrate the lessons learned from the past. They have been deemed necessary and sufficient by the safety authorities of advanced nuclear countries as fulfilling the most stringent Gen III safety objectives, until March 2011. Although the lessons learned from the Fukushima event are not yet all drawn, preliminary evaluations show that the EPR™ reactor would have withstood the earthquake and the subsequent tsunami. The soundness of the EPR™ reactor design was confirmed by the complementary safety evaluations (stress tests) performed by the European Regulators; however light improvements may be incorporated to satisfy new requirements (Example, additional connection point for pump to inject water).

4.7.2. INPRO assessment in the area of safety of EPR

4.7.2.1. EP1.1.1.1 Margins of Design

The EPR™ reactor was designed for being capable of high performance, for example, in load following and frequency variation control (between 25 and 100% up to 90% of the fuel cycle length) and low power operation (without penalty on the possibility to regain nominal power after which it would be longer to go back to nominal power). The fuel cycle length for EPR™ is increased compared to the operating fleet up to 24 months. The linear power rate is lower than currently operating PWRs in France.

The ERP™ reactor is licensed and meets the safety requirements in France, China, Finland and UK and the design certification is underway with the US NRC. Some safety margins are demonstrated and provide the highest level of confidence on how the EPR™ reactor to cope with normal operation, AOOs, incidents and DBAs.

Mr Lignini (AREVA) illustrated margins, i.e. safety margin and operation margins.

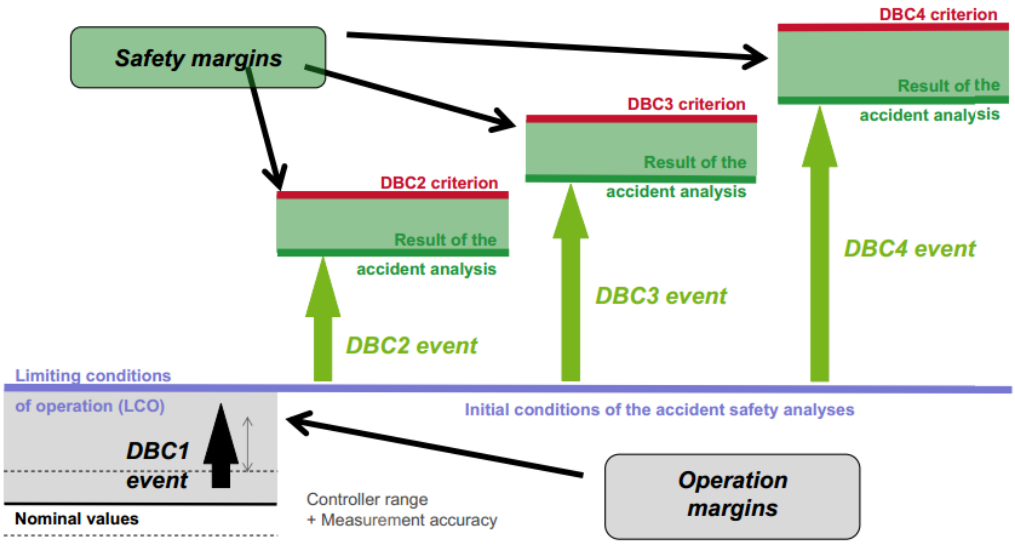


FIGURE 4.7.8. Safety margins and operation margins: DBC events

Given normal values and variations to take into account for example control and accuracy of measurements, normal transients will take the operation to limiting condition, thus constituting operation margins. These limiting conditions of operation (LCO) can be considered as the initial conditions of the accident safety analyses. For each kind of event, the EPR™ has different criteria to be satisfied. For a particular event in the accident analysis, when the result found is below the criterion and the gaps constitute the safety margins.

Mr Lignini (AREVA) noted DBC acceptance safety criteria adopted in the EPR™ which should be the same for other PWRs.

- Safety criteria are expressed in terms of radiological consequences
  - No short term countermeasures (shelters, evacuation, distribution of iodine tablets), no need for long-term relocation, no long-term restriction on the consumption of foodstuffs.
  - Effective dose (based on a 7 day exposure in 500 m distance from the plant border) < 10 mSv (sheltering)
- Decoupling criteria (examples for illustration)
  - No departure from nucleate boiling (DNB) in any DBC-2 event, and in the DBC-3/4 events involving a failure of the secondary side shell (e.g. main steam line break)
  - Fuel rods experiencing DNB in DBC-3/4 lower than 10%.
  - Decoupling criteria for LOCA:
    - Peak cladding temperature lower than 1200° C
    - Maximum cladding oxidation lower than 17% of the cladding thickness
    - Maximum hydrogen generation lower than 1% of the amount that would be generated if all the active part of the cladding had reacted

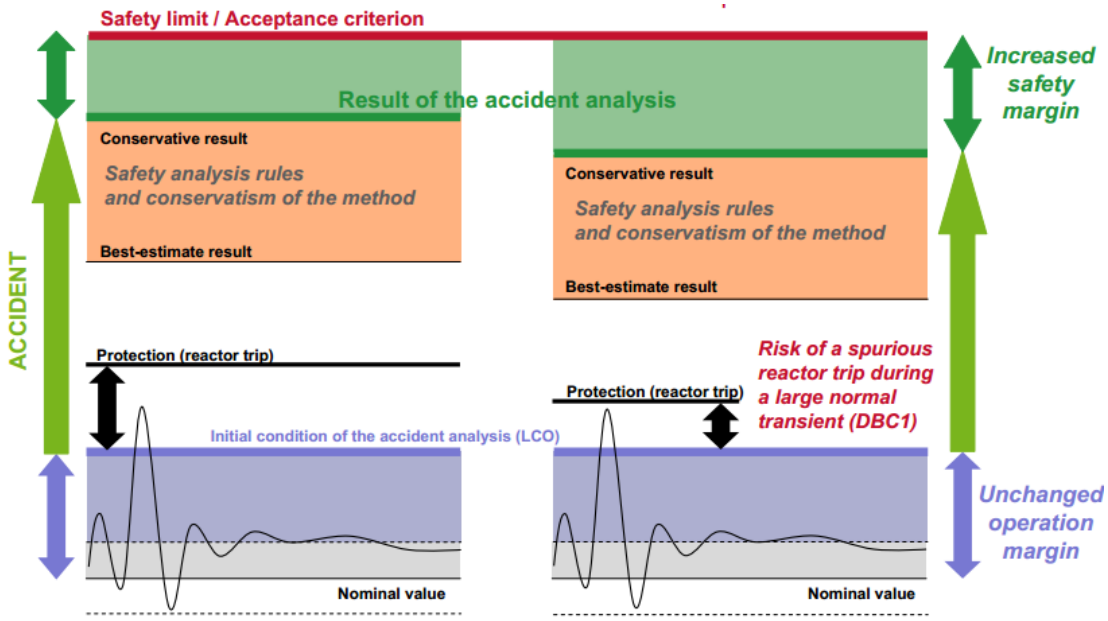


FIGURE 4.7.9. Balance Between Safety Margins and Operation Margins

Mr Lignini (AREVA) discussed balance between safety margins and operation margins, as shown in the figure. In case of event, the protection system will initiate reactor trip and there will be response of the mitigation system. Depending on how the analysis is performed, there are best estimate result and conservative result and also safety margins before the acceptance criteria. In order to increase the safety margin, designer has to reduce the operation margin. To increase the reactor margin and not to decrease the operation margins, then there will be a risk of spurious reactor trips during normal transient.

So, on margins of design versus margins of operation (EP1.1.2.1) there is a necessary compromise for new design between increased safety margins and increased operational performance. For most penalizing transients regarding safety function 'Control of reactivity', assessor has to take into account other criterion, i.e. CR1.3.6 Sub-criticality margin, which in particular the CR asks that sub-critical be met in the shortest time and reliably kept in sub-critical considering uncertainties and worst single failure in the shutdown system. Depending on national circumstances, designer may have to consider some specific requirements from regulator. To cope with these requirements and meet specific criteria, designer may have to adapt, .e. g. design of fuel elements, fuel pattern in the core, efficiency and different type of control rods, pattern of Rod Control Cluster Assemblies (RCCA), operation of the Safety injection System (SIS) and of the Extra Borating System (EBS), and I&C (signals + actuation of equipment). Hence, there is no a single design and there is flexibility in the design to cope with the specific requirements.

Mr Lignini provided several examples of margins of design:

- Reduction of number of reactor scram. Failure of one reactor coolant pump (RCP) implies a partial trip to approximately 40% of full power. Plant can still be operated in three-loop operation with maximum reactor power of approximately 70% full power. The RCP can be returned to service once the reactor power is reduced to approximately 30% full power.
- RPV nozzle shell. The higher nozzle axis improves fuel cooling in case of LOCA.
- Heavy reflector (HR) reflects effectively a large amount of escaping fast neutrons back to the core.

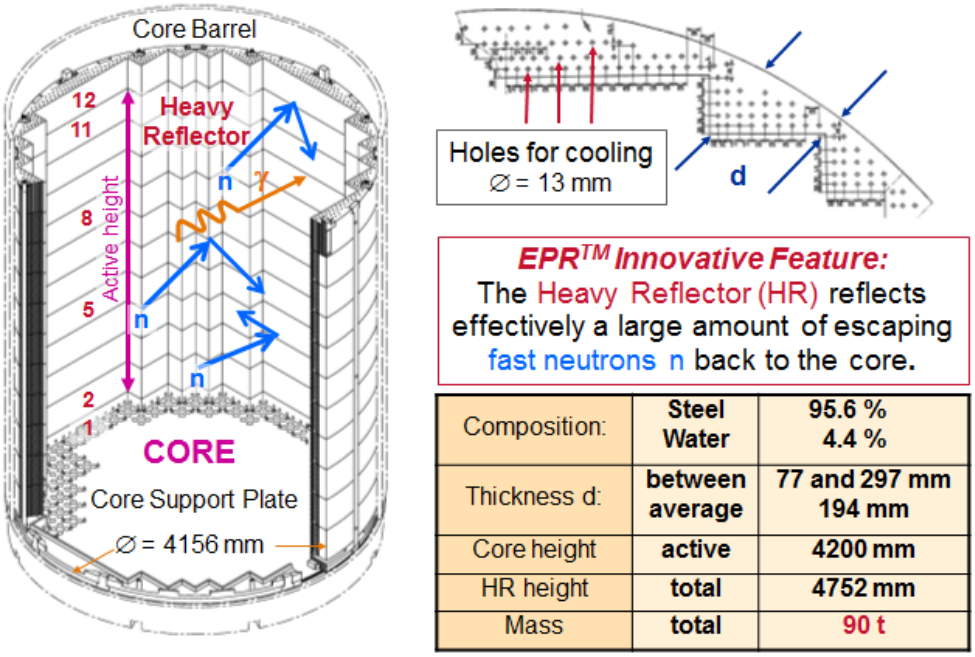


FIGURE 4.7.10. Heavy reflector in EPR™ reactor

- Flow Distribution Device (FDD). The enforced reactor coolant flow in the downcomer annulus generates strong turbulences in the RPV lower plenum, which are homogenized by means of the FDD. Hence, the FDD results in a distinct smoothing of coolant flow, i.e. uniform flow distribution at core inlet, better cooling of all fuel rods, and decreased probability of DNB.

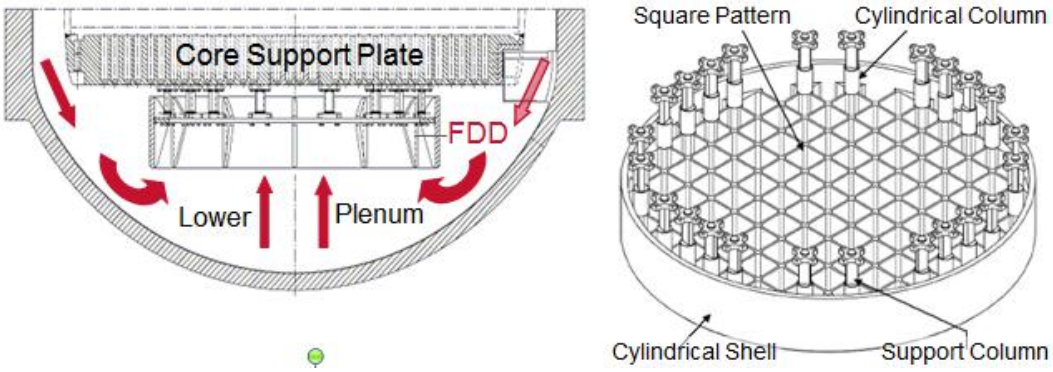


FIGURE 4.7.11. Flow Distribution Device (FDD) for smoothing of coolant flow

Mr Lignini (AREVA) considered that from reviewing the publicly available information, e.g. regulatory bodies who have licensed the EPR™ reactor or which are about to deliver Design Certification), an INPRO assessor may conclude that the EP 1.1.1.1 on 'Margins of design' is met.

4.7.2.2. EP1.1.1.2 Simplicity of Design

The EPR™ reactor has ~20% less equipment than the current 4-Loop US PWRs. To illustrate, study on pumps and turbines, heat exchangers, tanks and valves as shown below indicates reductions in terms of absolute value (except for pumps and turbines) and the normalized value per MWe.

TABLE 4.7.2. Reduction of equipment in the EPR

	EPR	4-Loop PWR	% Change (Absolute)	% Change (Count/MWe)
Pumps & Turbines	43	37	16	(16)
Heat Exchangers	34	44	(23)	(44)
Tanks	23	33	(30)	(50)
Valves	2,044	2,766	(26)	(47)

Other example is the Safety Infection System (SIS) in which the SIS configuration for EPR™ is shown to be simpler than that of N4 and KONVOI. The SIS itself is a 4-train system without headers between trains, hence there is 1 train per loop. Mr Lignini explained the reason why 4-train has the legitimacy by assuming that in case of LOCA if 1 train is lost at the break, then a single failure criterion (SFC) is considered, i.e. an additional train fails and another train is in the preventive maintenance, then the 4th train should be sufficient to cope with the LOCA (100% of the safety function).

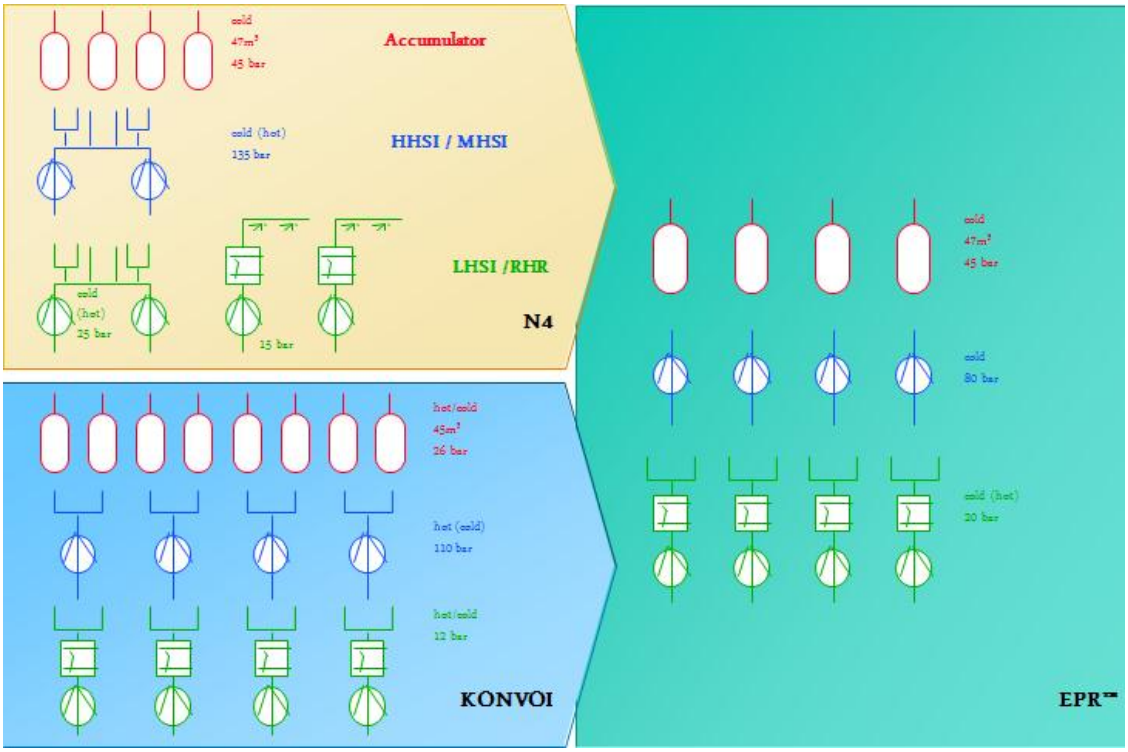


FIGURE 4.7.12. Safety Injection Systems of EPR™ in comparison with N4 and KONVOI

Further example of simplicity is given the fact that cooling water is now located inside the reactor building, i.e. IRWST, the 4 trains (LHSI, MHSI) can take suction from the IRWST hence there is no recirculation switch-over sequence in case of LOCA.

The SIS also can cope will all the LOCA assumed in the safety analysis (DBC 3 and 4) without the need to actuate containment heat removal system in spraying mode due to good condensation of steam, in particular after ~2 hours of possibility to switch safety injection from cold leg to hot leg. Containment spray is only need during severe accidents. The Low Head SIS (LHSI) also performs the Residual Heat Removal (RHR) function during plant shutdown.

Mr Lignini (AREVA) believed that from reviewing the publicly available information (e.g. regulatory bodies who have licensed the EPR™ reactor or which are about to deliver Design Certification) an INPRO assessor may conclude that the EP 1.1.1.2 on 'Simplicity of design' is satisfied.

4.7.2.3. EP1.1.1.3 Quality of Manufacture and Construction

Mr Lignini discussed several examples of improvements made in the EPR™ reactor design.

- Reactor Pressure Vessel (RPV)
  - General improvements of structure

    These include (i) alloy 600 has been systematically replaced by Alloy 690, (ii) neutron influence on the core shell region has been reduced with the use of heavy reflector, and (iii) manufacturing of one piece RPV upper part instead of two welded pieces (for N4) has been achieved.

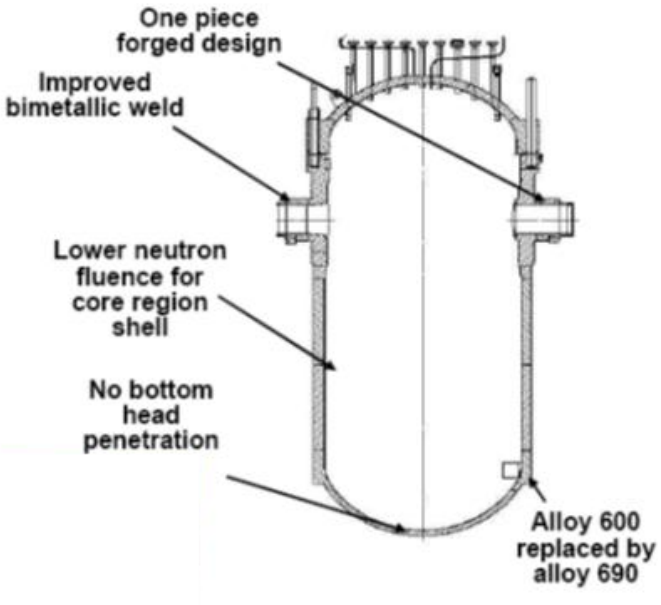


FIGURE 4.7.13. General Improvements on EPR™ Reactor Pressure Vessel Structure

- Reactor Pressure Vessel (RPV) internals

  Mr Lignini (AREVA) discussed the main design features that took advantage of general arrangement similar to N4 reactor, namely (i) similar flow distribution in upper plenum and closure head dome, (ii) forged core support plate, (iii) fuel assemblies resting directly on the core support plate, (iv) same principles regarding centering and maintaining inside the RPV, and (v) same materials with more stringent requirement on Co residual content. In addition, there are 3 main modifications made to incorporate German KONVOI design: (i) adaptation to an in-core instrumentation introduced through the RPV closure head (KONVOI design) i.e. there is no more penetration in the bottom part of RPV, (ii) installation of a dedicated annular structure under the core support plate to ensure a proper flow distribution at the core inlet, and (iii) incorporation of an innovative component which is the heavy reflector to contribute of protection of RPV irradiation.

  Overall, the general improvements on the RPV internals are (i) increased margin with respect to RPV embrittlement achieved through neutron fluence reduction, i.e. new heavy reflector design reducing neutron leakage at the core periphery and enlarged diameter of RPV, (ii) prevention of low induced vibrations (R&D / experimental programs performed showed all modifications do not generate vibrations, in fact, resulted in reduction of the vibrations) and (iii) better inspectability and easier replacements given the new design, i.e. the distance between heavy reflector and RPV is favourable regarding the inspection of the bottom part of the RPV.

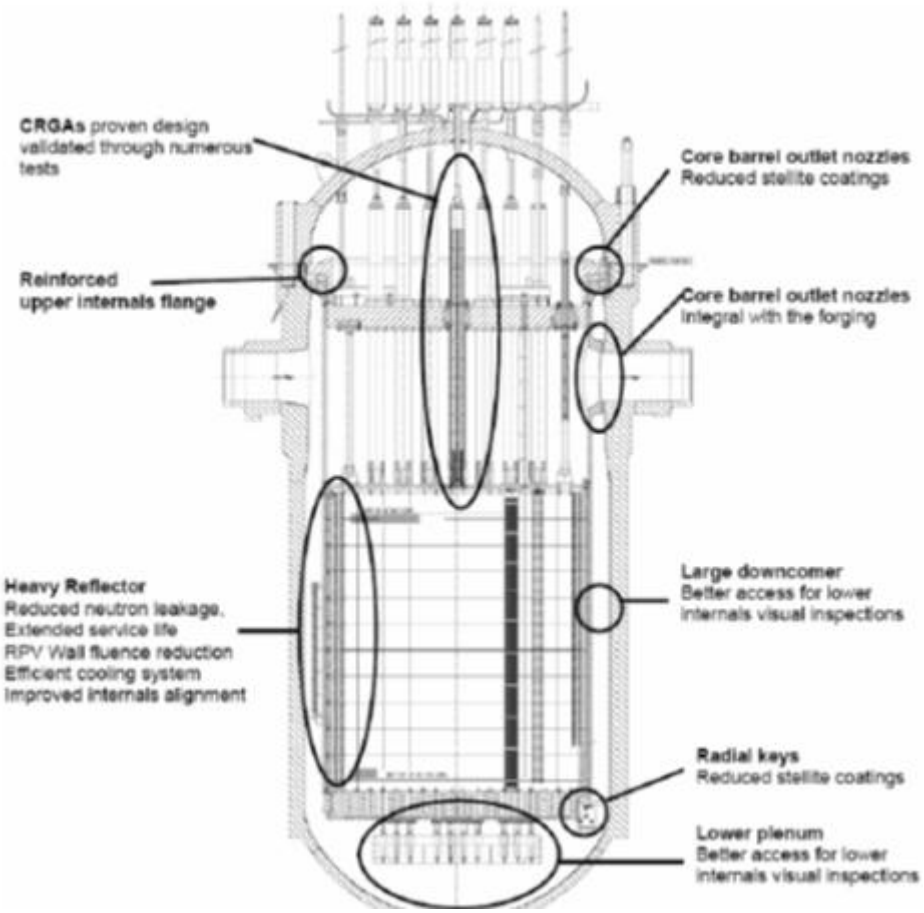


FIGURE 4.7.14. General Improvements on EPR™ Reactor Pressure Vessel Internals

- RPV lower internals: heavy reflector

  Replacement of conventional core baffle assembly by a heavy reflector contributes to a reduction in the fuel cycle cost, improvement of long term mechanical behaviour of lower internals and protection of RPV core shell against radiation embrittlement. Some improvements of the mechanical behaviour are no bold in the most irradiated areas, well managed temperature distribution in the structure, and very low depressurization effects in case of LOCA.

- Reactor Coolant Lines (RCLs)

  There is a significant reduction in the number of welds in the Reactor Coolant Lines. In particular the big nozzles have been integrated in the main coolant lines at the forging stage, which is an improvement in the integrity of the pressure system.

  There is improvement of the Leak-Before-Break concept. A minimum number of auxiliary nozzles are provided on top of the RCLs, leading to reduction of leak risk. The Extra Borating System (EBS) injection is introduced into the Reactor Coolant Line via CVCS or SIS nozzles.

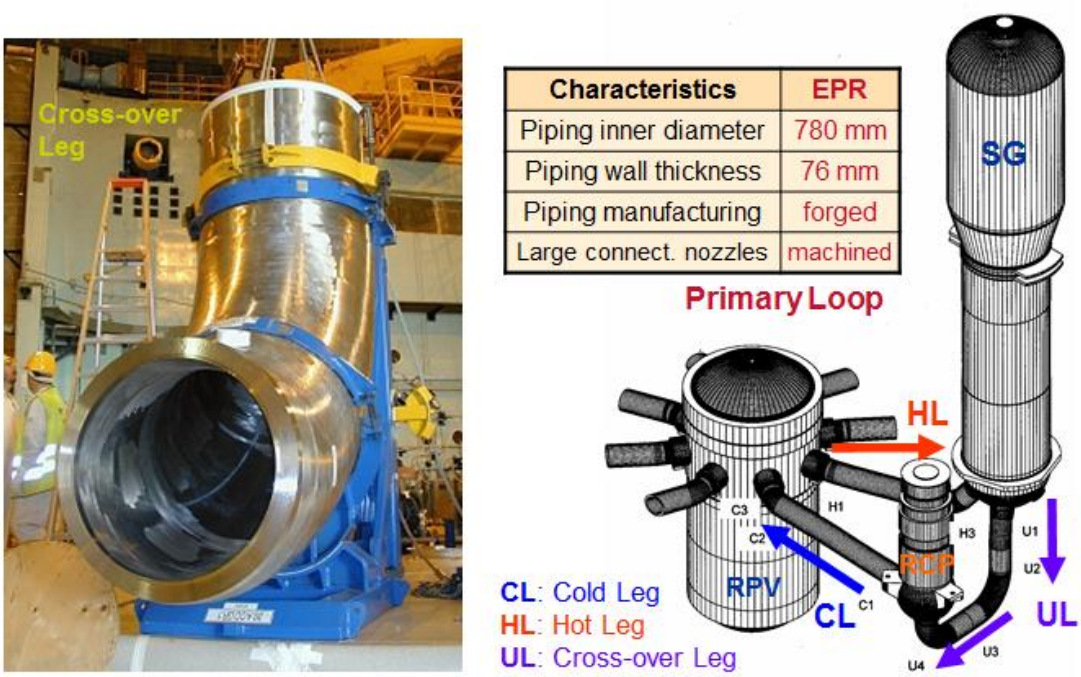
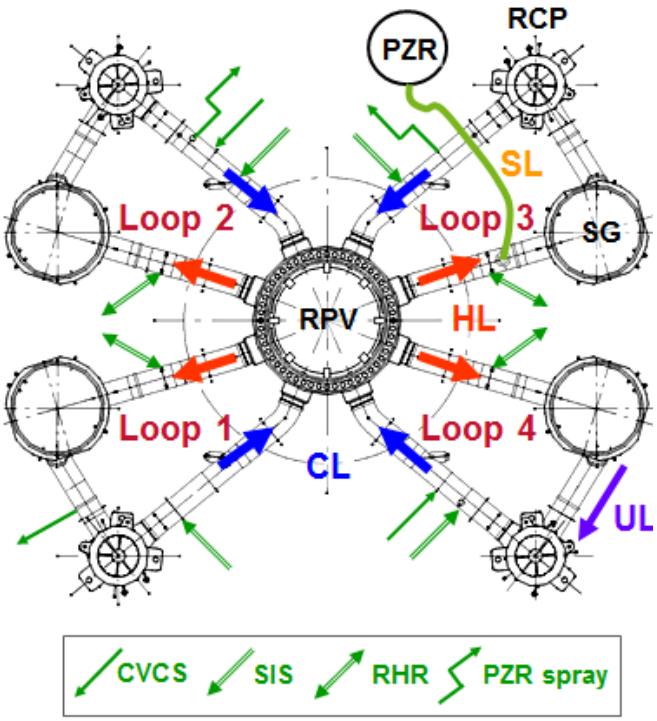


FIGURE 4.7.15. Reactor Coolant Lines in EPR™ Design



CVCS: Chemical & Volume Control System; SIS: Safety Injection System; RHR: Residual Heat Removal (System); PZR: Pressurizer; SL: Surge Line

FIGURE 4.7.16. Reactor Coolant Lines Layout in EPR™ Design

- Pressurizer

  There are improvements regarding:
  - Structure integrity with the reduction in the number of lances
  - Support welding optimization, i.e. reduction of weld length for the support attached to the shell of the pressurizer. Fatigue analysis has been performed to confirm feasibility of the improvement)
  - Mechanical behaviour for the bottom part by modification of the configuration of the heaters flanges from a square net to 4 circles, thus reducing manufacturing risk. Installation (welding) of sleeves is much easier (less radii to be considered: 4 vs 8). Accessibility for the required maintenance around the heaters flanges was checked.

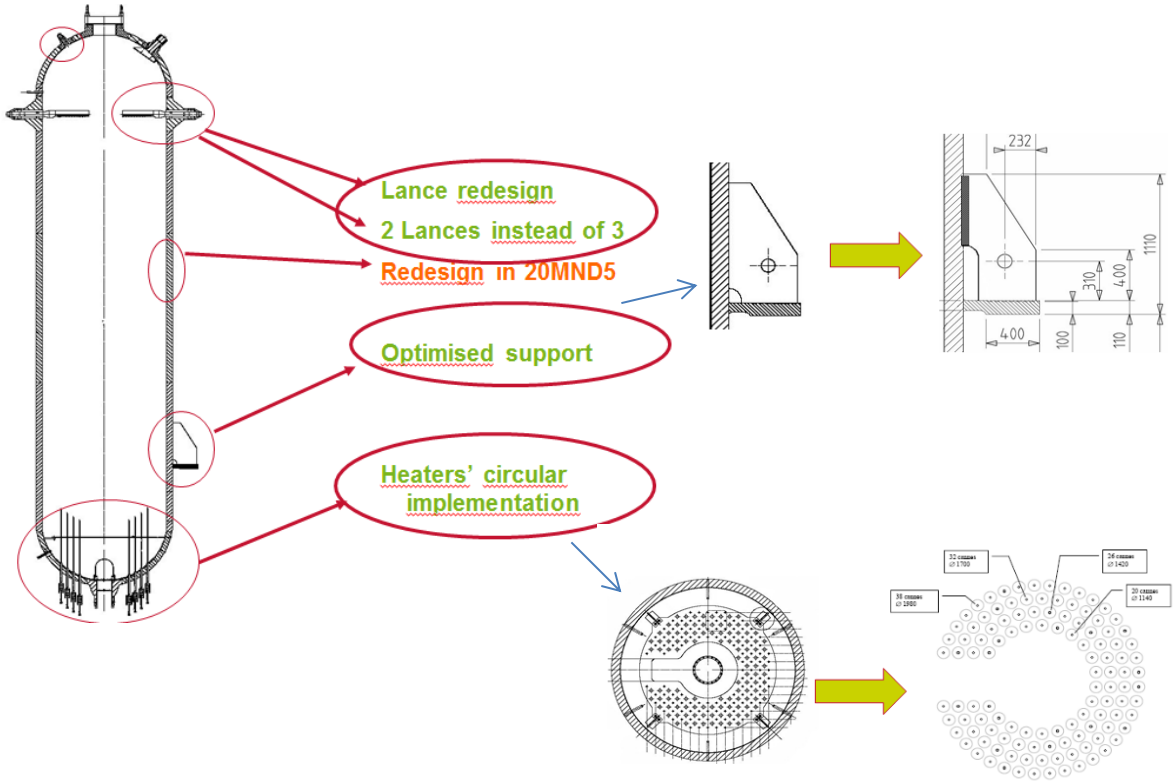


FIGURE 4.7.17. Reduction of welding length on support

From reviewing the publicly available information (e.g. regulatory bodies who have licensed the EPR™ reactor or which are about to deliver Design Certification) an INPRO assessor may conclude that the EP1.1.1.3 on 'Quality of Manufacture and Construction' is satisfied.

4.7.2.4. EP1.1.1.4 Quality of Material

Mr Lignini mentioned that AREVA has some Technical Centers for conducting R&D in France (Fluid and Structural Mechanicals Components Testing, Manufacturing Materials, Metallurgy and Corrosion, Chemistry), Germany (Thermal Hydraulics and Components Testing; Materials, Corrosion, Welding technology; Chemistry, Radiochemistry; Components Qualification) and the USA (Components Testing; Chemistry, Radiochemistry; Components Qualification). It is noted that the conclusion of Quality of Material derived from the tests performed in these facilities.

R&D programs focused on degradation modes of PWR components were conducted inter alia in AREVA's technical centers. Some preventive measures were integrated in the design of the EPR™ reactor particularly regarding ageing of materials during plant operation. Through these programs, AREVA has been able to demonstrate and convince the regulators through these measurements that the selected materials are adequate for given component taking into account all susceptible modes of degradation such as ageing during operation, neutron irradiation assisted stress corrosion cracking, thermal ageing, etc. To improve the quality of materials, some measures taken are redesign of pressurizer in 20MND5, reduce impurities in RPV material (P content < 0.008% and Cu content < 0.08%), abandon the use cast duplex austenitic-ferritic stainless steels for primary piping, and manufacture primary loops with austenitic stainless steel with integral elbows and nozzles to reduce the number of welds.

Regarding environmentally assisted fatigue, field experience on existing PWR fleet worldwide showed cracking in locations where loadings were ignored or underestimate. For the design of the EPR™ reactor, AREVA conducted some critical experiments to evaluate the real concern and to establish a justification methodology. Tests were conducted with load signals representative of the actual components, and loading tests used representative surface finish (ground surfaces, etc).

Mr Lignini (AREVA) believed that from reviewing the publicly available information (e.g. regulatory bodies who have licensed the EPR™ reactor or which are about to deliver Design Certification) an INPRO assessor may conclude that the EP1.1.1.4 on 'Quality of Material' is satisfied.

4.7.2.5. EP 1.1.1.5 Redundancy of Systems

The EPR™ design of the safety systems is based on redundancy, diversity and complementarity, to resist to exceptional events and prevent damage to the surroundings.

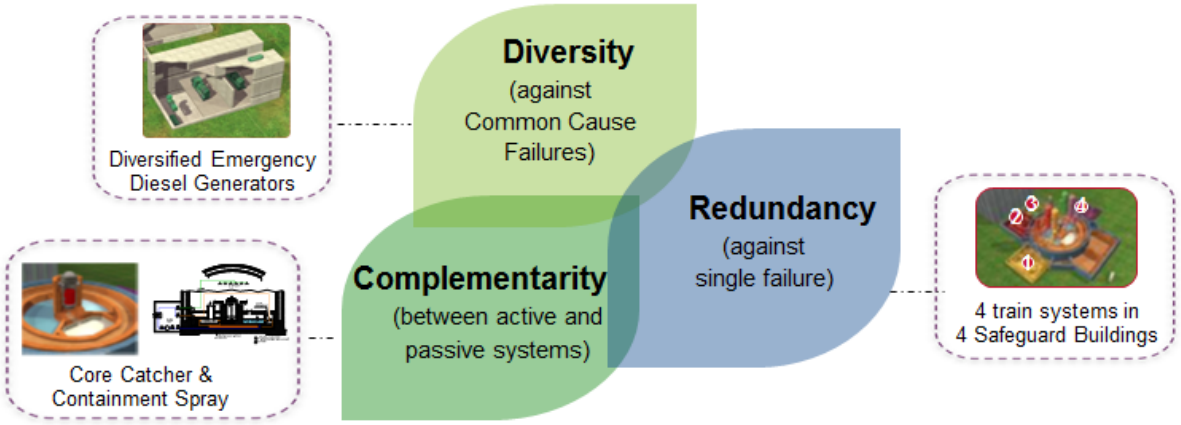


FIGURE 4.7.18. Diversity, Redundancy and Complementarity in the EPR™ design

- 4×100% capacity to allow for preventive maintenance at power (n+2 concept). The systems include MHSI, LHSI/RHR, CCWS, ESW and EFWS with 'passive headers'.
- Diversified safety system to avoid common mode failures, so every system has a diversified back-up (active and passive systems).
- External hazards through systematic physical separation of the safety system
- Clear separation between the 4 redundancies located in 4 separate buildings (location of the 4 trains) to ensure robustness against hazards (flooding, fire) and airplane crash. It was noted that train 2 and 3 are located in the airplane crash protected building, while redundancy 1 and 4 are not protected against airplane crash because they are located on the sides of reactor building.
- Reactor building, Safeguard buildings, and Fuel building on a single raft to cope with seismic and airplane crash loads

Regarding general organization of the safety systems, the objective of train ('train' designates a redundant portion of a system) separation is to preclude possible common cause failures due to equipment failures including headers or check valves. The EPR™ design has no header between the different SIS trains. However, for EBS, EFWS and CCWS headers are acceptable and useful (probabilistically) provided that isolation devices ensuring independency of trains in case of equipment failures are installed. Besides, the EPR™ has adopted a radial layout which results in separated trains and shorter piping lengths. The objective of separation into divisions ('division' designates a building of part thereof housing a train) is to prevent common cause failures due to internal hazards e.g. fire and flooding.

For design rules for safety systems, as previously discussed, four trains needed to cope with safety analysis in that 1 train is unavailable by application of the Single Failure Criterion, 1 train is unavailable due to the Preventive Maintenance, 1 train can be affected by the accident (e.g. LOCA) and the 4th train is sufficient to cope with the accident. The main systems involved are Safety Injection System SIS/RHRS, Emergency Feed water System (EFWS), Electrical Power Supply System, Component Cooling Water System (CCWS) and Essential Service Water System (ESWS). The trains are installed in 4 separate divisions providing efficient protection against internal and external hazards which could lead to common model failure, e.g. fire, flooding, airplane crash. With these redundancies, the EPR™ could be possible to perform the preventive maintenance during power operation.

Regarding electrical power supply system, EPR™ has 4 electrical diversified trains each of which contains a 10kV emergency busbar, an emergency diesel generator and DC busbars with batteries to take over such accidents. In addition, two diversified small diesel generators (690V) for trains 1 and 4 are used in case of failure of the 4 EDGs i.e. a SBO.

Mr Lignini (AREVA) had the opinion that from reviewing the publicly available information (e.g. regulatory bodies who have licensed the EPR™ reactor or which are about to deliver Design Certification) an INPRO assessor may conclude that the EP1.1.1.5 on 'Redundancy of Systems' is satisfied.

4.7.2.6. Proposed EP on Robustness against External Hazards

Mr Lignini discussed robustness against external hazards as a supplement to improve the INPRO Methodology. He drew attention to the safety objectives against external hazards which are the prevention of unacceptable radiological releases and the plant operation to the safe shutdown conditions. In the design of EPR™, the external hazards under consideration include earthquake, airplane crash, explosion pressure wave, site flooding, extreme weather conditions, extreme winds, rain, snow and ice formation, drought, hazards associated with the industrial activities or natural environment and with transport routes (missiles, offsite fire, movement of toxic or corrosive gases), and lightning and electromagnetic interferences.

Two examples were provided, namely the DBE and explosion pressure wave. The EPR™ has a DBE of 0.25 g at 33 Hz taking into account the quality of the soil. The standard Explosion Pressure Wave (EPW) is applied in the design of the following buildings, many of which house safety systems: Reactor building (inner and outer containment), Fuel building, Safeguard buildings including the main steam / feed water valve compartments, Nuclear auxiliary building, Radioactive waste processing building, Diesel buildings, Service water pump buildings (site specific) and connecting ducts to and from service water pump buildings (site specific). The stability in case of explosion pressure wave is ensured.

When assessing external hazards, AREVA does not systematically take into account single failure and preventive maintenance rather looking at the factors on case by case basis to see the trends. It should be noted that they are not systematic feature integrated into the safety analysis.

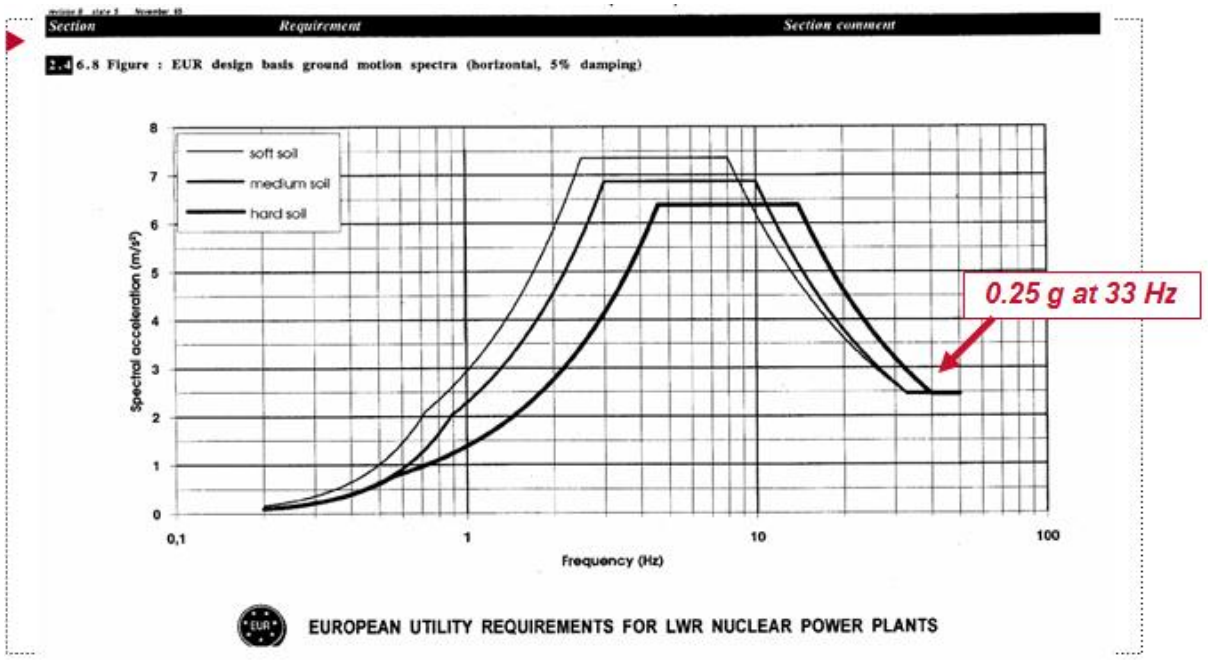


FIGURE 4.7.19. Design Basis Earthquake (DBE) for EPR™

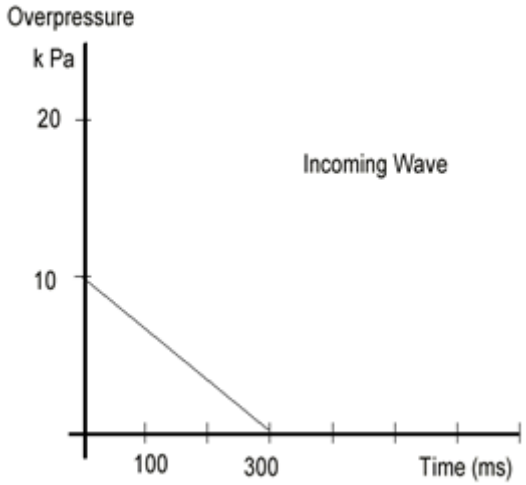


FIGURE 4.7.20. Explosion Pressure Wave for EPR™

- Airplane Crash Protection

  The approach for protection against aircraft crash is deterministic. The protection is achieved by designing safety-classified buildings called the airplane crash (APC) shell for aircraft crash loads and by geographical separation of safety-classified systems. Usually 3 load cases are considered associated with different types of planes, i.e. light aircraft weighting less than 5.7 tons, military aircraft and commercial aircraft. For the EPR™ design, the military and commercial aircrafts have been taken into account. The requirements for impact of the crash are stability of the buildings, surviving vibration or shock caused on the safety related equipment, no perforation of the structures, no scabbing of the structures and kerosene fire, i.e. external fire from the aircraft could not penetrate the protected building.

  There are two ways of prevention and mitigation:
  - Buildings protected by design include the Reactor building, Fuel building, Safeguard buildings 2 and 3. The protection requirement is provided by the APC shell, consisting of a bunker designed to withstand aircraft crash load scenarios. It is made of reinforced concrete. A LOCA inside the reactor Building must not be induced (and a fortiori no severe accident) so containment isolation is not required.
  - Buildings protected by geographical separation include the Safeguard Buildings 1 and 4, the Diesel Buildings and the Service Water Pump Buildings. At least one redundancy stays available after the aggression. The loss of diesel generators due to fire induced fumes can be ruled out due to the 2 by 2 arrangement on opposite sides of the reactor building and the associated distance of the air intakes.

  Protection against airplane crash is made with provisions for an impacting large passenger or military aircraft. A thick shell of highly reinforced concrete protects the inner walls and the inner structures from the direct impact and resulting vibrations.

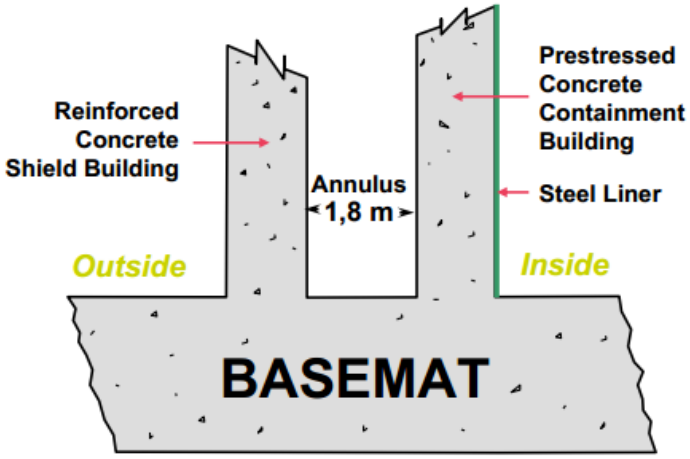


FIGURE 4.7.21. The Airplane Crash (APC) Shell

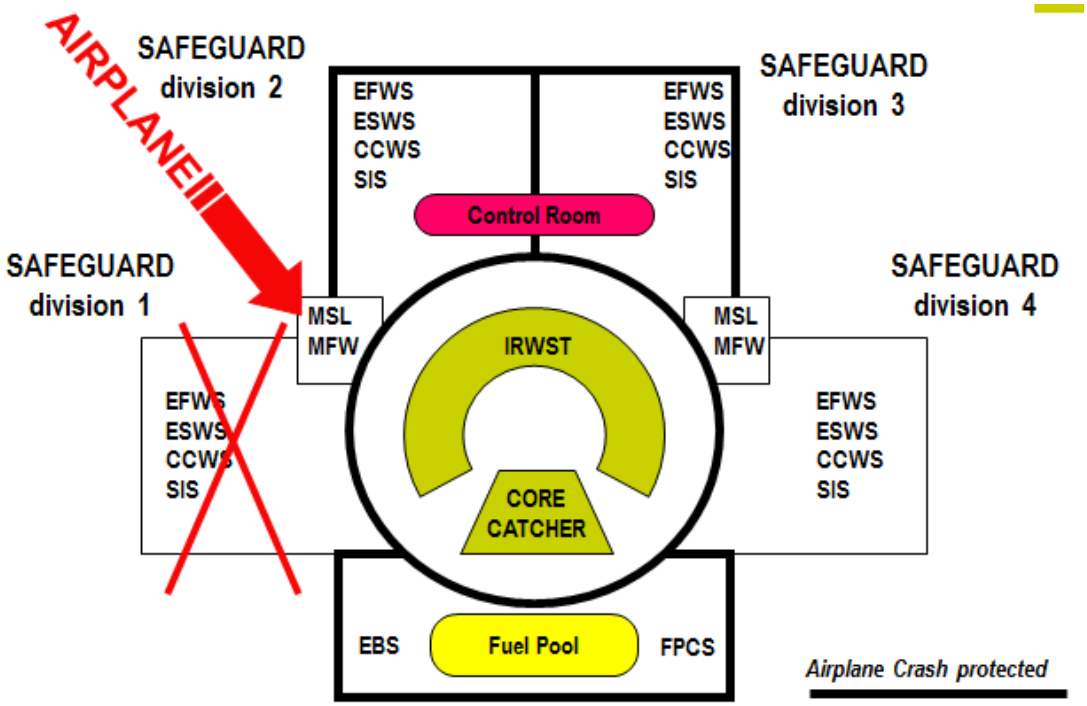
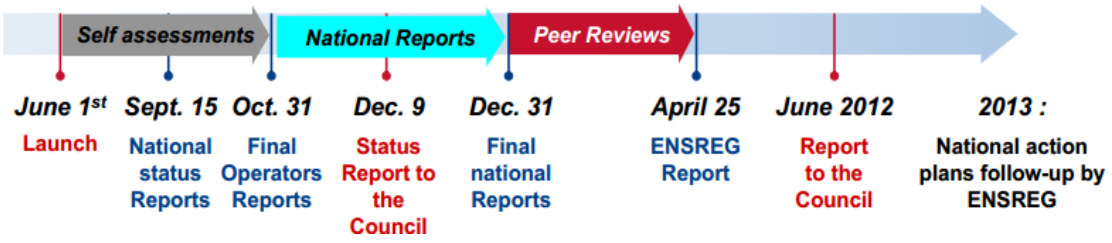


FIGURE 4.7.22. Protection of Buildings Housing Safety System Against Airplane Crash

- Robustness of the EPR™ reactor to a Fukushima like event

  Mr Lignini briefly explained the EU stress tests imposed by nuclear regulators as common approach of all European Nuclear Safety Authorities. The specifications were prepared by WENRA (Western European Nuclear Regulators Association) and adopted by ENSREG (European Nuclear Safety Regulators Group and Commission). The tests involved 14 countries and 143 reactors with exceptional mobilization and also neighbouring countries. The process is transparent comprising 3 steps: operators, national authorities, and European peer reviews. The implementation of the ENSREG recommendations joins in the continuity of the dynamics of safety in Europe.



The ENSREG specifications comprise performance of a complementary assessment based on deterministic approach, irrespective of the probability of the loss of line of defense. A sequential loss of the lines of defense and a long duration of events are considered, taking into account the entire affected plant (reactor, fuel storage). The objective is to identify weak points and cliff-edge effects, for each of the following extreme situations:

- Extreme earthquake, flooding (consequential or linked event should be considered such as fire, load drop for earthquake, bad weather conditions for flooding
- Total loss of power sources, total loss of ultimate heat sink, combination of both
- Severe accident, notably in case of the two previous situations

A particular attention should be paid to confinement failure risk and provisions to cope with weak points and cliff-edge effects, notably to improve the grace period before cliff-edge effects. These provisions should be considered as hardened equipment.

The resistance of the EPR™ reactor to a Fukushima like event was assessed by analysing the consequences of the external hazards beyond those taken into account in the design basis. As no guidance existed to define which "beyond-design basis external events" are to be considered and what criteria shall be met, the Fukushima load cases were taken as input data: earthquake of ~0.5 to 0.6 g Peak Ground Acceleration (PGA) at site and tsunami/flooding with wave height of 4 m above the platform level. The analysis shows that the design would withstand such an event and that a Fukushima scenario would result in a combination of a loss of offsite power (LOOP) and a loss of ultimate heat sink (LUHS).

- Resistance to a beyond design earthquake

  The 9.0 magnitude earthquake resulted in a ~0.5 to 0.6 g PGA at the Fukushima site. The EPR™ reactor is designed to resist to a 0.25 g PGA earthquake at the site. Seismic Margin Assessments (EPR UK) have been performed showing the capability of the design to withstand a 0.6 g PGA earthquake. A 0.5-0.6 g PGA earthquake at site would result in a total loss of offsite power (LOOP) which is overcome by automatic start-up 4 Emergency Diesel Generators (EDG). In addition, the Emergency Feed water System would automatically start up to feed the SGs in case of loss of the main feed water. Therefore, the decay heat would be safely removed.

- Resistance to beyond design flooding

  The earthquake at Fukushima triggered a ~14m high tsunami which flooded the plant up to 3-4 m water on the platform. The analysis of the impact of such a tsunami at an EPR site shows that the Safeguard Buildings, the Diesel Buildings, the Pumping Station would resist to a 4 m flooding of the site, including the dynamic effect of the wave. The leak-tightness is ensured since these buildings and their doors are designed to resist to external hazards (earthquake, external pressure wave) and they comply with Physical Plant Protection requirements which provide design margins. The safety systems needed to remove the decay heat remain available: EDGs, SBO-DGs, I&C and batteries, Emergency Feed water System, and others.

  However, a 3-4 m platform flooding could result in the loss of the Ultimate Heat Sink (LUHS) in case of the loss of the cooling chain (possibility of debris clogging the water intake). The diverse cooling chain, however, might not be affected thanks to site dependency, e.g. location of reservoir on the top of the hill above the Flamanville reactor

For mitigation of LOOP and LUHS, for reactor cooing the decay heat is removed by the SGs (EFWS pumps, SBO-DGs) and for the fuel pool cooling the 3rd FPCS/SRU train is used. A large plant autonomy is thus ensured:

- Phase 1 “Short Term”: 24 h without offsite means support nor on-site mobile means and no actions outside protected buildings
- Phase 2 “Mid Term”: 7 days with onsite fixed and mobile means support (for water reserve, fuel oil and lube oil inventory)
- Phase 3 “Long Term”: unlimited with offsite support given the connections for mobile means: connections for water tank refilling, for fuel and lube oil tank refilling, for mobile power sources to take over SBO operation.

Hence for EPR™ design, there are more than 7 days of autonomy.

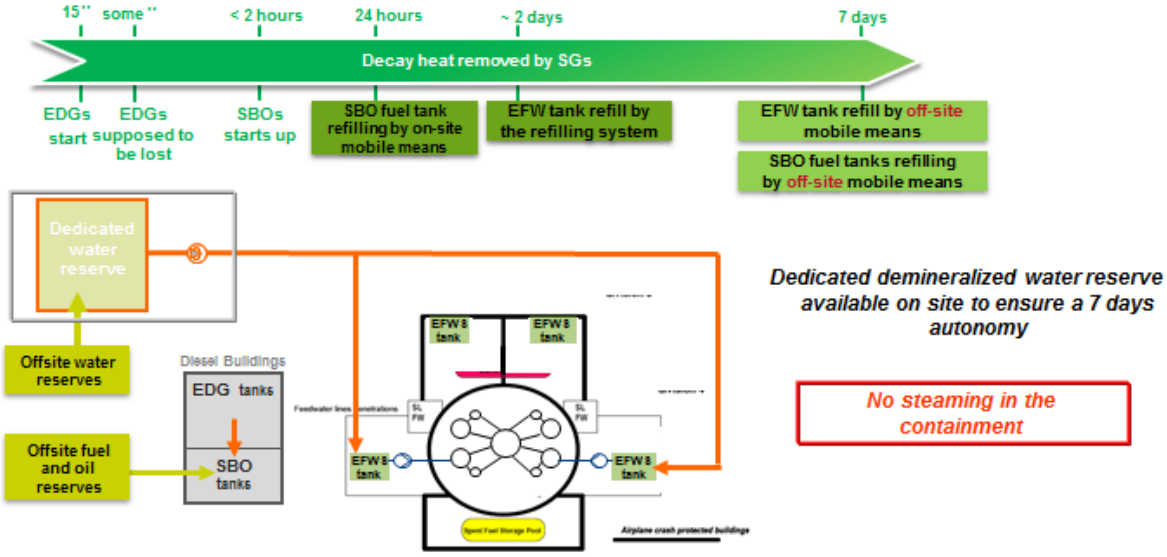


FIGURE 4.7.23. Mitigation of LOOP and LUHS for the EPR™ design

The initial design basis features of the EPR has been proved to be resistant to extreme events as a result of (i) deterministic means to reduce probability of core melt events and to mitigate severe accidents in all plant states, (ii) availability of multiple lines of defence, and (iii) design of the EPR buildings against severe hazards and malevolent acts providing a high degree of resistance to beyond design basis hazards such as those that occurred in Fukushima. The ASN (French regulatory) Complementary Safety Assessments conclusions highlight the EPR™ specific safety features and design to manage severe accidents stating that, “the safety objectives and the strengthened design of this type of reactor already offer improved protection against severe accidents compared to any existing operating nuclear power plant”. In addition, the EPR design in particular takes account of and incorporates measures to deal with the possibility of accidents with a core melt and combinations of hazards. All the systems necessary for the management of accidental situations, even severe, are designed to remain operational for an earthquake or a flood as defined in the baseline safety requirements.

Overall, Mr Lignini (AREVA) considered that from reviewing the publicly available information (e.g. regulatory bodies who have licensed the EPR™ reactor or which are about to deliver Design Certification) an INPRO assessor may conclude that the criterion CR1.1.1 on ‘Robustness of Design’ is met.

4.7.2.7. CR1.4.1 Major Release into Containment

Major release into the containment would result from severe core degradation and one has to look at the core damage frequency (CDF). The Probabilistic Safety Assessment (PSA) Level 1 (CDF) and 2 (LERF) were performed at the design stage to supplement deterministic safety assessment and confirm soundness of the design. This was performed for the standard EPR™ design with a generic site condition. The PSA takes into account internal events, internal hazard and risk analysis of fuel damage in the SFP. The results are the CDF for $<10^{-7}$/RY which comprise half for internal events and roughly a quarter each for internal and external hazards. The large early release frequency is much below 10^{-7}/RY. Note that the values of CDF and LERF are at least an order of magnitude less than existing designs.

TABLE. Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) of the EPR design

Plant condition	EUR design target	EPR evaluation
Core damage frequency : •all plant states * •all type of initiators	$<10^{-5}$ /r.y	$\sim 10^{-7}$ /r.y
Large Early release frequency	$<10^{-6}$ /r.y	$<10^{-7}$ /r.y

*particular focus on accidental sequences during shut down states

Mr Lignini (AREVA) believed that from reviewing the publicly available information (e.g. regulatory bodies who have licensed the EPR™ reactor or which are about to deliver Design Certification) an INPRO assessor may conclude that the CR1.4.1 on 'Major release into the containment' is met.

4.7.2.8. CR1.4.2 Processes & CR1.4.3 In-Plant Severe Accident Management

Severe accident can be defined as event beyond the DBC & DEC-A accidents, leading to core damage/melt (loss of the 1st barrier) and possibly to RPV failure (loss of the 2nd barrier). Mitigation approach aims at ensuring containment integrity (3rd barrier). Energetic phenomena that may lead to an early containment failure should be practically eliminated, namely (i) high pressure cold melt sequences, (ii) steam explosion and (iii) hydrogen detonation. Mr Lignini pointed out at the RPV melt-through experiment at elevated pressured in KTH Sweden (FOREVER experiment) showing a stream of corium ejected at high speed from the RPV at such impact on the concrete wall of the containment. A provision is needed to ensure that molten corium is not to be ejected at high speed. Commitments made to ensure the confinement integrity for any other sequences following a core melting, including those leading to the RPV failure at low pressure: (i) stabilization of molten corium, (ii) containment heat removal and (iii) limitation of radioactive releases, i.e. protective measures limited in area and time and no long term off-site contamination.

The typical sequences of event in case of severe accidents:

- Failure of water inventory and the core is uncovered and heats up
- In-vessel core meltdown: (i) cladding (Zr) oxidation, with H_2 production, transport, release and combustion, and (ii) corium relocation in RPV bottom end
- RPV bottom end failure: (i) reactor vessel wall attack / melt-through, (ii) if vessel failure at high pressure, core debris and coolant injection, and (iii) interaction of corium / water, with possible steam explosion
- Ex-vessel corium spreading: (i) interaction between corium-concrete, (ii) effective corium cooling leading to solidification, (iii) containment heat up and over-pressurization with failure risk, and (iv) fission product release if confinement leaks.

So it is important to ensure the basemat will be able to resist, and spread the corium to be able to cool / remove the heat to stabilize it.

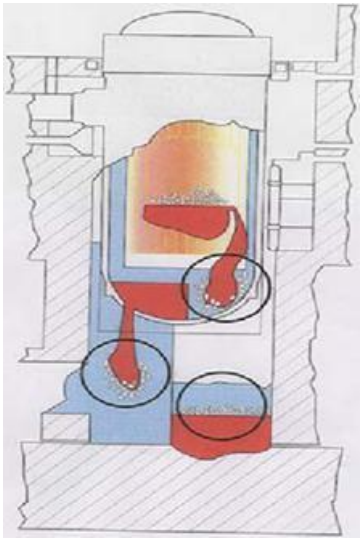


FIGURE 4.7.24. Sequences of event following a core melting

The practical elimination of energetic phenomena in the EPR™ design was further elaborated:

- Practical elimination of high-pressure core melt

  Should RPV fail at high pressure, the core debris and coolant ejection may lead to direct contamination heating. The EPR™ has in place a primary depressurization system that prevents high-pressure core melt sequences above 20 bar. The dedicated depressurization valves supplement the 3 Pressurizer safety valves and provide a safe and reliable Reactor Coolant System (RCS) depressurization. These redundancy and diversification assure high probability of successful operation.

- Practical elimination of steam explosion

  Steam explosion is a potentially destructive reaction between the molten corium and water. The design is that for in vessel steam explosion, there will be no risk of containment failure, i.e. the reaction is confined with the RPV. To avoid steam explosion outside the reactor vessel, the EPR™ design provisions ensure dry reactor pit and corium spreading area prior to corium flooding, thus there will be no water in the reactor pit at that time when the reactor pressure vessel fails.

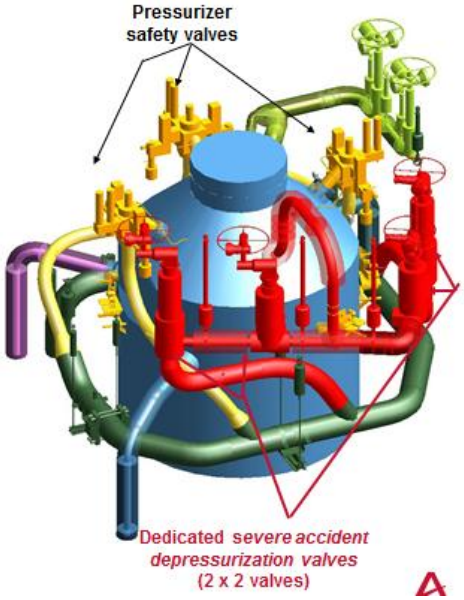


FIGURE 4.7.25. Depressurization system to prevent high-pressure core melt sequences

- Practical elimination of H_2 detonation

  H_2 detonation results from the oxidation of Zr fuel cladding at high temperature. The amount of hydrogen that can be produced from the total oxidation of Zr fuel cladding is 1684 kg H_2, which is equal to a concentration of >20%. The global detonation can be avoided as long as the average global H_2 concentration within containment is less than 10% by volume. In order to do this, the EPR™ is equipped with passive device which allows mixing and good spreading of diffusion of hydrogen into the whole volume of containment. Illustrations were provided on hydrogen mixing dampers installed at the wall between IRWST and annular rooms, and 'open' compartments within the large volume of containment (80,000 m^3). Also, there is no automatic early containment spray to avoid steam condensation.

  In addition, the reactor is fitted with Passive Autocatalytic Recombiners (PARs) so that the H_2 concentration remains low and detonation of fast deflagration is not possible. The calculation shows that the maximum global containment pressure in the most limiting scenarios with H_2 deflagration is 4.1 to 6.3 bars abs (containment integrity is up to 6.5 bar abs).

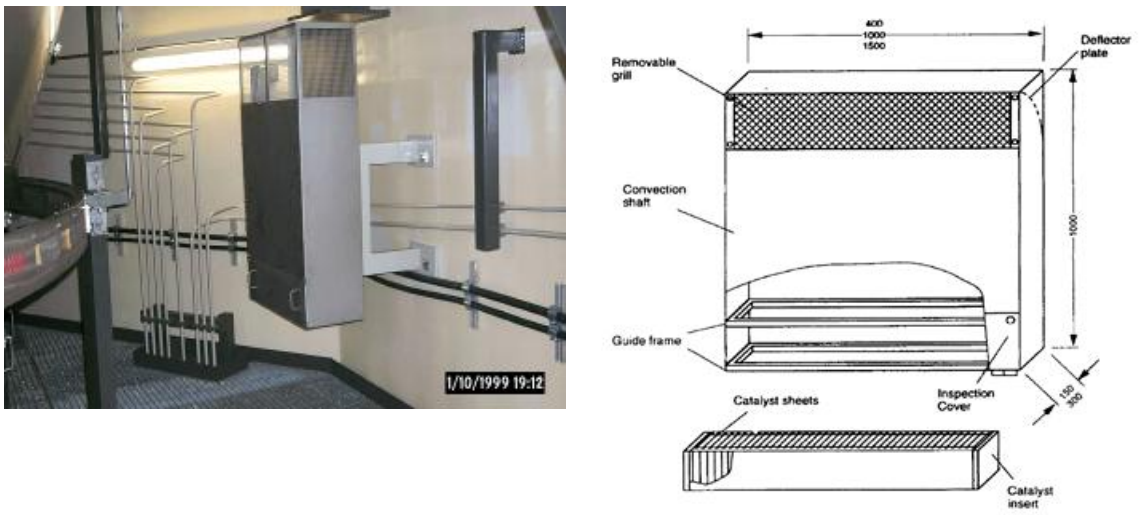


FIGURE 4.7.26. Passive Autocatalytic Recombiners (PARs)

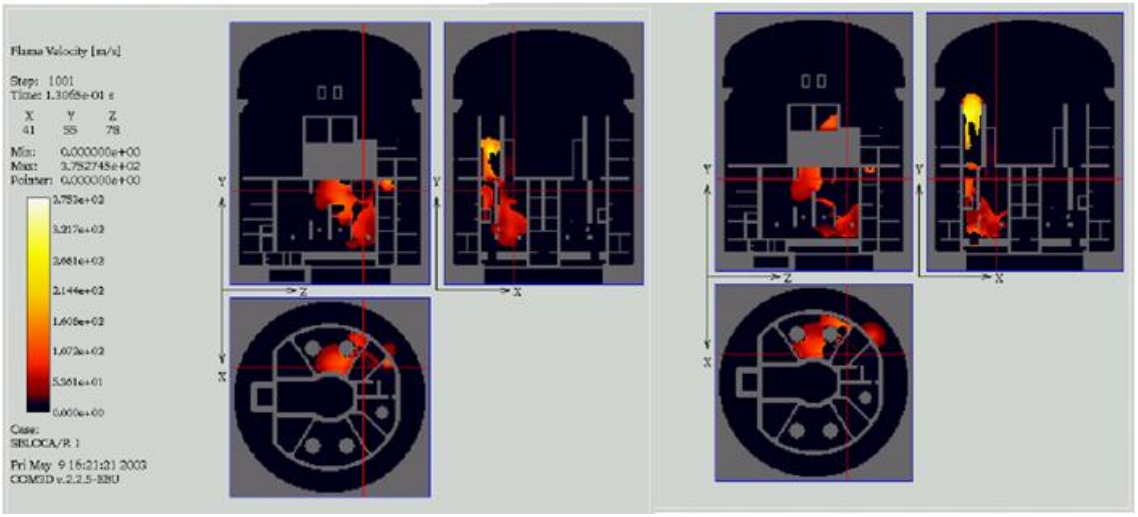


FIGURE 4.7.27. Analysis of H_2 combustion. Acceleration in the SG tower: left 0.13 s and right 0.18 s after ignition. In the dome, deceleration with low impact on the containment shell.

Mr Lignini briefly mentioned the sequences of the core melt retention. Should the RPV fails, the strategy is to prevent corium-basemat interaction to avoid significant releases and durable contamination of sub-soil and underground water. The corium would accumulate and be temporarily retained in the reactor pit. Metal gate and sacrificial concrete in the reactor pit would be consumed thus delaying the pouring of corium into the core catcher spreading area or core catcher. Once spread in the in the core catcher, the spreading area would be flooded and cooled by the IRWST water. All these stages (retention, spreading, flooding, cooling) are fully passive.

Mr Lignini (AREVA) explained that the spreading area is located away from the reactor pit in order to prevent potential damage to the spreading area resulting from the failure of the RPV. When the RPV fails, the corium is going to accumulate in the reactor pit, with the target to facility spreading in one event and to lower corium temperature and viscosity (sacrificial concrete). After the sacrificial material is consumed, the spreading may occur. AREVA has performed corium spreading tests at the French CEA VULCANO to validate this sequence using oxide fuel mixed with cladding material. The flooding and cooling is going to be passive, i.e. when the corium arrives at the spreading area, some fuse will melt which then triggers the opening of valves which allows overflow of water by gravity from the IRSWT. The water will first start to circulate underneath the spreading area, and after about 5 minutes, the level will increase and the water will start to pour from the top onto the corium. Thus cooling is achieved from the bottom and the side. At the equilibrium water level, cooling is established also for debris potentially remaining within the transfer channel and lower pit. Formation of a saturated water pool in the spreading compartment would take place with steam release into the containment. With these sequences, the EPR™ can have at least a grace period of 12 hours prior.

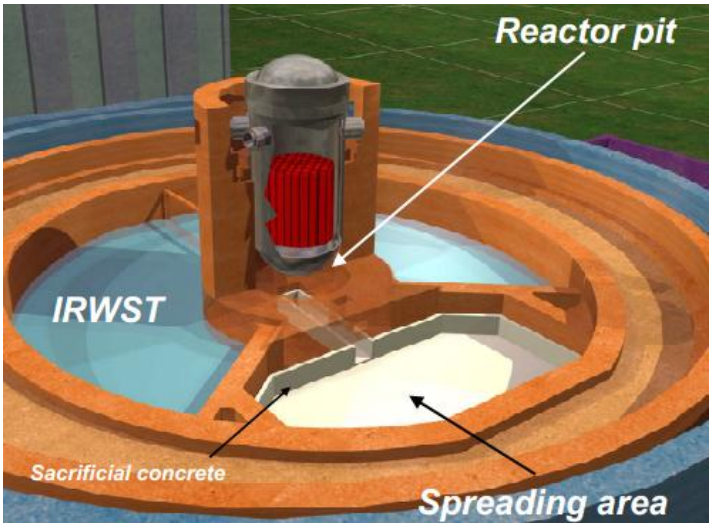


FIGURE 4.7.28. Core melt retention

For a long term stabilization of core melt, following a grace period of 12 hours the Containment Heat Removal System (CHRS) will be initiated. The CHRS can operate as a spray mode or an active mode to limit the pressure inside the containment hence there is no need for filtered venting. This has been accepted by several regulatory bodies though some other regulators may still impose on the use of filtered venting. Note that possible switch between passive and active core catcher cooling is not required as the formation of a sub-cooled water pool above the melt avoids the need for further containment spraying after some days (atmospheric pressure is reached). Effective cooling process would be achieved due to large surface-to-volume ratio of the melt, without interaction between met and structural concrete. Therefore, a stable state of the melt is reached within hours and total solidification of corium is reached within days.

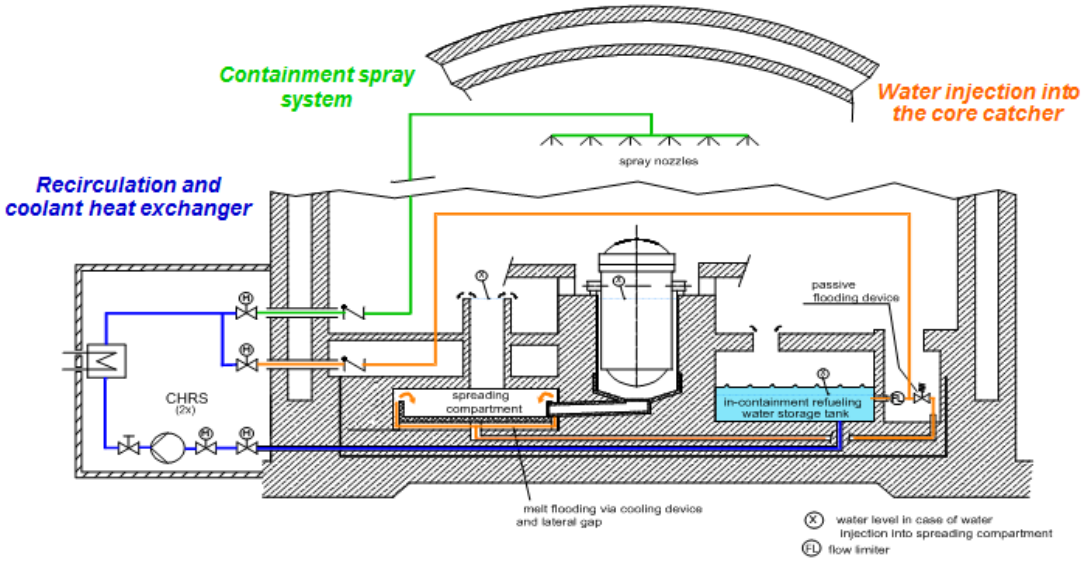


FIGURE 4.7.29. Melt cooling and containment heat removal system for long term phase, beyond 12 hours grace period

Besides, for the safety, there is double wall containment which is composed of outer wall and inner wall. The outer wall is a reinforced concrete shell, resistant to external hazards, while the inner wall is a pre-stressed concrete shell with a steel liner. The design leak-rate at the design pressure for a 24-hour period is less than 0.5% by volume at 5.5 bar. All leakages at the level of penetration are collected in the annulus and a reduction of radioactive aerosols is achieved by filtration prior to the release via the stack.

Mr Lignini considered that from reviewing the publicly available information (e.g. regulatory bodies who have licensed the EPR™ reactor or which are about to deliver Design Certification) an INPRO assessor may

conclude that the acceptance limits for indicators IN1.4.2 (Criterion 'Processes') and IN1.4.3 (Criterion 'Accident Management') are met.

4.7.2.9. CR 3.1.1 Occupational Dose

Mr Lignini (AREVA) focused on aspects as pointed out in the INPRO Methodology to achieve reduced occupational dose, which are through minimization of source terms, layout features and maintenance friendly design.

- Dose targets for the EPR™ reactor
  The collective dose target of the design is 0.5 man-Sv per year averaged over plant life. Other targets are covered and demonstrated in the safety analysis report:
  - Dose limit to occupationally exposed persons is 20 mSv per/ effective dose for normal operations. The design is aimed to meet ALARA principles, i.e. individual dose restraint of 5 mSv per year.
  - Dose limit for the public is 1 mSv per year effective dose in the vicinity of the power plant
  - Dose rate limit for areas outside of the radiation controlled areas: All radioactive systems and components are contained in a controlled area: external dose rate <2.5 μSv/h (inside buildings) and external dose rate <1 μSv/h (outside walls; owner protected area).
  - Local neutron dose rate is < 2.5 μSv/h (reactor building)

  The graph below describes how the design targets were established, showing the effective dose measured in some PWRs in various countries. AREVA may have different targets depending on the projects and national circumstances.

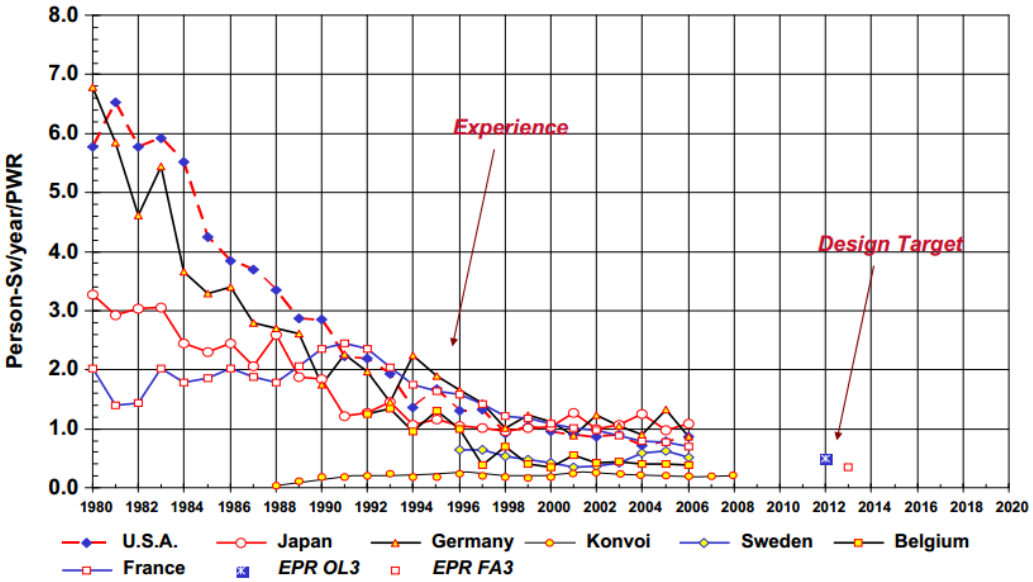


FIGURE 4.7.20. Collective dose evolution feedback on PWR

- Layout features adopted for the EPR™ reactor

  The table below provides the evolution of the layout of reactor starting from the first generation of PWR as a good practical measure resulting from operating experience.

TABLE 4.7.3. Improvements of Building Design

Plant Group, 1300 MW Building	1st Generation	2nd Generation	Pre-Konvoi	EPR / Konvoi / N4
	- SG separated from RCP motor - loop valve compartment - loop-specific stairwell - separated access paths - separated transducer compartment - HP cooler separated from recuperative heat exchanger	- separation of hot leg of loop from cold leg	- residual heat removal pumps separated from valves	- tanks/vessels usually set separately (e.g. liquid waste, coolant) (Konvoi and EPR) - shielded safety injection pump - improved shielding for HP cooler, recuperative heat exchangers, fuel pool and residual heat removal pumps (accessibility after accidents) - additional pipe ducts
	- High-radioactive tanks/vessels set up separately (e.g. Ion-exchanger) - Medium-radioactive tanks/vessels set up in pairs (e.g. coolant tanks) - Separation of medium low-level waste	- separate access for high radiation tanks (shield brick openings	- avoidance of shielding block openings (labyrinths, shield doors)	- separate access paths of many levels (avoidance of ladders) - no shield block walls - widened transport and construction routes - stationary platforms in the auxiliary building (large tanks/vessels)
	- common access to tank rooms			
	- mostly ladders for internal stairs			
		- fixed installed platforms at SGs and loops	- increase of inspection platforms in the auxiliary building	

- Examples of source term reduction for the EPR™ reactor

  Mr Lignini mentioned quality material and limitation of impurities as two examples, both of which had already been discussed earlier. He added another example, i.e. a reduction of Stellite™, which is a hardened material that may generate activation of cobalt. With the corrosion phenomena, the corrosion product may spread into the RCS and generate dose. The EPR™ reactor has achieved the minimization of Stellite™ compared to the French P4 by a factor of 2, hence reduction of the cobalt content which leads to the reduction of staff radiation exposure. However, it was noted that the amount of Stellite™ in German KONVOI is already very low, less than that of the EPR™. Mr Lignini indicated that EPR™ is a compromise between French and German reactors, keeping in mind that some internal configurations from the French reactors are much higher in terms of the amount of Stellite™.

TABLE 4.7.4. Amount of Stellite™ in EPR™ compared to French P4 and German KONVOI

STELLITE™ SURFACES (m²)			
COMPONENT	P4	EPR™	KONVOI – Co-60 Model (input SIEMENS EPR™ basic Design)
CRDMs	1.8	2.03	1.22 – 1.6
RPV Internals and Core Instrumentation	0.56	0.49	0.02 – 0.06
Main Coolant Pumps	2.5	1.4	0 / 0.3 - 0
Valves	2.4	0.03	0.56 – 0.97
TOTAL	7.3	3.95	1.8 / 2.1 – 2.6

- Some aspects of operation and maintenance for the EPR™ reactor

  The EPR™ boasts short outages for high availability. The reduction of outage duration owes to the preventive maintenance on safety trains (4×100% safety trains), a large set-down area to prepare outage work, and fast cool-down of the core. The outage time for normal refuelling outage would be about 16

days, refuelling only outage without maintenance would be about 11 days and the ten-year outage would be about 40 days for many periodic tests.

The high reliability of the equipment is due to proven evolutionary components based on hundreds of years of reactor operations, hence improved reliability. The unscheduled unavailability rate is lower than 5 days / year (3 days / year for NI) is 1.4%/RY. The design has the capability to cope with various grid failure situations and loss of equipment without reactor trip, with <1 unplanned reactor trip/year. The high reliability of equipment given the robustness of the design would allow high availability of 92%+.

In order to achieve short outages, fuel unloading could start early (end of 3rd day) because of the very fast cool down from hot shutdown to cold shutdown, no oxygenation phase of the primary circuit, and adequate size of the Fuel Pool Cooling System (FPCS). It could also perform early and limited maintenance on the safety trains by provisions of maintenance works during power operation, maintenance of one safety train during fuel unloading and simultaneous maintenance of two safety trains during SG inspection and control as soon as the core is unloaded. Besides, the design of the SG has been improved to facilitate and to shorten the time needed for inspection. Furthermore, regarding the restart system, it is possible for a fast start up from vessel closing to grid connection in particular by having periodical tests which can be performed by being not on the critical path and the use of the aeroball systems allows quick flux mapping.

Regarding the improvements of the building design, there is the four-train concept for preventive maintenance during operation with easier maintainability and higher availability. Also there is optimized layout with clear geographical separation between divisions and large set down areas to facilitate maintenance operation in the reactor building. It was noted that from the beginning of the design phase, maintenance was taken into account, in particular the need for area where the equipment can be deposited after being dismantled for maintenance and also there are some handling devices which are being incorporated and which are fixed in order to facilitate dismantling and handling of the equipment which have to be maintained.

The reactor building is partly accessible during power operation, which allows operator to remove preparatory and maintenance activities from the critical path, i.e. 7 days before the outage and 3 days after the start up. The main activities can be in the critical path, for examples, requalification of the polar crane, maintenance and requalification of the handling equipment of the vessel head and the internal structures of the reactor pressure vessel, tests of the refuelling machine, and leak tightness tests on some containment penetrations (service compressed air, demineralized water, and primary waste systems). All these contribute to reduce the duration of the outrages because these activities should be conducted during power operation.

The redundancy of Level 4 in safety system allows preventive maintenance of safety systems and operational systems during operation, for example:

- Component Cooling Water System and Essential Service Water System – scheduled so as to concurrently perform maintenance of the safety system associated trains (SIS, CHRS, etc)
- Electrical systems – performed concurrently during preventive maintenance of the safety system associated train (partially)
- Batteries
- Main feedwater pumps (4 × 33% motor-driven pumps)
- Start-up and Shutdown System pump
- Containment Heat Removal System

Improvements of the system and component design of the EPR™ reactor in comparison with the French N4 and German KONVOI are given below.

TABLE 4.7.5. Essential improvements of system and component design of the EPR™ reactor

Plant Group, 1300 MW Systems/ Components	1st Generation	2nd Generation	Pre-Konvoi	EPR / Konvoi / N4
	Ease of maintenance and operation - removable shaft for changing gaskets on reactor coolant pumps - machines for changing filters (gradually improved) - canned motor pumps for some systems - mechanical seals as exchangeable integral unit - sufficient number of manways of adequate size (esp.: SGs)			
			- elimination of welds to be inspected (e.g. SG cone, pressurizer)	- considerable reduction of welds (RCL, SG)
	- thermal mat insulation with sheet metal covers with screw connector	- thermal insulation in cassette form in area of weld which can be assembled and disassembled quickly (quick actuating closure) (initially soft design with metal foil; later hard design with sheet metal)		
		- gradual improvement of auxiliary extensions for pumps and valves		- if possible permanent lifting for exchanging pumps and valves
		Improved possibilities of decontamination		
		- setdown area for mobile equipment in containment (esp. for RCP) - connections to the tanks of the auxiliary building	- in part, permanently installed supply lines	- permanently installed equipment for RCP decontamination and permanently installed auxiliary equipment (Konvoi and EPR) - permanently installed decontamination system for components in the auxiliary building and waste building (waste building EPR and N4)

To achieve low collective dose, the design is optimized for radiation protection with personnel exposure target < 0.5 man Sv/year:

- Components
  - Low frequency and small effort for maintenance work
  - Equipped with quickly removable and reusable thermal insulation
- Selection of material
  - Activated corrosion products are kept low
  - Reduction of cobalt base alloys to a minimum
- Component layout and accessibility
  - Easily testable regarding operability
  - Easily replaceable, if necessary
- Maintenance and In-Service inspection
  - Tanks, vessels, and heat exchangers are designed to avoid radioactive deposits or at least enable easy removal
  - Adequate access and space provided for inspection and maintenance of components
  - Remotely controlled in-service inspection for primary components
  - Hot/cold separation of rooms and access ways

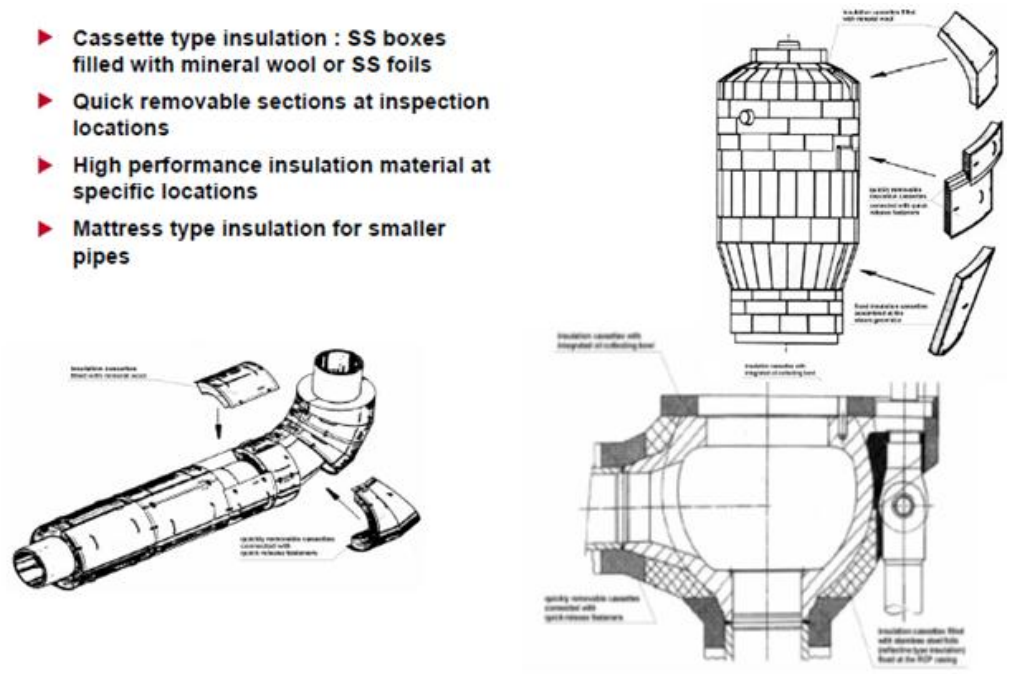


FIGURE 4.7.21. Insulation of equipment – which can be easily removed and reused

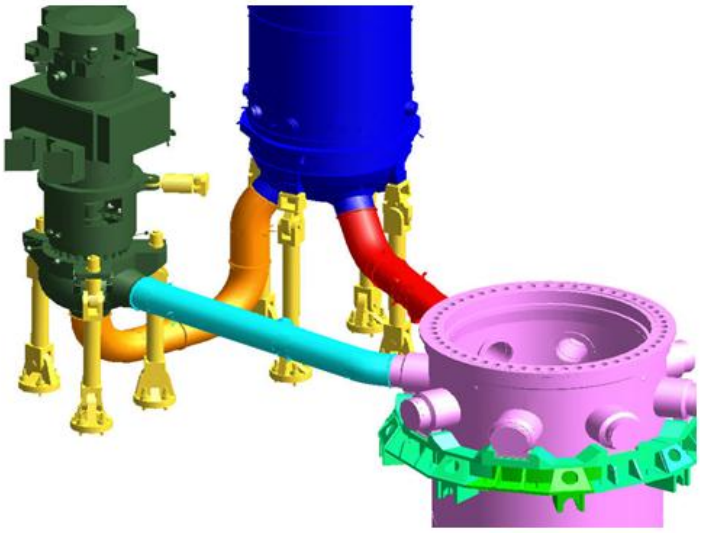


FIGURE 4.7.22. Example of main coolant line (MCL) without antiwhip supports

Mr Lignini indicated the design does not use the antiwhip supports allowing better access for inspection, i.e. less congestion, no gap measurement/follow and no risk of thermal interference. This is already accepted by regulators in UK and France, but not in Finland at least for Olkiluoto 3 though shall be applicable for future EPRs in Finland.

On occupational dose, AREVA looks not only into the normal operation but also accident conditions. Under accident conditions, the general tasks of radiation protection are to support post-accident management, make possible the repair of failed components, and provide accessibility to radioactivity sampling points (information gathering). The scope of investigation includes:

- Fuel building - compartments of post-accident atmosphere sampling system (location of sampling box)
- Fuel building – compartments samplers of aerosol and iodine radioactivity
- Nuclear auxiliary building – laboratory
- Safeguard building 1 to 4 - residual heat removal system compartments
- Safeguard building 1/4 - containment heat removal system compartments (CHRS)
- Safeguard building 2/3 - main control room

The individual effective dose for post-accident interventions is <50 mSv. For practical reasons the following design criteria are defined: maximum dose rate for local operations 1 mSv/h and maximum dose rate for routes 10 mSv/h. Individual dose for access to and from a local plant area <1 mSv of external radiation. For sampling, a maximum collective dose is less than 10 mSv with first sampling not earlier than 24 hours after reactor scram. Repair of pumps is done not earlier than 30 days after start of accident, and the maximum dose rate in the vicinity of a failed pump 0.5 mSv/h.

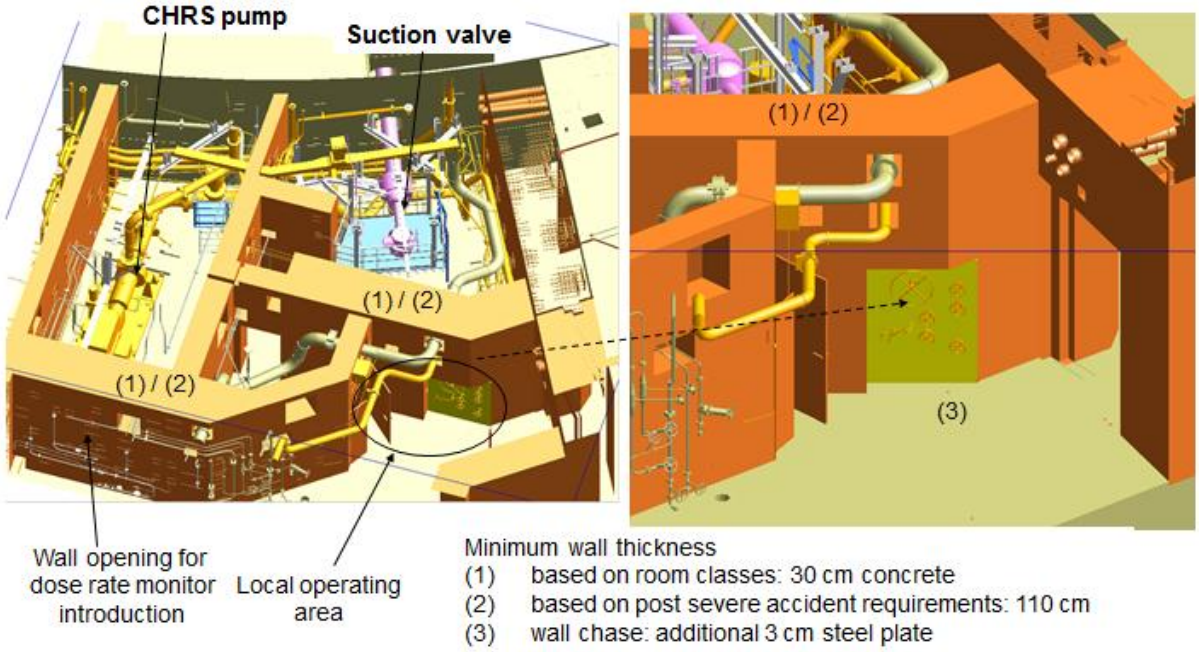


FIGURE 4.7.23. Containment Heat Removal System (CHRS) shielding

- Estimate of collective dose for the EPR™ reactor
  Mr Lignini took Olkiluoto-3 as an example. The following doses were estimated for the reactor building:
  - Outage with refueling only 90 man-mSv
  - Outage with partial maintenance 405 man-mSv
  - Outage with extended maintenance 745 man-mSv

  Taking an expected succession of outage types, the mean value per EPR is 280 man-mSv (12 months sequence) or 235 man-mSv (24 months sequence). The dose contribution from other NI buildings: 100 man-mSv, and the dose contribution due to activities during normal operation is 30 man-mSv. The collctive dose that can be achieved in EPR reactor, for the Finnish Olkiluoto is 0.41 man-Sv/year based on 12 months sequence, which is below the target. All estimates have been made at the end of the life when the source terms and activation of the facility are the highest.

TABLE 4.7.6. Main dose contributors inside reactor building

Task	Collective Exposure Assessment (man.mSv/y)	Rate (%)	
1-Reactor	36.11	12.84	Opening, closure, RPV internals & fuel handling
2-SG&PZR	65.04	23.12	All tasks on SG incl. inspection
3-Logist	59.45	21.13	Cleaning, decontamination, insulation works
4-Elproce_Handl_Ventil	23.75	8.44	All works for electrical equipment & I&C, ventilation, polar crane test
5-Valpump	83.94	29.84	All works for pumps and valves
6-Others	11.41	4.06	Other works, supervisions, health staff tasks
7-INSPNDE	1.61	0.57	No-destructive testing of welds, examination of supports
GLOBAL	281.31	100.00	

- **Basis: 12 months sequence**

In conclusion, Mr Lignini (AREVA) considered that from reviewing the publicly available information (e.g. regulatory bodies who have licensed the EPR™ reactor or which are about to deliver Design Certification), an INPRO assessor may conclude that CR3.1.1 on 'Occupational dose' is met.

4.7.3. Availability of References

Mr Lignini indicated the availability of related EPR™ safety information on the US NRC, UK ONR (Office for Nuclear Regulation) and France ANS websites. In the US NRC website, there are final safety analysis report and also US NRC safety evaluation report which still includes open items. He expected design certification by the end of 2014 or by the beginning of 2015.

4.7.4. Discussion

Questions	Answers
Mr Reyes (Chile) wanted further explanation for the difference in the construction time for the two examples (Olkiluoto in Finland versus Taishan in China).	**Mr Lignini (AREVA)** provided several examples. First, pouring of first concrete in Finland (Olkiluoto) was done in 2 sessions as compared to just one session in China (Taishan) so it was much faster in China. In Finland, he noted that AREVA as the leader of the consortium was responsible for the overall project although the civil work was contracted to a specific company. There was difficulty with the composition of the concrete leading to stopping of work to improve the process. Another reason for the delay was the techniques for welding of liner inside of container were not optimized. All lessons were learned and integrated into constructing the Taishan reactor.
Ms Yang (China) asked if there is any aircraft design basis standard for the containment design; and whether there is a standard for the design.	**Mr Lignini (AREVA)** explained that the outer containment is capable to resisting to military aircraft and large commercial aircraft. There is a standard but the information is confidential for security reason, but Mr Lignini believed it is a big aircraft.
Mr Dulera (India) asked the factors involved in reduction of fuel cost.	**Mr Lignini (AREVA)** explained that the main drivers for reduction of fuel cost are increase of fuel burn-up and implementation of heavy reflector inside the reactor vessel which limits leakage neutron outside the reactor which improves efficiency.
Mr Kupitz (Germany) asked the kind of material used for heavy reflector to improve neutron economy and to have less embrittlement of the surrounding pressure vessel	**Mr Lignini (AREVA)** indicated it is stainless steel

Mr Koji (Japan) wanted to know about pump trip given that EPR™ can operate without shutting down the reactor with 3 pumps. But under normal operation, if one pump trips, how to prevent reactor from shutting down automatically due to the low flow or high temperature signal.	**Mr Lignini (AREVA)** responded that he may seek the appropriate information from the specialist.
Mr Grudev (Bulgaria) wanted to clarify the statement given that for published document, the requirement is reached, whether for real non-published document, the requirements are met.	**Mr Lignini (AREVA)** responded that most of the time the information is owned by the client / utility and permission is needed for transfer of the information. The SAR and topical reports are posted publicly and can be accessed on the websites of US NRC, UK ONR, etc. Mr Lignini emphasized that he was not doing an assessment but simply provided insights and examples and where to find information. He was confident that potential assessor with such information can conclude that.
Mr Hj Khalid (Malaysia) wanted to know the main contributor in the reduction in waste	**Mr Lignini (AREVA)** explained that the main contributor is the increase of fuel burn-up because this will generate less waste regarding fuel cladding and also elements in fuel assemblies. It is the most significant contribution for medium activated waste.
Mr Reitsma (IAEA) considered the 16 day outage is very short. Online maintenance may be one of the reasons, but perhaps there are other steps in the design to achieve this.	**Mr Lignini (AREVA)** would discuss the question during presentation on the occupational dose criteria, and the fact that reduction in the occupational dose is relying on 3 factors, i.e. reduction of source terms, layout provisions and maintenance friendly features.
Mr Koshy (IAEA) indicated that while addressing tsunami effect on flooding on-site, the presenter mentioned up to 14 m the design could withstand. He wanted to know the onsite emergency diesel generator, whether the air intake and the exhaust are designed to be sealed from the effect of any flooding so it can withstand such a high level flooding on site or how AREVA addressed that.	**Mr Lignini (AREVA)** explained the wave was 14 m high, but effectively on the platform the water was only 4 m and this was considered in the analysis. Effectively the air intake in the emergency diesel generator building is higher; the bottom part of air intake is around 7 m high, so above 4 m on the platform.
Mr Korinny (IAEA) asked the availability of analysis and calculations to avoid criticality in case the debris following a core melting would block the pipe between the reactor compartment and the spreading area potentially causing criticality.	**Mr Lignini (AREVA)** believed there were calculations performed in the spreading area regarding the accumulation in reactor pit and transfer to the channel in the spreading area. In order to limit risk of criticality, the sacrificial material at the bottom of reactor pit includes some neutronic poison. There were some experiments conducted at AREVA and other testing facilities to demonstrate that once in contact with molten corium there is good diffusion of the sacrificial material in order to limit the risk of criticality.
Mr Koshy (IAEA) asked clarification for the two scenarios. First, when heading for severe accident, feeding is stopped feeding to reduce the possibility for H_2 production. Secondly, after temperature is progressing toward the worst scenario, it is very likely that the cladding melts first and comes down because fuel melt temperature is higher than that of cladding. So zirconium will be coming down in liquid form further down. As it spreads into the spreading area, there is fuseable link to provide extra cooling. So	**Mr Lignini (AREVA)** explained that for the second scenario, there is a metallic plug underneath the reactor pit which ensures that spreading is not taking place immediately. In fact, there is going to be mixing inside the reactor pit, so that it is not just zirconium that is going to flow first into the spreading area. He was not sure but believed there was a simulation conducted relying on experimental data to show that all H_2 that could possibility be generated from reaction of zirconium with water has already been generated. He believed the issue was sorted out. One has to take into account that in the spreading area, cooling starts from the bottom, i.e. dry cooling, then after 5 minutes there is water overspreading ensuring cooling from the top.

this means the melted zirconium will get a supply of water which again will cause explosion.	
Mr Koshy (IAEA) recalled the scaled down experiments that were conducted using the actual material to see that the melt down begins with zirconium which has the lowest melting point. If the experiments were done with fuel material and zirconium, then the behaviour in the scaled down version can be seen. So he wanted to confirm whether the tests were done with actual material or compatible material.	**Mr Lignini (AREVA)** confirmed the experiment was performed with the actual material.
Mr Tjahjono (Indonesia) asked about the material used for sacrificial material for core melting.	**Mr Lignini (AREVA)** could provide the answer separately, after the meeting.
Mr Reitsma (IAEA) pointed that in standard design evaluation filtered venting for containment is not needed, however, for the most severe case he wanted to know the overpressure as a function of time (day or week) in the containment.	**Mr Lignini (AREVA)** was not sure about the figures but the pressure should rise to ~5 bar after one day. Afterward, the pressure inside the containment will drop quite fast, and in the midterm it will stabilize at ~2-3 bar and slightly decreases over time. More detail information can be provided later on, or one can check on the publicly available website. When conducting the stress test, the design pressure for the containment is 6.5 bar abs, which was already verified and demonstrated. In fact there is some margin, and the containment could maintain its integrity up to a pressure between 10-11 bar.

5. FEEDBACK ON INPRO METHODOLOGY

Mr Kupitz (Chairman) requested vendors following presentations and during the afternoon discussion to provide feedbacks on the INPRO Methodology. At some point, he noted that the INPRO methodology was developed unfortunately without major involvement of the designers. The feedback provided in the Dialogue Forum will be taken into account since the methodology is under the revision. But vendors may have other areas and points for involvement in the future revision. The feedback is summarized as follows.

Mr Fujii (Mitsubishi) considered some INPRO questions require very detailed information including for manufacturing. Design of a NPP is done in a stepwise manner, from the conceptual, fundamental, detail and manufacturing design. When MHI obtains the design certification, not all components are fixed. From the business point of view, information like cost and impact to the environment also cannot be made available because of business competition. He also pointed out that without accumulation of operational data or history, MHI cannot estimate criterion such as probability of reliability. For example, the RV manufacturing method has been improved but to show the evidence how numerically the reliability has been improved is impossible. Maybe it can be done 30 years from now. But smaller number of welding seams increases the reliability but it cannot be shown numerically. Mr Fujii also noted that INPRO Methodology asks for the best design and many (INPRO) questions ask vendors to do what vendors will do for a particular NES. The fact is vendors prepare a variety of designs as requested by the utility company, which will be decided after the contract specifications are fixed, e.g. radioactive waste management or fuel cycle downstream policy which is decided mainly by the government. Vendor will design according to the request by the user's state / government. Also electric power companies like in the US and Japan have their own culture about the design.

Mr Bykov (Rosatom) questioned on the philosophy of assessment concerning DID, and the UR needs detail definition. He had the opinion there is a mixing of assessment principles because of mixing of functions in different levels. **Mr Drace (IAEA)** responded that this will be addressed in the revision of the INPRO manual, with more detailed guidance for the assessment. **Mr Korinny (IAEA**) added that the UR1.1

discusses the first level DID for normal operation, and indicated the manual will be revised and updated for clear wording.

Mr Bykov (Rosatom) also pointed out that EP 1.1.1.2 Simplicity of design (e.g. reduced number of components of the reactor system or reduced number of active components) is not suitable for evolutionary projects on the basis of active systems because the main principle of improving the reliability and protection against common failure is redundancy and diversity. He also commented that if the criterion EP 1.1.1.5 is related to the conditions of normal operation, to DID Level 1 or 2, then this option should not be considered in the particular UR. This option should be applied to DID Level 3 (DBA) as a basic requirement and as recommendation for Level 4 (BDBA). He added the necessity to discuss additional EP "Diversification" for fulfilment of safety functions evaluation assessment.

Mr Kolchinsky (Rosatom) stated that CR1.6.1 should be simplified. The INPRO methodology should be made more user friendly. **Mr Zhou (China)** suggested a list of specific design parameters be included in the revised version of the INPRO safety manual considering many similar design parameters such as for design margins and simplicity were presented by vendors. This would allow Member States to compare the design margins with the reference design. INPRO proposed that the revised version of INPRO safety manual would be distributed to the vendors to obtain critical comments and necessary design parameters.

Mr Beard (GE Hitachi NE Intl.) brought up question on chasing safety versus optimizing design, safety versus cost and issue on margins, i.e. chasing margin versus return on investment or return on increasing margin, and how these could be facilitated by INPRO (in the methodology). The INPRO criteria selected for the ESBWR presentation, according to **Mr Miller (GE Hitachi NE Intl)** are very valid because the criteria highlight the safety and operational features of the design.

Mr Seo (KEPCO-E&C) responded that there are not really significant inconsistencies among the criteria. But acceptance limit for each criterion is sometimes 'better than the reference plant" or "equal or higher than the reference plant'. All reactor designs discussed in the forum are advanced, so by definition they are the better designs than the reference plants. In other words, without going through the details reference plants or assessment, vendors can claim the advanced designs are better. So, it is how comparing the advanced design with the reference design, which is the problem with the methodology. When looking through all the criteria, Korean advanced design is definitely better than the reference plants. Otherwise, the advanced design is done wrongly. **Mr Phillips (IAEA)** pointed out the case with no reference plant, i.e. a first-of-a-kind (FOAK). But for large commercial NPPs, he doubted the applicability of the methodology. Additional comment is the fact that every vendor participating in the forum selected safety area, and not the other two areas: environment and waste management. This is because safety has a lot of technical issues, and not for the other two areas from vendors' perspective. However, when talking about safety, the commercial NPPs must meet active or current regulatory requirements as the basis. Beyond that, maybe vendor can design beyond what is required by the regulator as option, but the minimum is meeting the licensing requirements. All reference designs and advanced designs meet licensing requirements; the only difference is by some advanced novel features. So safety area may not be a good topic for INPRO Methodology. **Mr Phillips (IAEA)**, however, noted that it is logically impossible to explain to the community why safety is not a proper topic of sustainability.

Mr Lee (Candu Energy Inc.) pointed out that BP1 places more emphasis on preventative rather than on corrective or mitigative barriers for DID. Lessons learned from Fukushima show more emphasis is needed on corrective or mitigative barriers than has been previously considered. INPRO assessment criteria should also be updated to reflect IAEA Fukushima Action Plan, i.e. to include severe accident management, monitoring and emergency mitigating equipment. Mr Lee noted the assessment did not consider the capabilities and availability of operators and emergency response personnel to perform mitigative actions during severe accidents – and this should be jointly done by reactor vendor and operator. Some general observations about INPRO assessment methodology are as follows.

- Lessons learned from Fukushima Dai-ichi nuclear accident have an impact on many parts of the INPRO assessment methodology. This includes impact to the public and disruption in social life. INPRO assessment areas for economics, infrastructure, waste management, physical protection, environment and safety should be reviewed to account for the Fukushima lessons learned and the IAEA Fukushima Action Plan.
- INPRO assessment criteria have not accounted for cyber security, which affects infrastructure, proliferation resistance, physical protection and safety areas. In particular on cyber security, although

covered to some extent in INPRO Physical Protection area, he stressed the fact that digital I&C in modern power reactors means the use of digital measurement devices, e.g. transmitter, flow measurement. Conventional heating, AC and ventilation system utilize digital control. So additional new modes that can initiate plant transients in the context of safety resulting from erroneous maintenance on digital control or malicious software, should be taken into account.

- The methodology needs to factor in the economic consideration of ALARA in the safety area when looking at reduction of releases and doses because there is a practical limit of what can be achieved during normal plant operation.

Mr Lignini (AREVA) indicated that even if UR1.1 'Robustness' relates to Level 1 of DID, 'Robustness against External Hazards' was treated as well in the presentation since it was considered not obviously identified in the INPRO assessment methodology. He proposed robustness against external hazards to supplement and improve the INPRO methodology. **Mr Phillips (IAEA)** responded that in the revision process of the methodology, one of the major drives is to take into account lessons learned the Fukushima. The part of external hazards is actually covered in the methodology but perhaps it is not clear enough. It is based on BDBA, so technically external hazards should be included. It will be made clearer in the description of the revision text.

6. QUESTIONS AND ANSWERS

The section contains questions and answers to promote further interactions among Dialogue Forum participants: panel of reactor designers, Member States and the IAEA, by addressing both oral and written questions prepared by participants to further clarify issues.

Questions	Answers
Mr Grudev (Bulgaria) noted that when discussing safety, all vendors already discussed the two digits. One of which is CDF in which every design meets the agency (INPRO) and national requirements. It would be interesting to have a table of comparison because safety is what vendors are promising, i.e. quality not to have core damage higher than CDF, not to have huge release to the environment. The third and fourth digits is about the price, important for country / company as operator to take decision, i.e. how to achieve safety and how cheap. This is the first to be solved in the methodology. There are many other criteria that decision makers could found or have some impact. But the above mentioned are the 4 important digits.	**Mr Beard (GE Hitachi NE Intl.)** agreed with previous statement from Mr Seo (KEPCO-E&C) that vendors are trying to optimize design features, not chase safety above everything else. So in criteria there should be recognition that sometimes decreases in margin might be acceptable as long as meeting recognized regulation standards for a number of reasons. For example, every utility in the US up until a year ago was pushing for extended fuel cycle because it makes more economic sense. Given the natural gas resource now, these utilities are discovering that purchase of replacement power is much cheaper. They have done study, one of which showed that a 12-month fuel cycle with a slightly better fuel utilization is more cost effective for the utility. So it is a game changer, a totally different way of thinking. So vendors make optimization on enrichment rate, burn-up, fuel utilization, etc. So there is a need of acknowledgement that sometimes even the design is not good or better than the existing plant, but as long as satisfying the regulatory requirements, that ought to be acceptable. **Mr Kolchinsky (Rosatom)** indicated that the methodology does not define 'depend on evidence to meet criteria'. In some cases, it is possible to say only yes or no, i.e. CR is met. Meanwhile in other cases, the evidence needs lots of documentation, research and experience. So the methodology does not define 'depends on evidence' and this should be taken into consideration. **Mr Bykov (Rosatom)** also agreed with Korean and US vendors, and added that it is necessary to make very detail analysis of the all criteria from the point of view of contradictory among CR and EPs. For example, redundancy in contradiction with simplicity, also there is no diversity analysis. Also situation with basic project, designer looks at normal operation margins, transient or safety margin, find the project is not optimized because of the large difference for example between normal operation conditions and some operational margins. It is possible to increase power to improve economics of NPPs, to reduce cost of electricity, and slightly reduce the rate between the operational

	parameters and margins. This will go on, e.g. reduce margins but it is not reasonable from economical point of view.
Mr Reyes (Chile) noted that one of the most important things is cost. The concept of overnight cost gives an idea of the cost of NPP. But the overnight cost does not depend on interest rate and others. He suggested to start the concept of redundancy according to the basic design, e.g. redundancy with 2 or 3 loops, maybe the concept will be built and the cost will be 0.3 times; to give an idea, that is, the redundancy and improvement of technology according to the cost.	**Mr Beard (GE Hitachi NE Intl.)** hesitated to give the overnight capital cost, because of the significance influences that need to be discussed, e.g. local labour, local commodity cost, and localization requirement. Secondly on redundancy, when GE Hitachi NE Intl. made an optimization for ABWR which has to go with 3 times 100%, the reason to do that was that by adding another train, there is only 5% reduction of CDF yet at a cost of about $100 billion. That is an example of chasing redundancy providing very little additional benefit. **Mr Bykov (Rosatom)** commented that some authority requires not only redundancy but also diversity, like Finnish authority. So it is not possible to achieve simplicity in this condition. **Mr Seo (KEPCO-E&C)** added the fact that Finland is willing to pay.
Mr Susyadi (NPTDS) commented on the safety manual of the INPRO methodology for a NESA. For technology user country, his impression was that it seemed since vendors use own reference plant, and the fact vendor for Korea mentioned, the advanced design is definitely better than the previously designed reactor. He wanted to stress two things. The safety cheaper, the best assessors are the vendors. Vendors make the assessment and publish it. If done, INPRO Methodology is more of a structure way to sell their reactor. So it is not beneficial for newcomer country trying to look at their energy system sustainability, e.g. comparing safety aspect of later design with existing reactor. The question was the relationship with sustainability, whether the reactor has longer sustainability because of the better design or we want to have deeper answer / explanation. Another thing is technology user countries want to know from the different reactors, which one is the best. But the IAEA has already had other guidelines, i.e. technology assessment. This INPRO methodology provides more technical and quantitative, which is still useful. But when looking at the vendors compare only with their own reference	**Mr Phillips (IAEA)** responded that to understand how INPRO ends up with the methodology has to do how any project within the agency is done. Experts gathered from MSs in experts consultancy meeting, within the given situation and political nature of the agency. For example, technology selection. If one reads the technical document, it is completely contrary to how the work is done at the agency for obvious reason. The presentations have a lot of information and detail how the particular vendors through technical means meet the objective in a clear way. How member states to use that approach that in their own time, not published under the agency heading, also if how MSs want to use technology selection. But when we talk about assessing technology for sustainability, the only thing that INPRO can recommend from INPRO political position is MSs compare against the lineage of that technology but that does not mean the only thing MSs can do. INPRO expects MSs make the best commercial decision MSs can make. He further noted that INPRO Methodology has a long lineage, and all of these (philosophical) issues have been repeatedly discussed. Eventually, just like to do safety assessment to satisfy regulator, you need to show details why one can make a statement. INPRO is showing that in order to make any kind real technical statement on sustainability, there has to be a reason to make a statement. **Mr Beard (GE Hitachi NE Intl.)** would advise against the vendors doing the assessment because it would end up like doing EUR / EPRI assessment, where the assessment is done by vendors and then reviewed by government body to accept that. He agreed with Mr Phillips (IAEA). There are lots of data involved to point the MSs, and vendors certainly cannot point if there is an interest. But the actual assessment should be done by the country that is trying to do the assessment. **Mr Bykov (Rosatom)** added that looking at the INPRO methodology as a set of criteria, it is possible for each newcomer country to choose the most important criteria for the local conditions and to carry out assessment based on these criteria. He noted the same documents like US URD, EUR utility requirements, but site documents can be produced using these database. MSs can choose what to see in the new NPP **Mr Popov (Belarus)** shared view from technology user point of view. From the forum, it is seen no one can make better assessment other than vendors. But the question is whether vendors are really interested in doing the assessment. He had the impression that it is not so much

plant, then there is not much benefit. Maybe later if vendors produce their own assessment using the structured INPRO methodology, then user countries can get much more benefit. But as indicated, vendors do not want to provide any information without business deals, the question remains how user countries can get benefit from the INPRO methodology.	interested because they have enough work to proof advantages of the technology, e.g. SAR and documents presented to regulatory body. Secondly, there is some contradiction between which users of technology to get information as early as possible. Belarus did not have enough background to get the information. This explains why in Belarus report there is lack of information to make a good assessment.
Mr Reyes (Chile) asked if the enhanced designs of the reactors from all the vendors improved by reducing the amount of nuclear waste produced.	**Mr Fujii (Mitsubishi)** replied saying that each designer can provide information about waste reduction depending on the infrastructure of the country in which the NPP is situated. **Mr Beard (GE Hitachi NE Intl.)** adds that it also depends on the type of waste (high, intermediate, low level) but there is a slight reduction in most categories in the new design. Minimising maintenance minimises low level waste. High level waste, primarily fuel, is decreased by increasing fuel burn-up. New technologies are being tested and developed continually to reduce waste. **Mr Seo (KEPCO E&C)** also adds that reducing low or medium level waste can be done by installing a verification instrument to reduce the volume of solid wastes. **Mr Drace (INPRO)** suggests that there are different types of waste so there is no simple answer to this question.
Mr Tjahjono (Indonesia) asked that in some criteria, one has to compare the performance of evolutionary reactors to a reference plant according to the evaluation parameter (EP) considered. In many cases, the analysis data for the EP for both reactors are obtained from two different sources with different methods. In these areas, how one can decide which reactor is better than another. This is because for the same input data to be calculated using different methods would provide different results.	**Mr Phillips (IAEA)** explained that INPRO methodology is not designed to do technology comparison outside the lineage of the technology. The host state can look at the kind of questions and can use it as they want. But as a matter of policy, INPRO does not engage in technology comparison. On issue regarding comparison of results using different methods, this has come up in several places in the INPRO methodology in the past, and clarification that is always given is that one should try to make the things being compared as similar as possible. For example, on calculations and methods of calculation, when comparing dose to environment or personnel that is based on conservative estimate should not be compared to actual measurement from operation. It requires some work to try to find comparable things to make useful comparison.
Mr Tjahjono (Indonesia) asked concerning the reference plant to be compared with the innovative reactor, whether there are any criteria or methodology in choosing the appropriate plant for the comparison.	**Mr Phillips (IAEA)** stated that the recommendation in the INPRO methodology is to pick the most recent installed operational technology in that lineage, e.g. for AP1000 it would be PWR Sizewell B in the UK. However, in the case of the comparisons done by GE Hitachi NE Intl., there are a number of plants in the similar part of the lineage so there will have to be a judgment to make about which specific site is more appropriate. From the INPRO methodology, one must try to pick the one which is most recent or more comparable to what is being assessed. On innovative technology, if there is a reactor that is not comparable to anything, such as the ESBWR which is very different from the predecessor technology and could be argued as an innovative re-design rather than an evolution, then it is up to the assessor to make a technical judgment about what similar technology it should be compared with.

	Based on the assessment done by the vendor on the ESBWR, the INPRO criteria are sufficiently generic.
Mr Tjahjono (Indonesia) asked concerning the acceptance limit "superior to existing designs in at least some of the aspects discussed in the text", whether the aspects considered only depend on the assessor. If that is so, it could be rather subjective because the assessor could choose only some convenient aspects and leave the remaining inconvenient ones. He proposed that INPRO methodology determines also the aspects which should be at least considered in each evaluation parameter.	**Mr Korinny (INPRO)** acknowledged the issue and indicated there have been discussions on about which parameters should be obligatory and which should be only recommendation. INPRO will take this recommendation into account in the next version of the INPRO methodology which is currently under development.
Ms Yang (China) asked that since the INPRO assessment in the area of safety is based on IAEA safety standards, what the exact relation is between them, whether the IAEA standards are necessary for the INPRO assessment.	**Mr Phillips (IAEA)** answered that the IAEA safety standards are a foundational element in the original development of the INPRO methodology. It takes them into account. When we get a clearance on the revision of the document or the new document, the IAEA Safety Department clears on this document. INPRO assessment is not the same as doing safety assessment directly under the standards but not in contradiction. The assessor should be knowledgeable of the IAEA safety standards because on several occasions there is a presumption that an INPRO assessor in a particular area is a subject matter expert in that area. A subject matter expert from a host state must be knowledgeable of their own states regulation in that area and also of the international view of regulation in that area, which are the IAEA safety standards. Therefore, it is not a safety standards analysis or assessment. It is an INPRO assessment but the subject matter expert is expected to be aware of the IAEA safety standards.
Ms Yang (China) noted that the INPRO assessment in the area of safety is at the macro level, but most of the assessment answers are 'Yes' or 'No'. So, she wanted to know how it can be made more available and reliable for Member States. She later added that if some detailed information is provided then it could be more helpful to the Member States.	**Mr Phillips (IAEA)** explained that in many parts of the assessment there is a comparison asking whether something is equal to or better and occasionally there is a numeric value. In the case of equal to or better, the logic is about incremental improvement in the lineage of technology as an indicator of the direction of sustainability. In terms of the actual underlying evidence, this is not just a yes or no question as in order to answer one has to provide evidence of why it is 'Yes' or 'No' and this involves technical information like seen in the presentations, making an actual comparison of technical information necessary. In the case of some of the target numbers, they are generally drawn either from the IAEA safety standards drawn from some major regulatory systems so the origins of the numbers (e.g. frequency) are not arbitrary. The detailed information is for the assessor to provide as evidence in order to support the answer.
Ms Yang (China) asked how to assess the "practical elimination" of the major release of radioactivity.	**Mr Korinny (IAEA)** answered that the current version of the INPRO methodology manual was drafted in 2006 when the authors had a rather weak idea of what the "practical elimination" of the design and just put requirement that the frequency of containment failure should be significantly less than 10^{-6}. This is currently the only description of the "practically eliminated by design". Information has been accrued so in the next version of the INPRO methodology more discussions will be added on this issue, including information gathered in the Dialogue

	Forum from AREVA. **Mr Lignini (AREVA)** responded that as a representative of the industry's nuclear safety standard committee he attended a meeting that discussed the need to develop guidance about the assessment of 'practical elimination' because it is not covered by safety standards currently. There is no detailed explanation or guidance on how to assess whether something is effectively practically eliminated or not. It is important not to base the decision only on probabilistic assessments but to demonstrate effectively the probability of occurrence of a sequence which is claimed as practically eliminated to be very low. Expert judgement is needed to convince it because of the very low figures that may be interpreted and arrive at reasons as one wants. In the medium term the IAEA will provide a safety guidance / standard on 'practical elimination'.
Mr Ordabayer (Kazakhstan) asked about the advantages and disadvantages of SMRs of LWR type which are under construction and or planned, with an electricity range: 600-1000 MW. In particular, he would like to know about VVER-640.	**Mr Phillips (IAEA)** pointed out that the IAEA NTPDS has a project area that follows the technical developments on SMRs and the information is available on their website. INPRO has also had a dialogue forum on SMRs so there is also information under INPRO. Since the question is referred to a Russian product, he referred the question to Rosatom. He suggested that since it is a commercial discussion as it is about advantages and disadvantages, then it should be discussed directly with the vendor. **Mr Kupitz (Chairman)** added about a publication called "Design features to achieve defence in depth in small and medium size reactors" and there is also an advanced information system online in the Agency called ARIS (Advanced Reactor Information System). **Mr Fujii (Mitsubishi)** explained that based on his 30 years work experience for electricity company, he had to choose the capacity of NPP in Hokkaido. Because the electric company was not so large, the maximum capacity of NPP must be decided according to the grid capacity and economics. The index function to decide the capacity is cost, because for a large capacity of power plant, there must be a standby power plant during maintenance outage. For Hokkaido electric company, for example, the capacity of 1 unit of NPP should be less than 50% of the grid. **Mr Drace (IAEA)** added that the topic of this Dialogue Forum was the application of the INPRO methodology on the type of the reactors in discussion with the reactor vendors who were willing to discuss their design and the use of INPRO methodology. On the other hand when a country is doing an INPRO NESA, regardless of whether it is a limited or full scope NESA, they try to consider all different options which are good for that particular country. So for Kazakhstan who is interested in SMRs, it was suggested that if the country plans to do a NESA, then they can look at the advantages and disadvantages of large reactors over SMRs. **Mr Harper (IAEA)** pointed out that the IAEA ARIS database contains very detailed technical presentation of advanced and evolutionary reactors. All 7 vendors that have presented during the Dialogue Forum have given the IAEA their technical data which is available in the website in pdf format and also as a part of the database. The database was given to the IAEA, and then reviewed to remove bias or claims or superfluous information, and the information was sent back to the vendors for approval. So it is vetted and consistent across all the type of reactors. Currently there are 40 different advanced reactors in the database, so if one is looking to compare one system with another and this system includes light water, heavy water, boiling water, pressurized water reactors, SMRs, gas cooled reactors as well as fast reactors, then

	ARIS is an excellent source of information. It is clear, consistent and unbiased.
Mr Zidi (Algeria) asked Mitsubishi the tools for getting the number of the core damage frequency (CDF) and the large release frequency (LRF).	**Mr Fujii (Mitsubishi)** replied that he did not have the information about the names of fault tree and event analysis codes used but would send the information on the behavior and probability analysis codes once available.
Mr Zidi (Algeria) asked that for a full assessment, how much time will be needed and whether participants have all the data. On contacting vendor directly, he later argued that it means the country has already chosen the technology since vendor is already informed.	**Mr Phillips (IAEA)** indicated that in the opening presentation he had shown the estimate for a simple full NESA as requiring about 2-3 total man years for assessment in all areas but for a very simple system not including fuel cycle. Collecting the data has a sensitive effect on how long it takes. INPRO does not collect the data for assessors. INPRO has a support package and there are other supports within the IAEA. Assessors can also find information on their own. **Mr Kupitz (Chairman)** noted the data needed for an INPRO assessment are not the ARIS database fully. One needs to visit the websites provided by vendors or contacted them directly. **Mr Phillips (IAEA)** added that in several presentations there are listings provided by vendors on how to collect data from internet directly e.g. US NRC website (EPR, US-APWR), or from regulatory (EC6). For more, the assessor needs to go to the sources including to speak with the vendors.
Mr Zidi (Algeria) asked whether it is possible to have the results of assessment by INPRO methodology to AES-2006.	**Mr Phillips (IAEA)** responded that it was completed. There are a few places missing but with the presentations done in the Dialogue Forum plus Tecdoc 1716, the assessment is complete for AES-2006. It is site specific to Belarus so one would have to consider site issues separately
Mr Zidi (Algeria) asked whether (Tecdoc 1575) Volume 8 fulfils all the requirements for sustainability.	**Mr Phillips (IAEA)** replied no, saying that (Tecdoc 1575) Volume 8 considers only safety. One can have a reactor that is extremely safe and is entirely unaffordable, and this is not sustainable. One has to look at all the different areas. The areas are selected as what the MS experts feel are the minimum set of issues that have to be addressed to answer the question of sustainability.
Mr Calabrese (Argentina) wanted a concrete example of a gap in the INPRO methodology and ways to close it.	**Mr Phillips (IAEA)** informed that the INPRO methodology does not address the question of what is the proper follow-up action or the way to close a gap. How a MS chooses to do that is up to them. However the Tecdoc 1716 has examples of gaps in the full scope NESA that Belarus performed. He provided an example in the waste management which requires an end point for each type of waste. At the moment the Yucca Mountain project has been cancelled in the United States. This means there is no end point for high level nuclear waste in the United States and this is a gap. There are more detailed technical examples in the Tecdoc on Belarus NESA.
Mr Calabrese (Argentina) asked in which way an assessor is qualified to assess the robustness of a new design.	**Mr Phillips (IAEA)** noted it is a subject matter expertise. The success and efficiency of picking the correct staff to perform a NESA is very strongly dependent on the level of expertise of the assessor. Picking the right person will be much less painful and picking the wrong person means the assessment will probably never be achieved.
Mr Dulera (India) asked how the generation of hydrogen subsequent to a postulated accident is addressed.	**Mr Phillips (IAEA)** indicated that in two particular UR 1.4 and 1.3, under the heading of engineered safety features, it specifically describes hydrogen igniters and combiners. He suggested one keyword searches in the manual to find it.
Mr Dulera (India) asked that with reference to the new design of VVER, what is the basis of H_2 recombiner design, i.e. how long and how much H_2 can be	**Mr Kolchinsky (Rosatom)** explained that as for H_2 recombiners installed in Tianwan NPP in China NPP from Framatome manufacturer, and Rosatom continues work with the manufacturer. As for the capacity of the catalytic recombiners, it is chosen for all possible generated volume of H_2 in the core through the zirconium and steam reaction.

absorbed. When the limit is reached, he wanted to know what happens.	Nothing will happen after the recombiner is not effective anymore because there is no other source of hydrogen in the reactor building. Mr Phillips (INPRO) added the catalyst is not passivated by the reaction; it continues to act as long as there are hydrogen and oxygen. There have been tests by the manufacturer to study the behavior. **Mr. Lignini (AREVA)** commented on the number of recombiners that have to be implemented inside the containment. One has to consider not only the individual capacity of the recombiner but also its location and the circulations inside the containment. Simulations have to be performed in order to locate the recombiners. It is also important to make sure that the recombiners are not going to be affected by the sequence and phenomena which occur in the containment. So there is no simple answer to the question. One cannot simply divide the amount of hydrogen produced by the individual capacity of the recombiner; one must perform quite complex simulations.
Mr Dulera (India) asked how to ensure the re-criticality does not take place in the core catcher.	**Mr Kolchinsky (Rosatom)** answered that the internal material of the core catcher has gadolinium for neutron absorbing and the internal material also reacts with the damaged core as to dilute its fragments and make it impossible for the fission process to take place.
Mr Dulera (India) asked whether the vendors feel whether full or partial assessments of their designs done on the basis of the INPRO methodology would improve public acceptance of the design.	**Mr Phillips (IAEA)** answered that from a vendor perspective, the first constituency the vendor has to satisfy is the regulator and the utility. The question of broader public acceptance and international acceptance is really what IAEA standards and best practices cover. In a particular country, speaking from personal experience and not as an IAEA staff, the national regulation is the most important. From the point of view of neighbors and regions, it is a larger question.
Mr Calabrese (Argentina) asked about models used to validate innovations in designs, whether it would be necessary to use some standard procedures for these kinds of tests.	**Mr Phillips (IAEA)** commented that the question should be addressed by regulators as they decide these things. There are probably variations in what the standard procedures are. He was not aware of the details except in few narrow cases in which there are standard procedures. The only way to satisfy these questions is to follow the procedures of the regulator. If vendor sells in multiple countries, then the vendor gets to address multiple regulator procedures and methods. Mr Seo (KEPCO-E&C expressed his agreement. It comes against the question of the balance of international standards versus the sovereignty of the state, the latter of which is typically dominant.
Mr Calabrese (Argentina) asked about probabilistic safety assessments for innovations in design, that is, how it is possible to determine failure probabilistic for an original system.	**Mr Miller (GE Hitachi NE Intl.)** responded that looking at each feature which is innovative in a way or another, one certainly try to understand failure modes and mechanisms and usually those come down to physical properties. If the system is truly passive, lots of the common failure modes are removed. With probabilistic safety assessment, however, a large portion of that has uncertainty. **Mr Fujii (Mitsubishi)** explained it is very difficult to prepare such reliability data. Therefore, designer wants to use the reliability data having similar design. For example, if there design is improved, still there might be similar existing design and in such case, designer uses reliability valid data. This means even for innovative design, designers must use original old data. This means result of PRA does not change to good direction. That was done for the US-APWR, as already addressed in the presentation. For example, the reactor vessel break probability of 10^{-6} was decided and assumed more than 40 years ago, and ever since no data has been obtained. Such way must be used for innovative design. So innovative design can have more reliability but cannot be predicted. **Mr Lignini (AREVA)** added the importance of level of confidence in the data. It is important to stress that regarding reliability there is no simple and absolute reliability. For example, on passive system and

	component, the IAEA has launched collaborative research project, and within INPRO there was a collaborative project PGAP (Performance Assessment of Passive Gaseous Provisions) aimed at defining the reliability for passive system and also building a methodology to evaluate this reliability. For passive systems, it was shown that the effectiveness of the system really depends on the condition, mission and transient to respond. There are a lot of parameters which are going to evolve, like the roughness inside the pipe, which will influence the behavior of the system. So there is not absolute reliability, but it is impossible to whether for a particular transient, the system will perform as requested or not. So, there is no simple answer to the topic.
Mr Zidi (Algeria) asked whether the INPRO methodology has been benchmarked against other assessment methodologies.	**Mr Phillips (IAEA)** responded that in the first instance there would have to be another methodology with the same objective. He did not think that there is another methodology. In the past there have been some misunderstandings, for instance, that the methodologies developed by GIF are comparative when they are actually not. INPRO has had several collaborative projects that have involved members from GIF and INPRO and the GIF methodologies are much broader, not just asking on sustainability problems. GIF does not try to ask the minimum number of questions. They are much more generic in their approach. As an example, in the area of proliferation resistance, the INPRO manual is very short and the GIF methodology occupies an entire binder. So, to do a benchmarking, there must be two comparable methodologies. INPRO methodology is rather unique and for example, in the area of safety, it was derived from the safety standards. It tries to develop a minimum set of questions on the subject pertaining to sustainability only and not all the questions of safety. Since it is derived from those sources, INPRO hopes that it will help a country to discover their gaps and help in the planning.
Mr Zidi (Algeria) asked ABWR whether the feedback from Fukushima accident has been taken into account in the new design, and if so what the main improvements are.	**Mr Yasuda (GE Hitachi NE Intl.)** explained that mobile countermeasures and backup building are added to protect mobile equipment; there are also additional DC batteries and alternative AC generators, and so on.
Mr Silva (Thailand) noted that there are a number of criteria in the area of safety in the INPRO methodology. When gaps are identified, the effort to fulfill one gap may worsen another gap or even create a new one. Furthermore, it may create some other gaps in other areas. He wanted to know how the assessor can prioritize the criteria determined in INPRO methodology, especially criteria in different areas. In case of multiple assessors, he asked how to make sure that the recommendations to fulfill a gap will not create another gap.	**Mr Phillips (IAEA)** noted that the first question is basically deciding on weighting of priorities. INPRO methodology intentionally avoids the question of weighting and coming up with weighting functions to decide priority. This was decided within consultancies and within the INPRO group, that the structure of the manual would not consider weighting between different parameters since the objective is not to make a judgment call on what is more or less important in a given case. INPRO shows the whole space of important issues that an answer, one way or another, may have a significant impact on sustainability. When a country performs a NESA and they obtain a complete gap map, then the country has to decide the efficient way to balance these questions. On the second question, it is possible to cause a problem by making improvements elsewhere but this is the question of how to balance priorities again. In many cases, where this occurs, the structure of the methodology says that one makes improvements in some areas – to be clarified further in the revision of the manual. In the case of multiple assessors, INPRO recommends that these assessments be coordinated between areas at some point when the assessment becomes mature. If a NESA is coordinated as it develops, then these issues are generally resolved. If there are several assessors working in isolation, then there will be an uncoordinated outcome. There is a manual describing

	approaches to performing a NESA that addresses these questions.
Mr Silva (Thailand) asked that when the INPRO methodology addresses core damage frequencies or large release frequencies, whether the scope of the initiating events includes external events. If so, then what kind of external events are included.	**Mr Phillips (IAEA)** indicated that the question had been touched on before. If one thinks about what can be contained in the phrase 'design basis accidents', then all those things are theoretically captured. In the new manual INPRO will give more specific language about that so there is no confusion on whether external hazards are captured or not. It is captured. He later added that the format of the meeting allowed the vendors to select a limited number of criteria to present during this meeting because of the lack of time but this does not mean that all the criteria could not be assessed. **Mr Korinny (IAEA)** explained that INPRO does not specify explicitly that external or internal events should be covered in your assessment. INPRO does not distinguish these. Depending on the data available and depending on the situation and status of the design, one may not have site specific data for external events. In this case, these issues are not covered in the INPRO methodology yet, but in the updated version of the INPRO methodology it will be covered.
Mr Zhou (China) noted that Mitsubishi's and Rosatom's presentations were valuable and explained how to assess safety of the advanced PWR by analysing some URs and CRs in detail on the design results of advanced PWR. The EPs on design margin and simplification are major and important aspects to assess for an innovation reactor design. Based on those assessment, he suggested that a list of kinds of design parameters of PWRs which need to do 'design margin' assessment, and a list of kinds of reactor system or subsystem of PWRs which need to do 'simplify design assessment' would be supplied as an example in the user manual of INPRO methodology. This kind of example would be helpful for providing guidance on the assessment of fast reactors such as TNPP. Supplement example in the user manual of the INPRO methodology is an important work and major part of updating the user manual.	**Mr Phillips (IAEA)** explained that the existing manual currently provides examples in the text and one can see how details are handled in Tecdoc 1716, the full scope NESA performed by Belarus. It is not certain of what additional language may have been inserted in the current draft of the safety revision.
Mr Ibrahim (Egypt) asked how the new comer country can confirm the long term sustainability of its nuclear power programme.	**Mr Phillips (IAEA)** said that INPRO recommends performance of a NESA. However, if a country wants to ensure long term sustainability of its programme, this is an on-going process as the nuclear programme grows, where there are gaps and continuing plans to close the gaps over time. INPRO decided early on in the process of developing the INPRO methodology that it should be a process to achieve sustainability without expecting that it is just a onetime declaration. This is because generally nuclear energy programmes evolve considerably over decades so this question has to be considered as the programme develops.

Mr Ibrahim (Egypt) asked with respect to the new comer country, what the relationship among the tools for energy system planning, the INPRO methodology and the milestone approach is.	**Mr Phillips (IAEA)** answered that INPRO considers nuclear energy system planning in a broad sense to be a prerequisite to performing a NESA. When INPRO has an initial meeting to address the interest of Member States in NESA, if INPRO makes a determination that they may benefit from the broader energy system planning, then we immediately involve the PESS group in the IAEA that does the energy system planning. Then PESS will give them training and provide them with software tools to develop their own energy system planning. The INPRO methodology begins by presuming that the nuclear energy fraction of a broad energy plant has already been established, so INPRO does not address that problem which has been answered by tools such as in PESS. Many countries have their own established energy system planning already and INPRO recommends they use it. In terms of other areas such as INIG which has a strong focus (milestone approach) on first nuclear power plants or a second plant if there was a long delay since the first one, there is also a document called NE series NP/T/1.12 which describes the relationship between these services.
Mr Arshad (Pakistan) asked that since most of the advanced reactors are being proposed for large powers (1200 MWe or above), whether this will not decrease their clientele. He wanted to know whether these reactors are meant for large utilities only. He asked whether the advanced reactors can be built/designed in a lower power range (<1000 MWe).	**Mr Miller (GE Hitachi NE Intl.)** indicated that designing a NPP takes several years so a decision has to be made on what the parameters are and what the output is. A decision has to be made for the design to carry on. In GE Hitachi's case, there are different offerings; ESBWR and ABWR. The vendor also has PRISM reactor which is a smaller fast reactor. In larger reactors it would be very difficult to go back and decrease power significantly down to ~500 MWe output. This would be a redesign of the plant and a decision that vendor would have to make commercially. **Mr Seo (KEPCO E&C)** indicated that as a vendor, if there is a market, vendor will design and develop advanced smaller sized NPPs. That is the key factor. Recently the nuclear market was giving more attention to the bigger plants and that is why the vendors developed larger NPPss. For Korea, however, the APR1400 was developed based on the OPR1000 plant and afterward the same advanced features were then backfitted to the OPR1000 which was called APR1000. Korea has a completed design for this and ARIS has a report of this plant in the IAEA website. The APR1000 however has not been built in Korea so a newcomer country may not want to buy and operate a plant which has not been built and operated in the vendor country, although technically Korea can design smaller sized advanced reactors. **Mr Phillips (IAEA)** commented that the scale of a market strongly affects the business plan of a vendor. This is an intractable issue the way business is done. **Mr Fujii (Mitsubishi)** explained that in the Dialogue Forum the presentation was referred to the US-APWR because the document evidence can be easily obtained by the user. Mitsubishi has a variety of NPPs for the customers. For example, if the customer needs a 1000 MWe class output, then Mitsubishi can construct a Gen III plant. The objective of the Dialoguee Forum is to make an assessment, so for the purpose, the vendor only referred to the US-APWR. **Mr Kolchinsky (Rosatom)** informed that Rosatom has some designs for medium and small power reactors. The medium reactors have VVER technology and the small reactors have other technology. The VVER-640 is a design that worked well and was licensed in the Russian Federation in several sites in the 90s. The medium reactors may not be so efficient as high power reactors from the economic point of view as these designs cost more per MW of power. These types of reactors are therefore better for grids that are not as large as in Russia or the US. **Mr Phillips (IAEA)** considered that the particular question has more to

	do with the economics of the reactor and there is a manual on economics, which is a different part of INPRO methodology.
Mr Arshad (Pakistan) wanted to know from the vendors whether it is purely a market driven decision or the advanced features are economical only at higher powers. If there are many requests for 800 – 1000 MW advanced reactors, whether it will be possible to make these reactors feasible and economic at low powers or whether the advanced features are only possible and feasible at high powers such as 1500 MW or so.	**Mr Phillips (IAEA)** answered that there are some very small units under consideration at the moment that have advanced features but the economics of these are still unknown. In responding to the follow-up statement on the availability of information about SMR on ARIS, **Mr Phillips** responded that the difficulty is that until FOAK is built, it is hard to establish the actual cost. Also, there are site specific issues.
Mr Sato (Japan) noted that the INPRO manual has been developed and prepared as a methodology for comprehensive assessment for nuclear energy systems to confirm long-term sustainability in the 21st century. Therefore, the content is very detailed and complicated. However, the degree of details of a NES design varies at the design stage of NES, e.g. preliminary, conceptual design, detailed design, and basic design for construction and so on. It is difficult to assess some items such as site specific ones when the construction site is not determined, or when the fuel cycle system is not specified. He proposed that a simplified manual should be prepared depending on the design stage, on a selection of table showing items to be assessed depending on the development stage of the design.	**Mr Phillips (IAEA)** noted the importance of the question. INPRO has a two-track process for revision of INPRO manual. In the first track, INPRO does not consider proposals that would in a large way change the concept of how assessments are being done, instead INPRO answers specific technical questions about the current approach. In the track two, INPRO takes these recommendations into consideration and put them in the formal list. Certainly, this is an important question. There is some information about this phase approach and how one might use the existing manual. However, with respect to innovative design where there are tremendous amount of unknown. These will be the subject for future consideration in the INPRO methodology.
Mr Sato (Japan) viewed that the INPRO methodology is too detailed and complicated for technology users and new comers to use it without vendor's cooperation/support. He would like to know vendors' frank opinion.	**Mr Fujii (Mitsubishi)** answered that according to his understanding of INPRO methodology, it is a sort of reference user requirement and assessment of user requirement should be carried out by the user. So vendors do designs according to the requested user requirements and propose a design result to the customer, after which the customer should check based on the requirements. It is the role of the customer to do the assessments, not vendors. **Mr Phillips (IAEA)** added that to the extent possible on the current version of the manual there is considerable emphasis on self-assessment (by the user). In the future this is being emphasized even more. **Mr Lee (Candu Energy Inc)** referred to the concern that everybody has, that the INPRO assessor has to be a subject matter expert in the area. For a lot of evolutionary designs and innovative designs it becomes

	increasingly more difficult for Member States to become subject matter experts in some of the areas. There has to be some reasonable balance found in terms of the extent to which one has to rely on having the necessary expertise and knowledge to be the subject matter expert and to what extent one has to then rely on information able to be obtained from the vendor and take that information more at taste value rather than try to independently assess the validity of the information. For every one of the reactor vendors that the extent to which the R&D programmes that has been developed and the nature of the data that has been collected to support the teachers that has been included in the reactor require a very deep understanding of the underlying phenomena. It is a very big challenge for a lot of Member States to have acquired subject matter experts of the same level of understanding. **Mr Korinny (IAEA)** wanted to clarify that INPRO methodology is a rather flexible tool and INPRO recommends to use this methodology not only to those people who are already experts in this area and not to those countries that have already more or less established a mature nuclear power programme, but also as a tool of familiarization with all sustainability relevant issues for newcomer countries. INPRO assessments can be done on different levels of details and there are examples of these at different detail levels. This means that a newcomer country, depending on the objectives, would be doing either limited or full scope NESA with different levels of details. So this is not only a methodology for experts as it can be used for different purposes. A newcomer country without experts that wants to do a full scope NESA in full depths and analyzing detailed designs will probably fail. The country should be realistic in the plans on how to start the programme. If the country does not have experts, the country should think of how to create a group of knowledgeable people in the country for starting the programme by performing familiarization with the sustainability issue and then limited scope NESA. Later on when there are enough knowledgeable experts, the country can proceed and improve the assessment in more detail. **Mr Miller (GE Hitachi NE Intl.)** added that this is good example of where consultants would be helpful. There are plenty of expert engineers that would be able to help. **Mr Bykov (Rosatom)** indicated that from the point of view of the vendors, the simple answer is that it is impossible to do a very clear and correct assessment without the designers as many user requirements, evolution parameters and indicators need very precise design data. The first step to perform before the start of the assessment is to write the list of questions for the designer to provide initial data. It is impossible to do it through the common knowledge found on the internet. **Mr Kolchinsky (Rosatom)** noted that the assessment and participation of the vendor in the assessment is useful for both parties. However, these assessments require resources and people. The intent for the assessment must therefore be serious, for participation of vendors.
Mr Lignini (AREVA) indicated that perhaps Belarus could tell the forum a good example of how the country implemented the INPRO methodology and how it helped in the decision made to construct a NPP. He asked whether the assessment was done before or after the selection of technology and	**Mr Popov (Belarus)** said that the decision of introducing nuclear power in Belarus was done before the assessment was finished. However, to show economic reasonability of the decision and other areas of the INPRO assessment it was very useful for the country; not only to decision makers but also to public acceptance for the introduction of nuclear power in the country. In response to Mr Lignini's further comment whether the assessment was to confirm the decision already made before, Mr Popov said the assessment showed that the decision was made correctly and useful for the country. To conduct the assessment at a higher level than what had been done in the report,

whether it was useful for its selection, i.e. how the assessment is used in Belarus.	Belarus would need more information from vendors. Unfortunately, one could not always get this. **Mr Korinny (IAEA)** explained that INPRO methodology is not focused on technology selection and was not used for this in Belarus. It may be theoretically possible to be used for technology selection but INPRO does not advise this. In response to comment from Mr Goncalves Filho (Brazil) regarding comparative assessment in the second stage development of INPRO methodology, Mr Korinny explained historically INPRO had steps to try to validate the methodology. It was found that comparison of technology is not the purpose of INPRO methodology. So priority, aggregations, and weighting are not recommended. **Mr Phillips (IAEA)** noted that the IAEA will only discuss matters of international diplomacy, not political or commercial questions. What Member States choose to do is a national decision.
Mr Popov (Belarus) asked Rosatom what the negative consequences are, if any, regarding long term operation of the AES-2006 at reduced power of about 600-700 MW level. For example, whether it would not affect the lifetime of the turbine due to the possible change of steam humidity.	**Mr Kolchinsky (Rosatom)** explained that the derated operation does not pose problems on the turbine but it is only uneconomical to run the power plant in this way.
Mr Dahar (Algeria) asked the state of the art of development of the assessed reactors.	**Mr Miller (GE Hitachi NE Intl.)** said that the ABWR has an NRC certification and has been built in Japan and is also being constructed in Taiwan. For the ESBWR the final NRC design certification is imminent and GE Hitachi NE Intl. expects it within the next few months. The vendor currently has a development agreement with one US utility, and other agreements with 2 US customers applying for licensing. **Mr Kolchinsky (Rosatom)** commented that the new design is under construction in 3 sites in Russia and 1 site in Belarus, and now Rosatom is participating in a bidding procedure in the Czech Republic and Finland. Rosatom has licensing for construction in Russia and Belarus. **Mr Seo (KEPCO-E&C)** explained that 8 units of the APR1400 are being constructed and will soon operate. The licensing in Korea and UAE has been obtained but the US licensing is in the process. **Mr Lee (Candu Energy Inc)** said that for the EC6, Candu Energy has completed all phases of pre-licensing review with Canadian regulators and submitted proposals to build. The EC6 is just waiting for a customer. **Mr Lignini (AREVA)** responded that the EPR reactor is licensed and being constructed in Finland, France and China. It has been licensed in the UK but construction has not started yet. The design certification is underway in the US. **Mr Fujii (Mitsubishi)** explained that for the US-APWR the licensing design certification is underway and close to the final stage. The J-APWR construction permission has been suspended due to Fukushima. Mitsubishi is waiting for new decisions from the nuclear regulatory agency in Japan.
Mr Dahar (Algeria) asked information about quality control of the assessment.	**Mr Phillips (IAEA)** explained that this is up to the assessor since it is a self-assessment. **Mr Korinny (IAEA)** added that when a country is doing the assessment, in particular newcomer countries, INPRO suggested that the country requests peer reviews from the IAEA.
Mr Dahar (Algeria) asked as to what extent the validity of the	**Mr Phillips (IAEA)** responded that in the case of the safety manual most assessments are comparative to previous designs and lineage. So

INPRO requirement value is.	the logic is about making progress in that lineage. In the case of the numerical values, these are taken from the IAEA safety series which is derived from major regulatory positions so INPRO thinks of those as best practice.
Mr Grudev (Bulgaria) asked whether GE Hitachi NE Intl. considers designing the ESBWR with 30-50% load following.	**Mr Miller (GE Hitachi NE Intl.)** answered that their design can follow load although it is not optimum due to fuel economy and the plant was not designed for this. Nuclear should be the base load and among the cheapest power that one wants to run continually.
Mr Banches (Romania) asked INPRO is focused mainly to avoid relocation and or evacuation outside the plant site apart from the measures developed for industrial facilities. He asked the specific NPP parameters to prove this aspect in relocation in relation with the local community from the point of view of technology holders.	**Mr Lee (Candu Energy Inc)** pointed at the key parameters used to govern whether or not to evacuate the populous surrounding a NPP after postulated accidents or long term relocation is based upon IAEA intervention guidelines. When looking at the expected dose the populous may receive within a given time frame, the objective for evacuation is to ensure the populous does not receive a dose between 3-10 mSv integrated over a period of 7 days. If it is projected as possible, then population will be temporarily evacuated and returned once the projected dose drops down to ~1 mSv/month. For long term relocation, the projected dose is 1 mSv incurred over one year. The populous may return if the projected dose is reduced less than 10 mSv in one month. When exceeding LRF safety goal, i.e. inventory of radionuclides will be discharged to the environment, long term relocation would be necessary for some period of time. Specific parameters that are relevant to determine whether temporary relocation is necessary is quantity of I-131 exceeds 10^{-15} Bq such that the populous may receive 3-10 mSv in one week. Since I-131 has a 8.25 day half-life, after 90 days the inventory of I-131 and the associated short-lived radionuclides will be reduced by a factor of 1000 allowing the population to return. For long term relocation, the key indicator is Cs-137 which has a 30 year half-life. If released in the order of 10^{-14} Bq, the population could receive a dose of up to 50 mSv over a period of one year. This is used by the WHO to determine whether relocation is the right measure.
Mr Banches (Romania) asked how the 5 levels of sustainability developed in the consultancy meeting on collaborative project ROADMAPS can be supported by tech holder in national strategy to achieve NES sustainability. Some concept of sustainability can cover some aspects of fuel cycle.	**Mr Drace (IAEA)** considered the question was outside the scope of the meeting. The hypothetical study/concept was developed to better understand GAINS and SYNERGIES study, which requires a separate forum possibly in 2015**.**
Mr Zulquarnain (Bangladesh) asked while selecting technology the consideration of having a reference plant is taken as one of the single most important issue in technology user countries particularly in developing countries with limited technology assessment capabilities. Evolutionary reactors may have very attractive features but because of non-availability of the reference plant this is not considered in the country mentioned above. So he	**Mr Phillips (IAEA)** said the reference plant does not have to be in operation in the country that performs the assessment. For example, in case of Indonesia, the reference plant of AP1000 (as one of NPP options) is in the UK. So it is comparison not within the lineage in the same country, but in the world.

asked how the problem can be resolved.	
One participant asked how to get information on unit cost and electricity generation cost of evolutionary reactor in comparison with SMRs.	**Mr Phillips (IAEA)** explained that the forum tried to avoid comparative discussion on questions on cost. **Mr Korinny (IAEA)** added OECD/NEA had published a report comparing economics characteristics of SMRs. **Mr Phillips (IAEA)** cautioned the huge promotion of SMRs that may not have sound basis. At very low end, there are factory built modular units with no one knows much. Until they are actually built, they should be considered as speculative. **Mr Popov (Belarus)** shared experience of Belarus on unit cost. Initial economic of NES in 2010 used data from investment feasibility study and the final unit capital cost was obtained once contract to build NPP was available. The team repeated the economic study and the final result is in available in the report. **Mr Phillips (IAEA)** added the final cost is known when one actually starts the plant (given schedule delay, etc).
One participant indicated the evolutionary reactors presented in the forum have a design life of 60 years, and asked the basis of 60 years.	**Mr Seo (KEPCO)** explained the commercial power plants were designed for 40 years. In the US the life extension starts at 20 years. So the logical next life is 60 years. If longer lifetime, there is need to evaluate life-related factors in component and system design and this will cost more.
Mr Reyes (Chile) asked if INPRO methodology can be used for potential scenario in the country to check the advantages and disadvantages of the different types of reactor technologies in the scenario.	**Mr Phillips (IAEA)** explained INPRO does not propose this approach because INPRO does not technology comparison.
Mr Dewayatna (Indonesia) asked that since ESBWR has no secondary loop, in case the fuel rod fails the steam going to the turbine is contaminated with fission products. The risk of release of radioactivity is increased. He asked the lowest value of calculated probability of radioactivity release to the environment.	**Mr Miller (GE Hitachi NE Intl.)** noted the figures were not available in the meeting. Radiation in the main steam / feedwater steam is monitored. When there is a large increase in radioactivity which would indicate fuel failure, then immediate closure of the main isolation valve to contain. Some contamination would get to the turbine but will be bypassed to condenser, retained and scrubbed. Recent experience with BWR fuel has been excellent, and there have been few cases of fuel bundle/pins leaking.
Mr Dewayatna (Indonesia) asked vendor of EC6 pointing that to achieve economics, most power plants raise power output. Can the same be done to EC6 to further improve economics?	**Mr Lee (Candu Energy Inc.)** said that EC6 is designed only for 700 MWe and the electric output can be increased by a few percentages. For 1000+ MWe output, the vendor can provide design similar to those at Darlington or Bruce site which have 480 fuel channels instead of 380.
Mr Calabrese (Argentina) asked cyber security is an item to be included in the physical protection of the INPRO methodology. He also asked whether the INPRO methodology has covered natural disasters like tsunami	**Mr Phillips (IAEA)** said that it is already in the physical protection manual but since it has brought up several times INPRO will make sure the discussion is sufficient. On natural disasters, there is a learning process in any industry, as one said every 7 years we learn some new major things about safety in LWR. This is also true in regulatory system where the regulators can only regulate against problems that they are aware of. INPRO methodology cannot be more pre-essence than any other system and will include new issues. Tsunami is a site evaluation problem and in generic sense, it is a natural hazard which is already covered in the methodology.
Mr Dulera (India) asked about with an increase in the level of	**Mr Lignini (AREVA)** said that vendors are very confident because vendors take into account in design all situations that have been

NPP safety, how confident we are in locating power plant near population centers, which is a typical goal for innovation design. This is a problem for countries with high population density.	predefined as well as specific requirements. So design of reactor is made to fulfil the objective of very limited consequences in case of accident. So it is more of political issue than technical issue. For example, for EPR, the consequence for accident is very limited and does not necessitate evacuation of population nearby. When further asked whether the regulator also feels the same confidence, Mr Lignini believes so otherwise regulators would not license the reactor. In his opinion, it is a matter of national policy. **Mr Seo (KEPCO-E&C)** indicated that no regulator defines the location of a new NPP based on the amount of radioactive materials release by certain design. **Mr Lee (Candu Energy Inc.)** shows confidence the reactor designed today can be located next to the population. CANDU reactors have been designed that way since 1970s, where the 500 MWe class reactors have been sited at Pickering near Toronto. The vendor is confident that in any postulated severe accidents there will be very minimal impact on the environment and populous.
Mr Zhou (China) asked regarding assessments of NES, whether the assessment shall be done by users and technology holders only provide necessary design data, or the assessment can be done jointly by users and vendors. He wanted to know which of these ways would draw convincing assessment conclusions.	**Mr Phillips (IAEA)** noted the answer referring to the introductory presentation. It has more to do with the efficiency of completing the assessment. Theoretically the assessor can work to the same standard but this implies that the assessor has to become an expert on the technology. This depends on how much of the information is available publically. In most cases there is sufficient information for safety experts to do an INPRO assessment to very good quality. In other cases it may be impossible and the assessor has to work with vendor. From the point of view of efficiency, if the assessor works with the vendor from the beginning it will be much easier as the country does not have to build an expert base.
One participant asked about licensing issues that may arise for innovative designs, e.g. any aspect of design innovation that may not be easily acceptable to a licensing authority	**Mr Lignini (AREVA)** admitted indicating there are aspects that may be of concern to regulatory body, e.g. digital I&C. It is always a sensitive matter to regulatory body with respect to the reliability of software for programming and pure use of activation within the digital I&C. **Mr Phillips (IAEA)** provided a generic example on significant innovation to put a large number of NPPs within a single building with one control room. This is new to regulation and the regulatory answer to this is not yet known. It is an R&D problem. With the current regulation on number of staff, the design will requires 30-40 staff in one control room. He predicted in the next 10 years very completely new questions may be raised by new designs.

7. STATEMENTS FROM MEMBER STATES

Member States were asked to deliver a statement on the following six issues:

Q1. What is your country's plan in terms of deployment of nuclear reactors in the next 10 to 20 years?

Q2. Do you discuss in your country / organization the sustainability of your national nuclear energy system?

Q3. Did you get useful information from this Dialogue Forum as you had anticipated?

Q4. How would the information from the Dialogue Forum be used in your country / organization?

Q5. Aside from safety of nuclear reactors, there are other design-related issues in the INPRO Methodology e.g. waste minimization, environmental stressors. Do you have any specific interests in these areas?

Q6. Do you have any recommendations for follow-up Dialogue Forum meetings?

7.1. Algeria

1. The Algerian Minister of Energy and Mines announced last June 2013 at St. Petersburg that the country is planning to have its first NPP around 2025. This will be followed by 1 NPP for each five years to reach the goal of 13% of nuclear in the energy mix.
2. Generally the sustainability of a nuclear energy system is usually discussed when talking about including nuclear energy in the energy mix. But up to now there is not a full study of the whole energy systems that have to be considered in long term planning.
3. The dialogue forum was very interesting. The applicability of the INPRO assessment methodology in some pertinent area of safety has been demonstrated along with some drawbacks which have to be improved in the forthcoming revision of the methodology.
4. The information from the Dialogue Forum will help to:
   - know how to apply the INPRO methodology and get familiarized with
   - learn how new safety issues can be handled by designers/vendors of evolutionary reactors
   - understand the challenges and proposed solutions
   - get more information about the future market trends of NPPs
5. Algeria is interested in infrastructure, economics and security.
6. Algeria suggested human resources development for evolutionary power reactor including SMRs in order to build an appropriate infrastructure that can handle new NPP projects.

7.2. Argentina

1. In the short term respected current plans and recent announcements of the Minister of Energy Planning up to 2023 include incorporating the 4th NPP comprising 2 HWR modules of 750 MWe each and the 5th NPP comprising 1 or 2 LWR modules of 1,200 MWe each to the electricity net. The objective is to reach 30% of nuclear energy participation in the final energy demand, considering a diversified energy mix formed by 30% of nuclear, 30% fossil, 30% hydraulic and 10% of other renewable energies.
   In 2014, Embalse NPP will start life extension programme for several years.
   - In 2018, CAREM 25 will start operating.
   - In 2019, Atucha I will start with the lifetime extension programme and then restart in 2021.
   - In 2020, CAREM 150 will start operating along with the first out of the two 750 MWe reactors that will be part of the 4th NPP. The second reactor will start operating in 2022.
   - In 2023, the 5th NPP will start up with a 1200 MWe reactor (This value of power was adopted instead of a generic 1000 MWe in corresponding to the speech delivered by the Minister of Federal Planning, Public Investment and Services of Argentina in Saint Petersburg in June 2013).
   - In 2027, another CAREM 150 MW will start operation.
   - Atucha I and Embalse would be permanently shut down in 2046 and 2050 respectively.
2. Argentina considers the efficient use of uranium and minimization of wastes. Argentina has uranium resources.
3. Yes
4. Information is shared with technical groups.
5. Yes
6. No recommendation.

7.3. Armenia

1. Armenia will be constructing one new unit with a capacity of 1000 MWe of AES-92 type.
2. Yes, last year Armenia adopted sustainability of energy development concept and discussed future development in this context for the deployment of AES-92.
3. Yes, Armenia received much information from vendors about news unit and experience dealing with other user's development of NES in countries.
4. The Dialogue Forum was a very good area for exchange of information or receiving information on future deployment of NPPs.
5. For Armenia, the important areas are safety and waste management area. Economics has been understood.
6. IAEA arranged and organized the Dialogue Forum very well and there are no gaps.

7.4. Bangladesh

1. According to the present plan (Power System Master Plan), Bangladesh is expected to have a generation capacity of about 2,000 MWe and 4,000 MWe from nuclear by the next 10 years and 20 years respectively.
2. Yes but need to consider in a more organized way, maybe by conducting a NESA.
3. Yes, with a lot of valuable information and a great opportunity to hear directly from technology developers.
4. The information gathered from the Dialogue Forum will be shared with the top management as well as with the relevant divisions/professionals of the organization so as to facilitate decision making with regard to the deployment of evolutionary reactor in the country.
5. Bangladesh is interested in those areas (waste minimization and environmental stressors) as well. Being a densely populated country, Bangladesh has particular interest in understanding the impact of evolutionary reactors on the neighbouring population during normal operation as well as during abnormal and accident conditions including BDBA.
6. Competent human resource is one of the most important key elements for sustainability of NES. As such, the Dialogue Forum may consider the issue of HRD as well as knowledge retention in the area of NES.

7.5. Belarus

1. Belarus plans to introduce nuclear power sources are known and consist in introduction of two units of AES – 2006 design in 2018 and 2020. Although optimization studies of the structure of the Belarusian energy system show the economic feasibility of the introduction of another same unit until 2035, but such a decision is not considered yet in our programs of energy system development.
2. Effective work of the NPP in economic terms is, in our opinion, one of the conditions for sustainable development of NES. Therefore, in the discussions that take place in our institution and beyond, at the Ministry of Energy, for example, we try to prove the necessity of nuclear power plant operation at maximum utilization rates of its installed capacity, which are achieved in the project. There are several obstacles to this, for example, an overabundance of gas facilities, a large share of cogeneration energy sources, etc. Therefore, the efficient operation of NPP is only possible if Belarus energy system will be adapted to introduce large nuclear energy sources.
3. At the forum Belarus has received a lot of useful information. Information from Russian colleagues regarding AES-2006 was particularly interesting for Belarus. Presentations of other nuclear energy technology holders also were extremely interesting from a cognitive point of view and gave excellent examples of the use of INPRO methodology.
4. Information from Russian colleagues regarding NPP-2006 will allow Belarus to considerably complement research using the INPRO methodology. Familiarity with nuclear technology through direct contact with their holders can be useful in the search for further ways to develop nuclear energy system.
5. Areas of special interest are still the economy, the management of radioactive waste including spent nuclear fuel, and manpower resources.
6. It may be advisable that one of the future forums could be devoted such an important area as radioactive waste and spent fuel management. The main recommendation is to continue INPRO activity with the same spirit.

7.6. Brazil

1. First to complete the construction of Angra III, rescheduled for 2018. Next to build one or two new NPPs of approximately 1000 MWe in the north-eastern region of Brazil. Site location has been started.
2. Currently discussion of the sustainability of our national NES is restricted to the INPRO areas of economics, safety and waste management.
3. Yes, not only from the formal presentations and discussions but also from parallel conversations held during the meeting intervals.
4. The information would be shared with other interested organizations of the national nuclear sector through the usual channels.
5. Our main interests are in the areas of economics of NPPs and high level waste management.
6. Recommendations are to distribute the proceedings of the Dialogue Forums as working materials for the follow-up Steering Committee meetings (SCM) and its Conclusions and Recommendations as an item of the SCM agenda to make decision on evolution of program.

Additional remark: By comparing a potential Innovative Nuclear Energy System (INS) candidate to existing, the current version of the INPRO methodology in the area of reactor safety is not making a case for the

sustainability of nuclear energy (the main objective of the INPRO methodology) but measuring progress of evolution of the nuclear technology. The secretariat should consider at least track 2 of the revision of INPRO methodology to review safety area, perhaps along the line of the current UR 1.5 of the safety area on release into environment.

7.7. Bulgaria

1. Bulgaria considers further using of nuclear energy based on extending of life of units 5 and 6 and building of new units. Decision is taken to build Unit 7 at KNPP site, and also option is discussed for restarting the BNPP project.
2. In the National Energy Strategy is written that nuclear energy will be supported institutionally not only as a promising resource for generation of emission-free electricity, but also due to the accumulated successful experience and professional capability for operation of nuclear capacities.
3. Receiving information for innovative nuclear reactor systems directly from vendors is helping in developing and deploying of nuclear energy strategy including dynamic evolution.
4. The information will be used for discussion of different options for further developing of nuclear energy strategy. It will be useful in the university / education, since until now only VVER technology is used in the country.
5. Yes. Spent fuel management, impact to environment and human resources.
6. Bulgaria recommends INPRO to continue with organizing such Dialogue Forum meetings by inviting vendors.

7.8. Chile

Chile to understand INPRO methodology and explores possibility to use NESA as a tool to study nuclear power options.

1. Chile has energy policy aspects such as sustainability, mixed energy, improved competitiveness of electrical market and environmental aspect for development of energy mix in the next 20 years. The electrical market is 100% handled by private companies. The decision of technologies to be introduced in the mixed is taken by Gencos under the local market conditions and environmental assessment studies. The role of government is to guide the market with technical and legal framework and assure low cost of electricity for the final consumers based on model of economical low dispatch at marginal cost in a competitive market. Under this market scenario, nuclear power is considered as alternative in the next 20 years. One incentive is to improve the level of knowledge in nuclear technology aspects and to compare technologies available. This aspect is necessary to develop appropriate legal and technical framework for possible NPP in Chile.
2. Sustainability in energy level in the energy sector, like capability of accepting technology with the highest standard and environmentally friendly. So in this context nuclear power is a real option. The sustainability of nuclear power covers aspects such as nuclear waste volume and management option, low emission, low cost of electricity and strong nuclear safety.
3. The information is useful to understand current technologies available in the market.
4. The information will be used to understand technical aspects and reactor design in the current technologies in the new reactor design.
5. -
6. Future Dialogue Forums on Fukushima lessons learned and improvement of safety and include impact on levelized cost.

7.9. China

1. There are 15 NPP units in operation in China now. The capacity of nuclear power is about 2% of total electric capacity. Up to 2020 there are about 58 units in operation and 30 units on construction. All those units are thermal reactors. Based on strategy study of nuclear energy in China, there will be about 100-200 units in operation in 2030, and the proportion of the capacity of nuclear power will reach average value of the world. Among these hundreds units, there are at least 2 sodium cooled fast reactors. During 2020-2030, a partly closed fuel cycle system will be on demonstration operation. The fleet of NPPs would be very big and with various reactor types in the future.
2. The sustainability in China is assessed in 3 levels:
   - Level 1: The China academician associations of engineering and science have finished strategy study of nuclear power development for 2030 and 2050 in China. The main topic of study is how to

develop nuclear energy on sustainability, and when and what kind of the closed fuel cycle system would be established.
   - Level 2: Based on those studies, China national energy agency draws up development strategy and plan of nuclear energy. It contains items of uranium supply, fast reactor and reprocessing plant, to assure the nuclear energy will be developed in a large scale and on sustainability.
   - Level 3: China institute has finished the study on technical roadmap and cycle mode of closed fuel cycle. The suggestion of how to use depleted uranium, recycled uranium, plutonium and minor actinides, adopt what kind of reactor and fuel type, use what kind of reprocessing technology has been provided to administration agency in charge of R&D.
3. China has finished INPRO assessment in the area of safety of a fast reactor conceptual design, and proliferation resistance assessment on a fast reactor and its fuel recycle system using INPRO methodology. There are questions and puzzles in mind. From this Dialogue Forum China learned how to expand in detail when assessing, and to select what kinds of parameters or systems to assess to certain UR.
4. The working team in CIAE will assess again on safety and proliferation resistance of fast reactor design using the information from the Dialogue Forum based on enough new design data. Then, CIAE hopes to expand working team and assess other areas such as economics, waste Management.
5. Except safety and proliferation resistance, the areas of economics and waste management are the major concerned aspects or topics to fast reactor and its fuel cycle. CIAE should do these two areas assessment, to obtain relatively whole conclusion.
6. China recommends to assess one area such as economics, but on all BPs, URs and Ins and select only one kind of design for whole assessment, or select several designs referring to certain BP, UR and IN, and the assessment on several designs would cover all BPs, URs and IN.

7.10. Egypt

1. Egypt has a peaceful nuclear power program to introduce a series of NPPs by 2022 with diversity of power generation resources.
2. Yes, nuclear power system is considered in energy planning studies as a candidate for energy demand expansion. There is cooperation and coordination among Egyptian organizations such as NPP authority, regulatory body and electrical and energy production. Egypt issued its Nuclear Law for the Nuclear & Radiological Activities in 2010.
3. This information gives us a vision of the current and future trends in nuclear technology related to NPPs and nuclear fuel cycle considering INPRO documents.
4. The information can be used to overcome the challenges faced during execution of nuclear power programs such as design, contracting, management, and financial issues and can be adapted with R&D related to INPRO activities.
5. Some issues related to INPRO methodology such as economics, waste management, and infrastructure are essential focal points to be considered.
6. Egypt recommends the forum should stress on enhancing nuclear safety with result from Fukushima accident.

7.11. Germany

1. Germany is in phase out. There were 17 reactors in operation, after Fukushima 8 reactors were immediately shut down, and others will be shut down by 2020. The plan is to continue to emphasize on renewables and by 2050 80% is generated by renewables. It is not clear how it is done considering renewables are intermittent.
2. No
3. The information is useful, with real serious amount of technology work and construction program in various MSs.
4. This is to be reported back to the government that nuclear energy is continuing with more countries relying on nuclear energy. Germany should reconsider the decision taken.
5. Germany is interested in waste management and looking for new options for waste storage. The salt dome has been put on hold and will be discussed again.
6. Germany recommends Dialogue Forum on final waste storage.

7.12. Hungary

1. Currently in Hungary about 40% electricity generated is by nuclear, 38% by gas, 18% by coal and the rest 4% is by others. The capacity of NPP is 2,000 MWe from 4 units of VVER-440, each with a capacity of 500 MWe. The operating time of existing units was finished by 2009, the parliament took note of the lifetime extension so the operational lifetime has been changed from 30 to 50 years. The operational time will expire between 2032 and 2037. Working and licensing of the lifetime extension is on-going. Currently the 1st unit has received the license for lifetime extension for 50 years. For the 2nd unit, the application for lifetime extension has been received by Hungarian atomic authority recently. In 2009 the parliament passed preliminary permission for preparatory work for construction of new units at existing site. The scenario is to construct 2 units each with 1,000 MWe capacity from Gen 3+ by 2030. The reactor type and the number of new units have not been decided yet. Between 2030 and 2032 the total capacity of electricity generated by nuclear will be 4,000 MWe. From 2032 the existing units will start to shut down, and from 2037 the capacity will be 2,000 MWe again. In addition to NPP, the national NES consists of spent fuel temporary storage near NPP with a lifetime of 50 years and the site for final disposal is being prepared; geological disposal facility from low level waste (LLW) and intermediate level waste (ILW) originated from NPP, there is a near surface disposal, also for ILW and LLW from industry and hospital.
2. For authority, it is important to ensure safety and reliability. Furthermore the design and construction requirements for construction of these reactor types are done by other big authority e.g. Russian and American, which can be incorporated into national regulation considering the state legislative.
3. The information about new reactor types is very useful.
4. Presentation and documents will help the delegate working with the authority.
5. There is no recommendation for a follow-up Dialogue Forum.

7.13. India

1. India has a unique, sequential three-stage nuclear power programme based on closed fuel cycle. The programme is aimed at optimum utilisation of India's nuclear resource profile of modest uranium and abundant thorium. The spent fuel of each stage is reprocessed and acts as resource for the subsequent stage. The first stage is mainly based on natural uranium (0.7% ^{235}U content) - fuelled PHWRs. In the second stage, plutonium and reprocessed uranium from the spent fuel of PHWRs would be used for fuel fabrication of Fast Breeder Reactors (FBRs), including metallic fuel once, in the second stage. During the later period of second stage, thorium would be put in the blankets of FBRs so as to obtain ^{233}U after reprocessing. Subsequently, the third stage systems would be based on ^{233}U-Th fuel. When the spent fuel is reprocessed to produce fuel for the next stage, it multiplies manifold the energy potential of the fuel and greatly reduces the quantity of waste. It is thus an optimum solution for meeting the country's energy needs in a sustainable manner, securing its energy freedom in the long term. Currently India has 21 operating reactors with about 5,800 MWe installed capacity. 18 of them are PHWRs, 2 are GE BWRs and one is recently commissioned VVER. The first stage, with many PHWRs in operation and many in construction and planning stages, has reached a state of commercial maturity. There are 4 PHWRs with total of 2,800 MWe capacity under construction, and each year during 2015 and 2016, two are likely to be commissioned. The second unit of 1 GWe VVER is also likely to be commissioned during next year. India has plans to increase its nuclear power installed capacity to 21 GWe by 2020. This includes about 10 GWe imported LWRs. The first commercial 500 MWe FBR of second stage is at an advanced stage of construction and is likely to achieve criticality around end of next year. Besides these reactors, India is also developing innovative reactors mainly based on thorium-based fuels. Construction of 300 MWe Advanced Heavy Water Reactor (AHWR) is likely to be started within next few years. High temperature reactors, such as Compact High Temperature Reactor (CHTR) and Innovative High Temperature Reactor (IHTR), being developed with an aim to produce nuclear hydrogen by high efficiency water splitting process, are at various stages of technology development, and are likely to be commissioned by early next decade (2020-2025). Besides, there are plans to set up research reactors (including high flux research reactors) for multiple applications, including material and fuel irradiation studies.
2. The concept of sustainability is inherent in the Indian three stage nuclear power programme and is the central most important criteria. Accordingly, India has adopted closed fuel cycle. Analysis for key parameters such as 'energy produced per tonne of mined uranium', and 'waste generation per unit energy produced' etc. are regularly done.

3. Yes, the dialogue forum was useful in enriching knowledge of application of INPRO methodology. Earlier INPRO methodology had been applied for India's innovative reactor concepts, which are FOAK reactors. Information about evolutionary reactors from major vendors was additional advantage.
4. This will help in India's own assessments, since the Dialogue Forum provided exposure to detailed assessments of many evolutionary reactors.
5. All INPRO areas are of interest but the most important ones being: Safety, Economics, Waste Management, and Environment
6. Separate Dialogue Forums on assessment of individual INPRO areas with its applications to few sample cases would help Member States in better understanding about the methodology and in their assessments.

7.14. Indonesia

1. The use of NPP is mandated by the national law term development plan which state nuclear electricity. Nuclear should commence by 2019 but government decision is yet to be made. Regardless the decision, the development on nuclear infrastructure such as site and non-site feasibility studies, HRD including NESA and regulatory aspects are ongoing. Under current national energy policy, the role of new and renewable is to be promoted. Currently the role of renewable mainly hydro and geothermal is 5% while the target in 2025 is 23%. The gap should be filled by new energy such as nuclear, coal bed methane, and liquefied and gasified coal.
2. Yes, Indonesia started NESA in 2011. The objective is to evaluate sustainability for planned NES, find gaps or unconformities and ways to resolve the gaps.
3. Yes, Indonesia obtained very useful information from the IAEA and vendors to help its NESA team to make the assessment, and correspond and relate to vendors and other countries to discuss nuclear energy.
4. The information will be used to complete Indonesia's assessment on planned NES, which has progressed quite well, using INPRO methodology
5. Indonesia is interested in safety of nuclear fuel cycle facilities such as conversion and fuel fabrication, interested in NPP design that can accommodate high local content and local participation, with simpler design and shorter construction time to meet infrastructure and economics principles on NESA using the INPRO Methodology.
6. Indonesia recommended the IAEA facilitate holding more intensive dialogue with embarking countries and vendors / technology holder to improve capability in NES which in turn will accelerate NPP deployment in embarking countries.

7.15. Japan

1. Japanese government is currently reviewing from scratch the policy to enable zero operation NPPs in 2030's decided in September, 2012 by the former administration of the Democratic Party of Japan, and developing the new basic energy plan with responsibility from the view point of securing stable energy supply, reducing energy costs, and so on. It is uncertain whether the new basic energy plan including nuclear energy will be announced by the end of December 2013 as planned, because it is not known how many reactors, including 3 units under construction, can be brought back online after passing the safety review under the new regulatory requirements for commercial NPPs which was taken effect on July 8, 2013. It is not clear whether construction of new reactors will be stated in the new basic energy plan. This new regulatory requirements cover:
   - severe accident measures
   - measures against large-scale natural disaster such as tsunami and earthquake, terrorist attacks including aircraft crash and the like
   - prohibition of construction of reactor buildings and other key facilities above active faults based on the criteria
   - 'back-fitting' system where the new regulations are applied to the existing NPPs.

   Currently all NPPs, 50 units in total are offline. TEPCO has decided to use the intact Unit 5 and 6 in the Fukushima Daiichi nuclear power station as research facilities for decommissioning, not as power reactors, responding to the request by Prime Minister Abe, which TEPCO should concentrate on the decommissioning work of Unit 1 to 4. Safety review of 14 units is being conducted under the new standards. The Nuclear Regulatory Authority (NRA) determined that the faults running under the two sites, Japan Atomic Power Company's Tsuruga NPP and Tohoku Electric Power Co.'s Higashidori NPP, are active, resulting in dispute with the electric utilities. Further two NPP sites are being investigated for

active fault lines. The new regulatory requirements for nuclear fuel facilities, research reactors, and nuclear waste storage/disposal facilities will be issued in December 2013. It is hoped that the new plan stipulates the nuclear energy utilization with closed fuel cycle including fast reactors.

2. Yes. Since Japan is a country with scarce national resources, it is necessary to secure long-term domestic energy resources. Therefore, Japan has been developing a closed nuclear fuel cycle with fast reactors since the beginning of nuclear development as a national policy. This new basic nuclear policy is under discussion due to the Fukushima accident, but the representative believed that Japan will go back to the policy giving importance to the closed fuel cycle with fast reactors in the future.
3. The representative could recognize the outline and features of evolutionary nuclear power reactor design being developed including assessment results of INPRO safety criteria by seven vendors. Vendors' and INPRO member states' comments on the INPRO Methodology such as its usefulness, convenience in the Dialogue Forum are very important and will be very useful for the future review and revision of the INPRO manual.
4. The representative will share the information from the Dialogue Forum with related persons in Japan Atomic Energy Agency (JAEA) and ministries.
5. The representative is interested in waste management, environment and economy covered by the INPRO methodology.
6. In order to familiarize INPRO MSs with the details of INPRO methodology, assessment results of reference plants throughout the safety area, for example, should be presented and discussed in the Dialogue Forum.

7.16. Jordan

1. Jordan has identified in its short term energy strategy, in the next 10 years, the role of different available technologies, including nuclear, in meeting its near future energy needs. This will ensure that nuclear energy will be available to contribute to meeting the energy demands in Jordan, shortly.
2. Achieving nuclear energy sustainability would require an expansion of future NES in Jordan to constitute the electricity mix in the national long term energy strategy. Therefore, national effort should be accumulated for developing nuclear energy strategy towards sustainability.
3. This Dialogue Forum was a good opportunity to be familiar with the INPRO Methodology and how much it is useful to use this methodology to assess the sustainability of the future NES in Jordan. Furthermore, the demonstration of the use of the INPRO Methodology to assess the safety of reactor design conducted by different reactor vendors was a unique experience.
4. In Jordan, there has not been any discussion yet within Jordan Atomic Energy Commission or at national level in details the sustainability of the planned NEs. However, with the assistance of the IAEA and the selected vendors and upon the availability of the necessary resources, Jordan is looking forward to using the INPRO Methodology to assess the sustainability of its planned NES in different areas of interest such as safety, infrastructure, physical protection, proliferation resistance as well as economics. In addition, other areas of interest such as waste minimization and environment will be assessed using the INPRO Methodology.
5. Developing the capacity building of Jordanian experts, identifying gaps in the Jordanian nuclear power program, developing technical document to support decision making and identifying potentials for regional and international collaboration, all of these benefits will be obtained from performing an assessment of our national nuclear energy systems using the INPRO Methodology, hopefully, in the near future.
6. The representative recommended INPRO Group to organize in the future a Dialogue Forum in the area of the INPRO Methodology application in the newcomer countries: Challenges and Feedback.

7.17. Kazakhstan

1. Construction of NPP in Kazakhstan is on the stage of consideration at the moment. Country supplies electricity by itself. However, trend of increasing energy demand forces to plan ahead, considering energy diversification including NPP sources.
2. Establishing energy sustainability is essential for Kazakhstan. Nuclear energy is one of the main options to achieve following objectives: (i) Energy diversification, (ii) Optimisation of fossil fuel and hydrocarbon resources usage, and (iii) Attraction of high-technology manufacturing and innovation
3. Information learned from the Dialogue Forum has been useful and fulfilled expectation and gaps.
4. After the decision on NPP construction, Kazakhstan intends to apply INPRO Methodology and carry out detailed work with the IAEA.

5. For mentioned directions, the country is open to collaborate in a frame of INPRO Methodology after decision on NPP. Meeting format has met all the expectations.
6. The recommendation is to involve participants very quickly with discussion, it would be useful to give information about criteria of the INPRO methodology for vendors.

7.18. Kenya

1. Kenya is in the process of undertaking a milestone approach to nuclear power production within the next 17 years. A pre-feasibility study has been conducted and a strategy developed.
2. Yes.
3. Yes.
4. It will be helpful in deciding an appropriate reactor model.
5. Yes, Kenya is interested to know more about environmental stressors.
6. Kenya recommends to focus also on environmental stress factors.

7.19. Korea, Republic of

1. The basic plan for long-term electricity supply and demand is announced biennially by the government. Based on the 6th plan, the newest plan announced in February this year, nuclear capacity will increase from 20.7 GWe to 35.9 GWe in the year of 2027 so approximately 11 additional units will be needed.
2. Korean government is going to update the national energy plan next year. For that purpose, the public debate is going on. The share of nuclear power is one of the most controversial issues in that debate, especially about the safety and high level waste management including nuclear spent fuel.
3. It was a very valuable chance to enhance the knowledge about the advanced feature of various evolutionary designs from vendors.
4. The information will be good references in the debate regarding nuclear safety.
5. As already mentioned earlier, managing high level waste in a sustainable manner is a crucial issue in Korea so the country is highly interested in minimizing the waste.
6. Korea hopes that the Dialogue Forum will play as a good venue to share the ideas about common sustainable nuclear future. (On being questioned by Russian Federation and Chairman whether waste will be supported by Korea as topic for future Dialogue Forum, the representative will report to the decision maker in Korea).

7.20. Malaysia

1. In 2006 the government of Malaysia has agreed to include nuclear as a possible energy source post 2020. The decision was made in view of depleting indigenous sources namely gas and oil as well as increasing dependence on imported coal. In 2011 NEPIO was formed to undertake study on NPP deployment so that government can make informed decision as well as develop the Nuclear Power Infrastructure Development Plan (NPIDP) and Regulatory Infrastructure (NPIRDP). A new nuclear bill has been drafted and proposed to be passed by Parliament in 2014. The NPIDP is supposed to be completed by the end of 2014. The government decision on whether to go nuclear or not will be based on the report.
2. In 2010 the government has produced a blueprint with an Economic Transformation Programme (ETP) to increase the per capita income of the country and to make Malaysia a developed economy by 2020. Numerous high impacts Entry Point Projects (EPP) were proposed including NPP as one of the EPP. It was believed that NPP can sustain the energy needs of the expanding economy as well as enhanced the technological development and services industry in Malaysia. EPP is planned to bring jobs as well as increase the Gross National Income.
3. The Dialogue Forum has brought new knowledge especially on various designs of NPP and how the safety features incorporated made the plant withstand severe accident postulated either due to natural causes or man-made.
4. The information will be used in completing and input to the NPIDP report. The knowledge will also be used to understand more of INPRO methodology and to propose workshop on the subject including NESA.
5. Waste from NPP has always been brought up by the public in any meeting or session with regard to nuclear energy. The subject related to innovative fuel cycle especially if it can reduce the amount and toxicity will be very advantageous to the industry. Since thorium fuel cycle is said to be more efficient and generates less waste as well as more abundance, INPRO should put greater initiatives into the subject.
6. Malaysia recommended the followings:

- Thorium as a future energy resource material
- Public information and acceptance (PI/PA) is a very important factor to ensure successful and sustainable implementation of nuclear energy. Therefore INPRO should also include PI/PA on the innovative nuclear energy initiatives.
- Building a strong basis for future NPP program (infrastructure, feasibility study, assurance and sustainable fuel supply and fuel cycle, waste management)

7.21. Nigeria

1. The decision to explore nuclear energy is Nigeria has large deficit in electricity production. With less than 40% of the population have access to electricity, the country is one of the lowest electricity consumption per capita in the world. To support nuclear power in the next 20 years, the government has formally inaugurated the Nigeria Nuclear Regulatory Authority, energy commission of Nigeria, Nigeria Atomic Energy Commission including some institution. The Nigeria Atomic Energy Commission is the focal point of all nuclear activities in the country. In terms of development of nuclear energy:
   - A National Roadmap for Nigeria's Nuclear Energy Programme has been developed and approved by government.
   - A draft law for the implementation of the national nuclear energy programme has been developed, and subjected to detailed scrutiny by all major stakeholders with technical input of the IAEA.
   - The law establishing the Nigeria Nuclear Regulatory Authority is being revised and modified to adequately empower the agency to effectively regulate and license NPPs (following effort to acquire small research reactor). Currently it is under consideration in the National Assembly.
   - A draft regulation on the Safety and Regulatory Requirements for Licensing of Sites for Nuclear Power Plants has been developed.
   - A National Policy on Radioactive Waste Management has been finalized by NAEC in consultation with the Nigeria Nuclear Regulatory Authority and other stakeholder institutions.
   - Ratification and domestication of relevant extant international statutes, treaties and conventions.
   - The framework for the establishment of a national nuclear insurance policy and scheme to adequately address the civil liability component of the nuclear power industry is being developed in conjunction with Ministry of Justice.
   - Capacity development. Two years ago the commission recruited young scientists and engineers for training in nuclear science and engineering. The first batch of trainees is now concluding their Master's programme in Nigerian universities.
   - Identification of possible sites for NPP. Preliminary studies concluded for 6 sites, now narrowed down to 3 potential sites.
   - Public acceptance through awareness programmes, seminars and workshops.
   - Energy system analysis has been carried out using IAEA tools – MAED, MESSAGE etc, by the Energy Commission of Nigeria. Efforts are now being made to revise the analysis.
   - Nigeria is expected to have the first NPP in operation in the early 2020s.
2. On sustainability of Nigeria nuclear power program:
   - Long-term national and political commitment to the nuclear power programme and its sustainability. With political development in Nigeria, it is hoped the government will support the program.
   - How to retain the current crop of young scientists and engineers being trained – in case there is a large gap.
   - Nigeria is an observer of INPRO and needs to perform at least a limited-scope NESA. To do this, Nigeria needs to join the INPRO community. Chairman of NAEC indicates positive steps taken so that Nigeria will become a full member of INPRO.
3. Yes. The information obtained here will serve as mind opener when dealing with issues on choice of technology and will be useful in discussions with potential suppliers.
4. Information from the Dialogue Forum will be disseminated to larger fora through seminars and workshops. On the overall, Nigeria would like to use INPRO methodology as a tool for development of its national nuclear energy programme.
5. Nigeria also has interest in waste minimization and environmental stressors.
6. Nigeria recommends a follow-up dialogue forum on back-end fuel cycle.

7.22. Pakistan

1. The government has advised to raise the nuclear power generation from currently ~750MWe to 8,800 MWe (or more if possible) by the year 2030. Half of this can be absorbed overnight by Pakistan's power

hungry grid. For this purpose, 3 small reactors are already in operation and two are about to be commissioned in 2014, and the rest are in planning stage.
2. Sustainability of national nuclear energy system is discussed with respect to availability, economics and safety.
3. The interaction with the vendors did provide useful information regarding new design trends and application of those in improving the safety of nuclear reactors.
4. The information obtained during the Dialogue Forum will provide background for selecting future NPPs.
5. Waste minimization is one of the most important aspects of nuclear energy.
6. At least one more Dialogue Forum should be held with the vendors on the topic of waste management and minimization. If possible, some information provided by the vendors may be provided to the participants in advance so that participants can interact in a more conducive way.

7.23. Poland

1. In the next 10-20 years Poland plans to build 2 NPPs, each comprising of 2-3 units, of total power of 6 GW(e). The start of construction of the first unit is planned for 2019; commercial operation is estimated for 2025.
2. With reference to the sustainability of the future Polish nuclear energy system it has to be pointed out that at present most of the production of electricity in Poland is based on coal (more less 90%). Poland is aware that the situation has to be changed mainly because of the level of emission of main air pollutants from coal power plants and also of the security of energy supplies reasons. That is why Poland plans to diversify sources of energy and reduce dependence on coal by increasing share of both renewables and nuclear power in energy mix. According to the long-term strategic programme called "Energy Policy of Poland until 2030" the contribution of nuclear energy to the yearly electricity production should reach more than 10% (approximately 15% of installed power).
3. It was very informative meeting, very well organised and it fully met expectations especially in the area of safety of nuclear reactors.
4. All the information obtained during the forum will be passed to the investor, nuclear regulator and TSO to facilitate in planning of the future development of nuclear programme and proper selection of applied technology.
5. Poland is interested in issues of waste minimization and spent fuel management.
6. No recommendation.

7.24. Romania

1. National nuclear strategy development/Energy strategy.
   - Romania is currently operating 2 CANDU units (Cernavoda 1&2) and will construct 2 more CANDU units on the same site (Cernavoda units 3&4) in 2020.
   - After 2035-2040 new Gen III+ NPP (i.e. EPR1600, AP1000, ABWR1300, APR1400) is to be constructed following the site selection project which was finished in 2009.
   - For Cernavoda 3&4 Romania needs to finish the procedure to find a new investor.
2. On sustainability, until 2008 Romania has national sustainable development strategy that covers 2013, 2020 and 2030. Authorities & investors periodically debate about how to implement EU regulations, and also mass media debate. After 2012, several companies address the sustainability in their project in balance with the capacity of society to support the increasing cost of renewable technology (e.g. wind, hydro).
3. Yes; technical data, procedural practice and lessons learned and NESA applied by technology holders.
4. Romania is preparing to start NESA about NES alternative scenarios for the long-term. The information would be used to acquire assessment and to start effectively.
5. The main interest will be in waste management, economics and infrastructure.
6. Future Dialogue Forum should focus on the back-end of the fuel cycle and the end spent fuel end state achievement.

7.25. Russian Federation

1. Russia has one of the oldest and biggest nuclear power programs in terms of reactors under operation and construction and also is one of the INPRO founders, donors and leaders. Therefore some questions in the recommended questionnaire require no answer because they are well known.

2. That Russia is now implementing the Program of closed nuclear fuel cycle and planning to operate a mixed system of fast and thermal reactors, therefore the country is deploying a truly innovative NES. Assessing the sustainability of this NES is already a routine part of development in Russia.
3. The Dialogue Forum is useful by definition as a place for meetings of technology users and holders. This Dialogue Forum was really interesting for us as a possibility to present Russian reactor design in terms of INPRO as a vendor country, as well as to learn about the approaches of colleagues from other vendor countries and, what is especially important, to participate in discussions and to hear questions from user countries.
4. First of all it is necessary to support the activities aimed at further modernization of INPRO Methodology and approaches to NESAs.
5. Russia has a complete nuclear fuel cycle – therefore the answer is certainly "Yes".
6. From this Dialogue Forum and many other forums and meetings, one of the main impediments for many countries embarking on NP is the back-end issue. Therefore, one of possible topics for future DFs could be related to back-end solutions for countries with initial or small nuclear power programs, including regional and multilateral approaches.

Additional remark: Small and medium nuclear power was often mentioned during the forum. In recent years Russia was performing expert studies of all our SMR projects using the INPRO Methodology. Russia has received very interesting results and experience and also formulated some proposals for improving the INPRO Methodology to make it better fit to specific SMR issues. Russia intends to present these results at next dialogue forums.

7.26. Thailand

Thailand is considering joining the INPRO Group as soon as acquiring resources and human.

1. The national plan was revised three times during this three year and is not yet finalized. The latest tentative plan is to introduce two 1000 MWe NPPs by 2026 and 2027.
2. There is no particular discussion on sustainability of NES. However, there are discussions on sustainability of the overall energy supply system and also in particular areas introduced in the INPRO methodology.
3. Thailand received a lot of useful information, especially technical information of reactors and the sources of the information. However, Thailand representative was not very sure whether there are enough country experts and competencies to perform INPRO assessment in Thailand. Thailand may need to go through the Tecdoc-1575 and other related documents more closely.
4. Technical information and know-hows gained from of the Dialogue Forum will be shared in the Thailand Institute of Nuclear Technology Institute and also to related institutes. In addition, the institute together with two universities is doing a survey study on the feasibility of SMRs, which includes the assessment of each SMR. The information will be used to perform a limited scope of NESA following the way that all vendors have done in the forum.
5. Thailand is interested in economics and environmental stressors.
6. In order to further familiarize Member States with the INPRO methodology, it is useful to hold more Dialogue Forums on other particular issues. However, though the comparison of the reactors is not the objective of the INPRO methodology, it is the opinion that asking the vendors to evaluate their reactors toward the same set of particular criteria would offer more benefit to Member States.

7.27. Ukraine

1. Nuclear energy system of Ukraine in 2020 – 2030:
   - Lifetime extension of the existing NPP units of Ukraine for 20 years. Lifetime of 11 units (the installed capacity of each unit is 1 GW) is supposed to be extended.
   - Construction and commissioning of new units with total capacity of 5 GW by 2030 (according to the basic scenario and considering the completion of units 3 and 4 at Khmelnitska NPP to 2020).
   - Commencement of construction of new units to replace those to be decommissioned after 2030 (it is necessary to start construction of 12 GW of installed capacity).

   Total installed capacity of Ukrainian NPPs in 2030 may be about 19 GW, with nuclear share in total energy mix of 47-48%.
2. Ukraine discussed sustainability and the near plans:
   - Assessment of current NES using INPRO methodology, to finish in 2014
   - Elaboration of an action plan for sustainable nuclear generation until 2030

3. Ukraine obtained the big set of information especially related to safety of new reactors. At the same time Ukraine has to emphasize that NES is not only reactor safety in NPP design. There are a lot of other types of information expected to be received from vendors for assessment of NES according to INPRO methodology, such as economic issues, radioactive waste management aspects, spent fuel management and storage and front end and back end of fuel cycle.
4. Ukraine has to collect all information which was received from vendors and will provide internal assessment taking into account information and results assessment from others organizations, such as European Utility Requirement group.
5. Ukraine has specific interest in the area of waste minimization and spent fuel management, as two biggest tasks for the near future.
6. The organization by IAEA of the meetings with vendors is very useful and Ukraine recommended INPRO to provide the same meeting (with vendors, to include some specific area such as radioactive waste and spent fuel management) in the future.

7.28. Vietnam

1. In November 2009, Vietnam's National Assembly approved the government plan to build the first two NPPs Ninh Thuận 1&2 with a total capacity of 4,000 MW. Russia and Japan have been chosen as partners for Ninh Thuận 1&2 respectively. In July 2011 the Prime Minister approved master plan on electricity development for period 2012-2020 and to 2030. According to the plan, Vietnam nuclear power capacity will increase from 1,000 MW in 2020 to 10,000 MW in 2030.
2. Development of sustainability of NES is not only required but also objective to ensure economic competitiveness, safety, security, environmental protection and non-proliferation of nuclear power program. Vietnam is interested in lessons learned on infrastructure for nuclear power
3. Many issues presented and discussed in the Dialogue Forum is interesting and very useful.
4. Information is disseminated and used in studies related for making policy for Vietnam's nuclear power development and preparation work for Ninh Thuận 1&2 NPP project
5. As a newcomer embarking on NPP, Vietnam is interested in new generation reactor with innovative design and improvement for increasing economics efficiency, more safety and secure, and waste minimization.
6. In order to enhance meeting outcome, the follow-up Dialogue Forum should have presentations on some other advanced reactor types.

8. RESULTS OF QUESTIONNAIRES

A short questionnaire was distributed to representatives from Member States prior to the Dialogue Forum to gain understanding of the following issues:

1. Familiarity with INPRO Methodology and a Nuclear Energy System Assessment (NESA) using the INPRO Methodology
2. Main expectation in attending the Dialogue Forum knowing vendors' are going to demonstrate the use of INPRO Methodology to assess safety of reactor
3. The extent the topic of sustainability has been discuss in the respondents' country
4. Ranking of the importance of benefits of performing a NESA using the INPRO Methodology, to Member States
5. INPRO areas which may be of interest to MSs for a partial / full NESA

A total of 33 responses were received from 26 Member States, comprising 21 INPRO Members, 4 INPRO Observers and 1 other Member State.

Question 1: Rate of familiarity with

a. INPRO Methodology

Most of the Member States score 3 (average) for their familiarity with the INPRO Methodology, and the average score is 3.2. As expected, INPRO Members in general have better familiarity with the methodology compared to INPRO observers and non-INPRO Member. However, it is noted that the range of scores among INPRO Members was quite wide, from as low as 1 (no familiarity) (Jordan) up to 5 (very familiar) (Armenia, USA). Overall, the results show that INPRO needs to further familiarize INPRO Members with the Methodology for a better outreach.

b. A Nuclear Energy System Assessment (NESA) using INPRO Methodology

The results show that most of the countries are familiar with a NESA as much as with INPRO Methodology, with an average score of 2.9. There were 5 INPRO Members (Germany, India, Kenya, Pakistan, Russian Federation) who provided lower scores for NESA (from scores of 3-4 for familiarity with INPRO Methodology down to 1-3 for a NESA).

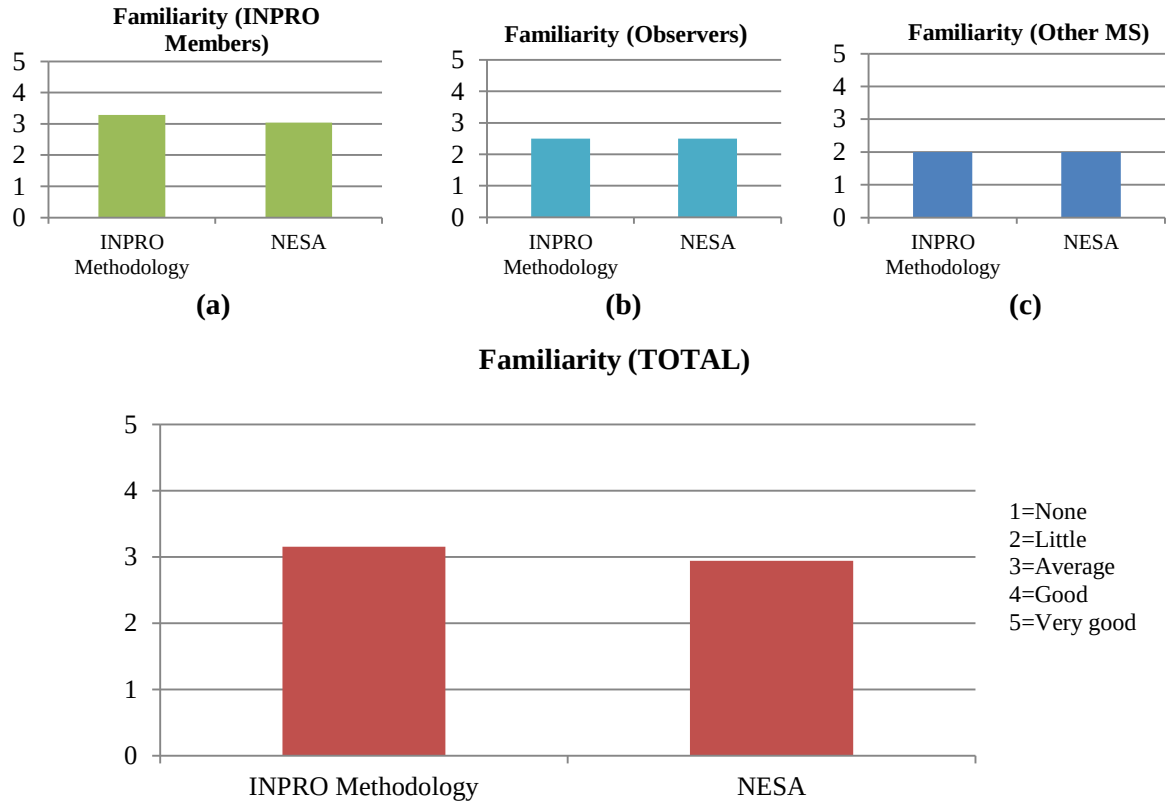


FIGURE 8.1. Familiarity with INPRO Methodology and NESA. *(A total of 33 responses were received from 26 Member States, comprising 21 INPRO Members, 4 INPRO Observers and 1 other Member State)*

Question 2: Main expectation in attending the Dialogue Forum, knowing that several reactor vendors are going to demonstrate the use of INPRO Methodology to assess safety of reactor design.

Respondents provided a range of expectations from this Dialogue Forum, which can be summarized as follows:

- To familiarize with INPRO methodology and NESA
- To observe assessment using INPRO Methodology by vendors
- To checking the update of INPRO Methodology and its revision process
- To find out current reactor designs and updated safety features post Fukushima
- To obtain data from vendors including the extent of data provision
- To obtain feedback on INPRO Methodology: implementation, clarification of terms, simplification
- To see experience of other Member States
- To seek the importance for authority / regulatory requirements.

Questions 3: The extent the topic of sustainability has been discussed in the respondent's country, considering INPRO Methodology is targeting at sustainability of a nuclear energy system, whether a current or future / planned system.

The survey shows that 21 out of 35 respondents, which correspond to 18 Member States, claimed that sustainability issue has been well discussed in their country. The remaining admitted that it was not so well or never discussed. The actual responses are quite interesting, since countries with huge nuclear power program (China) and a NESA country (Ukraine) fall in the latter category.

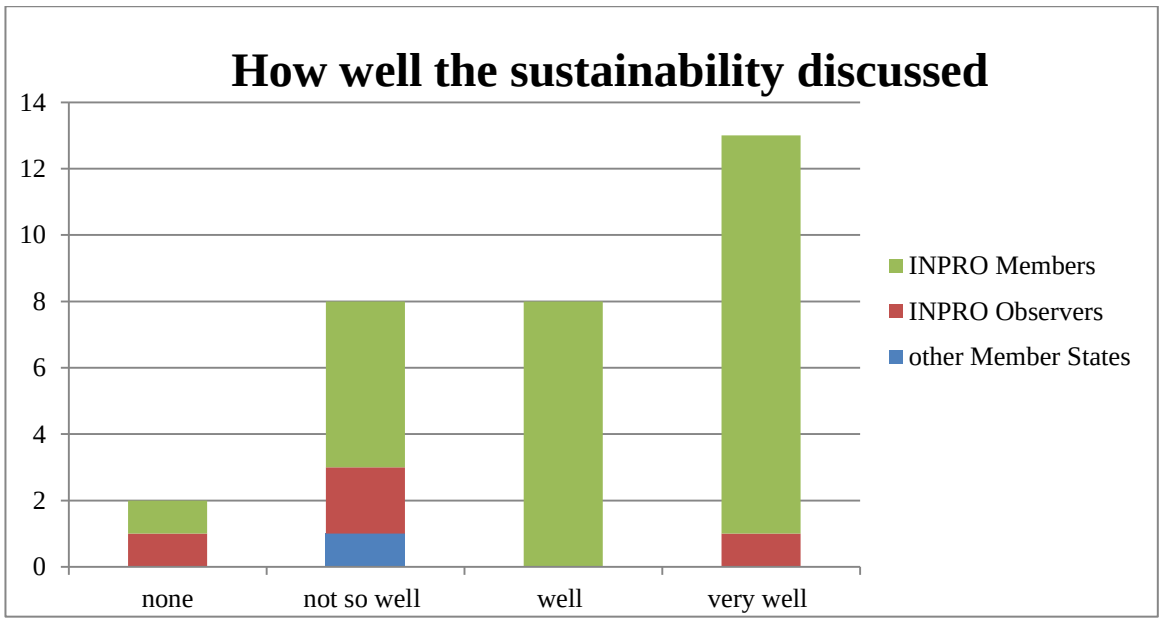
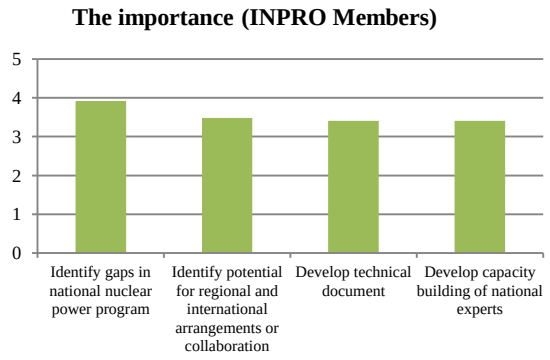


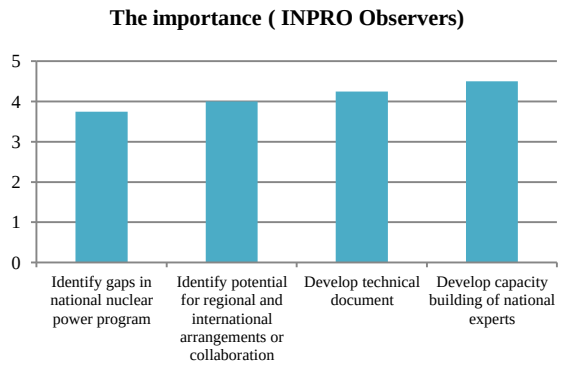
FIGURE 8.2. Discussion of sustainability in Member State. (*A total of 33 responses were received from 26 Member States, comprising 21 INPRO Members, 4 INPRO Observers and 1 other Member State*)

Question 4: Rank the importance of benefits which can be obtained from performing an assessment of a nuclear energy system using the INPRO Methodology, with respect to respondent's country.

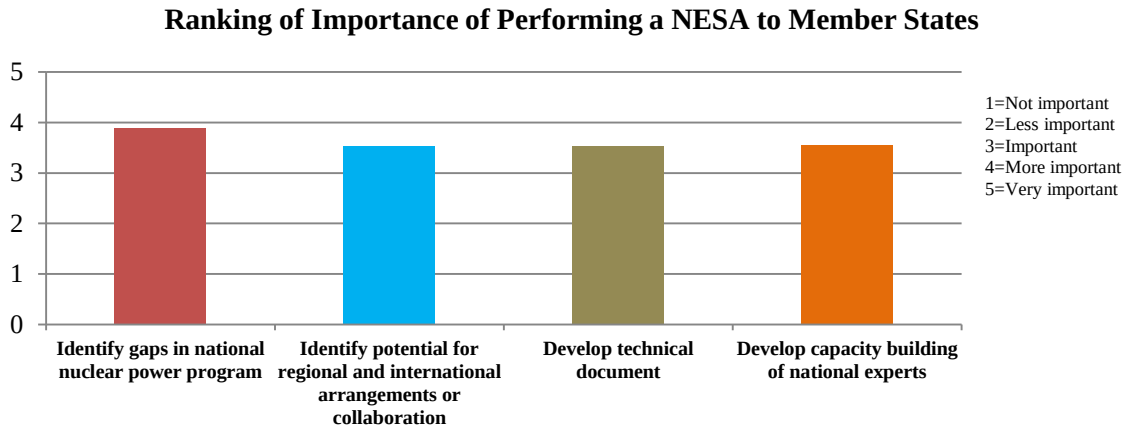
In general, the respondents from Member States provided similar responses to the four benefits given in the survey, with the average score ranging from 3.75 to 4.5. The INPRO Members placed 'Identifying gaps in the national nuclear power program which will require follow-up actions' as slightly more important than the other benefits. Meanwhile, INPRO observers selected 'Developing capacity building of national experts' as relatively more important. The difference in the view of INPRO observer and also one other Member State (non INPRO Member/Observer) suggest that INPRO may need to emphasize on the capacity building of national experts as additional way to promote NESA using the INPRO Methodology to support infrastructure for newcomers who are planning for deployment of their nuclear energy system.



(a)



(b)



(c)

FIGURE 8.3. Ranking of the importance of benefits from performing a NESA using the INPRO Methodology to Member States. *(A total of 33 responses were received from 26 Member States, comprising 21 INPRO Members, 4 INPRO Observers and 1 other Member State)*

Question 5: Areas in the INPRO Methodology that may be of interest to Member States for a partial / full NESA.

Most countries considered Economics and Safety as INPRO areas of interest for a NESA using the INPRO Methodology, each with 31 and 30 tallies respectively. Waste Management received 25 tallies. A mix of technology holders and newcomer countries selected 'Infrastructure' as another area of interest. Infrastructure is in the fourth place with 20 tallies. Technology holder countries like Germany, India, Japan, Korea, Russian Federation, and the US considered Environment as another INPRO area of interest. It is to be noted that a number of respondents selected the whole 7 INPRO areas; they are from Algeria, Belarus, Chile, Japan, Jordan, Kenya, Malaysia, Vietnam and Nigeria.

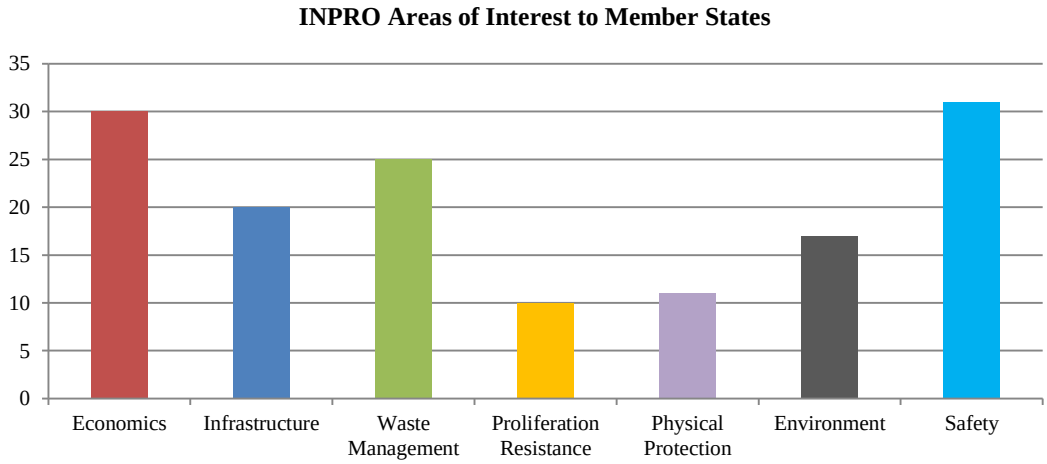


FIGURE 8.4. INPRO Areas of Interest to Member States. *(A total of 33 responses were received from 26 Member States, comprising 21 INPRO Members, 4 INPRO Observers and 1 other Member State)*

9. CONCLUSIONS AND RECOMMENDATIONS

The IAEA INPRO Group convened the 7th Dialogue Forum to develop and promote INPRO assessment services to Member States to help in their long range and strategic nuclear energy planning. The forum brought together representatives from 32 technology holder and technology user countries and 12 reactor designers of evolutionary reactor technologies from AREVA, Candu Energy Inc., GE Hitachi Nuclear Energy International, KEPCO Engineering and Construction and KHNP, Mitsubishi and Rosatom. The designers helped demonstrate exemplary INPRO assessments in the area of safety of their reactor products. The meeting achieved its objectives to provide familiarization of the INPRO Methodology to Member States as a tool for sustainability assessment of NES, to discuss salient features and assessment of evolutionary

nuclear reactors in the area of safety of nuclear reactors, case studies and lessons learned as well as the important role of vendors to support a NESA.

The IAEA presented an overview of the INPRO methodology and how it was developed from the concept of sustainable development within the UN system. The structure of the INPRO methodology covers the key issues that influence sustainability of nuclear power and the infrastructure issue required, which are all translated into the 7 INPRO areas namely economics, infrastructure, waste management, proliferation resistance, physical protection, environment and safety. The methodology is of holistic in nature covering all reactor types and fuel cycle facilities and all phases on a NES from cradle to grave. The methodology comprise 14 Basic Principles (BPs) as goals for development of sustainable NES, 52 User Requirements (URs) indicating what should be done by designer, operator, industry and State to meet goal defined in Basic Principle, and about 150 Criteria (CR) with indicators (IN) and specific acceptance limits (ALs) to check whether a User Requirement is met. INPRO requirements in the area of safety of nuclear reactors were discussed to familiarize the Member States with the requirements for safety, as topic selected by designers for the Dialogue Forum. The forum was informed how INPRO safety manual was developed based on the IAEA Safety Standards specifically targeting at the sustainability, i.e. INPRO assessment is looking at continuous improvements for long sustainability, and not at evaluating compliance with the IAEA Safety Standards or checking licensability. The main outcome of a NESA is a sustainability metric of defined NES through identification of gaps to define a path toward a sustainable NES. Member States were also briefed by the NSNI from Department of Nuclear Safety on the structure of IAEA Safety Standards considering NESA assessors need to access to safety analysis reports that must comply with certain safety requirements.

The expert designers successfully delivered INPRO assessments in the area of safety of their corresponding reactor design, namely APWR, AES-2006, ABWR, APR1400, EC6, ESBW, and EPR. All reactors are evolutionary improvements of currently operating NPPs with some already under construction. Each vendor selected a number of INPRO safety criteria / evaluation parameters and showed the type of data that were used for the assessment, how the assessment was performed, and how their design features have met or should meet the acceptance limit of INPRO safety criteria / evaluation parameters. As many as 31 out of 54 INPRO safety evaluation parameters and criteria were covered combinedly for the seven different reactor products. Robustness, high quality of operation, I&C and inherent characteristics, grace period for normal operation and DBA, frequency of DBA, reliability of engineered safety features, barriers, controlled state, sub criticality, frequency of release into containment and to environment, accident management, and independence of DID levels were among the mostly chose topics. Incorporation of lessons learned from the Fukushima accident into the design was widely elaborated. Additional features such as portable power/generator, on-site/off-site additional power supply system, and recharge battery have been adopted by the designs. Some designers also depicted how their evolutionary designs could withstand the Fukushima-like accident, including the reliance on passive safety system. Insight into criteria on RD&D understanding was provided, i.e. description of sample RD&D performed, scaling and computer codes used for verification and validation. Other issues addressed outside the safety area were, among other, overall reduction of costs, extended reactor lifetime, higher plant efficiency, increased power output, increased maneuverability, and minimization of waste. Flexibility of the design was cited to meet country's site specific requirements.

- APWR by Mitsubishi

  Mitsubishi discussed the 1700 MWe US-APWR design given the availability of the safety reports in the US NRC website. The reactor was designed with major objectives to achieve the best performance in safety, reliability, operability and maintainability, and to support large electric grid utilities aiming at economical production costs. PWR (four-loop) Ohi 4 in Japan is used as the reference plant for the INPRO assessment in the area of safety. Six design aspects were discussed, namely, operational safety margins, reactor vessel manufacturing and welding, simplicity and reliability in safety system design, reliability data of equipment and CDF, detection of anomalous condition and safety analysis, and integrated full digital I&C system and advanced operator console. These correspond to INPRO safety criteria: robustness, operation, failures and disturbances, I&C and inherent characteristics, frequency of DBA, safety features, sub-criticality, frequency of release into containment, processes, frequency of release to environment and human factors. A specially prepared handout containing technical aspects to supplement the assessment was provided to the forum.

- AES-2006 by Rosatom

Rosatom presented INPRO assessment in the area of safety of the 1,197 MWe AES-2006, an evolution of the reference plant AES-91 (VVER1000/V-320). The main principles and approaches for the AES-2006 design are maximum use of well-proven solutions and technologies as well as minimizing costs and timeframes by using of experience obtained during construction and operation of the reference NPP. Assessments of the following INPRO safety criteria were presented: robustness, I&C and inherent characteristics, grace period, inertia, frequency of DBA, grace period (DBA), safety features, barriers, controlled state, sub criticality, frequency of release into containment, processes, accident management and independence of DID levels.

- ABWR by GE Hitachi Nuclear Energy International

  GE Hitachi NE Intl. presented INPRO assessment in the area of safety of its 1,360 MWe ABWR, taking BWR5 in Kashiwazaki Kariwa, Japan, as the reference plant. Assessments of the following INPRO safety criteria were presented: robustness (quality of manufacture and construction), use of worldwide operating experiences, grace period, inertia, frequency of DBA, grace period (DBA), safety features, frequency of release into containment, processes, independence of DID levels, confidence in innovation, and occupational dose.

- APR1400 by KEPCO-E&C and KHNP

  KEPCO-E&C and KHNP delivered sample INPRO assessment in the area of safety for their APR1400 which was developed based on the currently operating OPR1000 design, incorporating advanced design features to enhance safety and operational flexibility, using proven technology enhanced constructability and maintainability. The OPR1000, Shin Kori 2 in Korea is taken as the reference plant. Assessments of the following INPRO safety criteria were presented: I&C and inherent characteristics, safety features, accident management, RD&D, computer codes and scaling.

- EC6 by Candu Energy Inc

  Candu Energy Inc discussed the EC6, taking the generic CANDU6 plant as the reference plant. Particular criteria/evaluation parameters were selected to address lessons learned from Fukushima that have been incorporated into the EC6 design. Assessments of the following INPRO safety criteria were discussed: robustness (margins of design), operation (redundancy of system), safety features, barriers, controlled state, frequency of release into containment, frequency of release to environment and independence of DID levels. Candu Energy Inc had also prepared a report on application of INPRO comparative assessment for Enhanced Candu 6 pertaining to lessons learned from the Fukushima Dai-ichi nuclear event.

- ESBWR by GE Hitachi Nuclear Energy International

  GE Hitachi Nuclear Energy International presented INPRO assessment in the area of safety of the ESBWR, taking BWR6 as the reference plant. Five design aspects were discussed, namely, simplicity of passive safety, redundancy and reliability of safety systems, severe accident prevention and mitigation, risk-informing design with probabilistic safety assessments, and confidence in safety features. These correspond to the following INPRO safety criteria: robustness (simplicity of design, redundancy of system), operation (high quality of operation), grace period, inertia, frequency of DBA, grace period (DBA), safety features, frequency of release into containment, processes, accident management, frequency of release to environment independence of DID levels, confidence in innovation and risk informed approach.

- EPR by AREVA

  AREVA presented the 1600 MWe EPR, which is an evolution mature French N4 reactor (1,475 MWe) and German KONVOI reactor (1,365 MWe) incorporating proven technologies, lessons learned from the past as well as innovations validated through R&D programmes. Information on the following INPRO safety criteria were presented: robustness (margins of design, simplicity of design, quality of manufacture and construction, quality of material, redundancy of system), frequency of release into containment, processes, accident management, and occupational dose. Although lessons learned from Fukushima are not yet all drawn, preliminary evaluations show that the EPR™ would have withstood the

earthquake and the subsequent tsunami as confirmed by the complementary safety evaluations (stress tests). Light improvements may be incorporated to satisfy new requirements.

The technical capabilities of assessors committing to a NESA ultimately depend on availability of data. Member States were informed on how to best proceed with obtaining safety design data, i.e. through open source such as regulatory body website. For commercial data like cost or economics, experimental data and other proprietary data, assessor countries are advised to establish close contact with the vendors or related parties, in which commercial or legal agreement might be required. Overall, it was agreed that involvement of vendors is important to provide technical information and interpretation of the design to deliver a high quality assessment. It was pointed out, nevertheless, that assessor from user countries should be the subject matter experts, and in doing so continuous communication with INPRO should be done to succeed in data selection.

From survey and comments, representatives from Member States noted that participation of vendors to provide hands-on assessment was highly anticipated and valued for better understanding of the use of INPRO methodology. The survey indicates that sustainability issue is well discussed at the national level and identifying gaps in national nuclear power program is deemed to be of most importance. Development of capacity building of national experts and identifying potential for regional and international arrangements or collaboration are also targeted. Economics, safety and waste management received high priority and waste management in particular was proposed to be the topic for the future Dialogue Forum. Several Member States have expressed interest in conducting a NESA using the INPRO methodology.

Altogether, the INPRO Group received important feedbacks and important comments from reactor designers and the forum on the applicability of INPRO Methodology. They include aspects, *inter alia*, on (i) requirements for reliability of the data which necessitate accumulation of historical numeric data that cannot be attained for new designs / evolutionary reactors; (ii) optimization of criteria for robustness, such as simplicity, redundancy, and diversity; (iii) better clarification of independence of DID levels in the methodology; and (iv) lessons learned from Fukushima including external hazards. These are inputs to be incorporated in the on-going revision and update of the INPRO safety manual.

To sum up, designers of evolutionary reactors have provided very valuable contribution toward familiarization of INPRO Methodology to the Member States with their exemplary assessment in the area of safety of nuclear reactor, and also to important feedbacks for the on-going revision of the INPRO manual as well as to the validation of the INPRO methodology. The important role of vendors in providing the necessary design data and interpretation of the design for a NESA project by Member States is understood. Meeting participants are looking forward to the next Dialogue Forum to further discuss enhancement of sustainability of nuclear energy system using the INPRO methodology.

ANNEX I. TERMS OF REFERENCE

INPRO
International Project on Innovative Nuclear Reactors and Fuel Cycles

IAEA

Terms of Reference for Technical Meeting
INPRO Dialogue Forum on the Sustainability of Nuclear Energy System Based on Evolutionary Reactors

IAEA Headquarters, Vienna, Austria, 18-22 November 2013
Boardroom A, Building M, Vienna International Centre (VIC)

Background

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was launched in the year 2000, based on resolutions of the IAEA General Conference (GC(44)/RES/21). INPRO's objective is to help ensure that nuclear energy is available in the 21st century in a sustainable manner. Moreover, INPRO seeks to bring together all interested Member States, including technology holders and users, to jointly consider actions to achieve desired innovations.

Four projects are reflected in the Action Plan for 2012-2013, as endorsed by the Steering Committee: Project 1: National long-range nuclear energy strategies; Project 2: Global nuclear energy scenarios; Project 3: Innovations (technical and institutional); and Project 4: Policy coordination and the INPRO Dialogue Forum. The current Technical Meeting will be the 7th INPRO Dialogue Forum on the Sustainability of Nuclear Energy System Based on Evolutionary Reactors.

Additional details on the INPRO project are available on the INPRO website: www.iaea.org/INPRO.

Activity: 7th INPRO Dialogue Forum

To assist Member States in long range and strategic planning of nuclear energy programmes, INPRO is offering support to Member States in assessing sustainability of nuclear energy systems (NES) as part of their country's energy mix.

In selecting certain technology (e.g. a reactor design) for the NES sustainability assessment (NESA), the assessor is dependent on the information available for each particular design of interest. Some INPRO Methodology criteria require that specific input data be collected and analysed by an assessor. In some cases, engagement with technology design authorities and their direct participation in an assessment may be the most practical approach. This INPRO Dialogue Forum will provide an opportunity for nuclear technology designers/vendors of evolutionary power reactors to meet with interested IAEA Member States' participants and for the vendors to present exemplary INPRO Methodology assessments of their reactor products. It is also an opportunity for the technology holders and users to discuss approaches to further their cooperation. To prepare for the Dialogue Forum, Consultancy Meetings with participating vendors are planned for the 3rd quarter of 2013.

Meeting objectives

The objectives of the 7th Dialogue Forum are to:

1. Familiarize Member States with the INPRO Methodology, focused on the sustainability assessment of NES, to support long-range and strategic planning of a successful nuclear power program.
2. Discuss assessment of evolutionary nuclear reactors in the areas of *safety of nuclear reactors, environment (stressors)* and *waste management (minimization).*
3. Discuss case studies and lessons learned as well as support to the on-going and future projects on national NESAs.
4. Discuss salient features of evolutionary nuclear reactors relevant to sustainability of NES, and the important role of vendors in a NESA – particularly in providing necessary data and familiarization with the design to support the assessment.

Expected participation

The meeting is open to 50 or more participants from Member States including Albania, Algeria, Argentina, Armenia, Australia, Bangladesh, Belarus, Belgium, Brazil, Bulgaria, Canada, Chile, China, Croatia, Czech Republic, Egypt, Finland, France, Germany, Georgia, Hungary, India, Indonesia, Israel, Italy, Japan, Jordan,

Kazakhstan, Kenya, Republic of Korea, Latvia, Lithuania, Malaysia, Mexico, Mongolia, Morocco, Netherlands, Nigeria, Norway, Oman, Pakistan, Poland, Romania, Russian Federation, Saudi Arabia, Slovakia, Slovenia, South Africa, Spain, Sudan, Sweden, Switzerland, Thailand, Tunisia, Turkey, Ukraine, United Arab Emirates, United Kingdom, United States of America, Uruguay, Vietnam, the European Commission, FORATOM, Generation IV International Forum (GIF), International Framework for Nuclear Energy Cooperation (IFNEC), International Science and Technology Center (ISTC), Sustainable Nuclear Energy Technology Platform (SNETP), the OECD Nuclear Energy Agency (OECD/NEA), the World Nuclear Association (WNA).

Participants include representatives from national authorities, technology developers (vendors/reactor designers), utilities, technical support organisations (TSO's), nuclear energy project implementing organizations (NEPIO's), and members of the INPRO Steering Committee.

Location/date

The meeting will be held at the IAEA Headquarters, Boardroom A, Building M, Vienna International Centre (VIC), Vienna, Austria, from 18-22 November, 2013.

Programmatic context

The activity is being implemented under Project 4, Activity 1.1.4.4/3 and Activity 1.1.4.4/7 of the IAEA Programme & Budget 2012–2013.

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ANNEX II. AGENDA

INPRO Dialogue Forum on Sustainability of Nuclear Energy System Based on Evolutionary Reactors

Vienna International Centre (VIC), IAEA Headquarters, Vienna, AUSTRIA

19th – 22nd November, 2013. IAEA Board Room A, Building M

Time	Date / Topic	Participant(s)
Tuesday, 19th November		
08:00 – 09:00	*Access to the VIC. Seating of delegates*	*All*
09:00 – 09:25	Opening remarks by DDG-NE	Mr Alexander Bychkov
09:25 – 09:55	Overview of INPRO Project	Zoran Drace (INPRO)
09:55 –10:40	INPRO Methodology for Sustainability Assessment of Nuclear Energy System (NES)	Jon R. Phillips (INPRO)
Group photo and Break		
11:00 – 11:25	Safety Assessment and NPP Design Requirements	Hussam Khartabil (NSNI)
11:25 – 11:50	INPRO Methodology Requirements on Safety of Nuclear Reactors	Andriy Korinny (INPRO)
11:50 – 12:15	Overview of Mitsubishi Advanced PWR	Sumio Fujii (Mitsubishi)
12:15 – 12:40	Assessment items selected for 7th Dialogue Forum	Sumio Fujii (Mitsubishi)
Lunch break		
14:00 – 14:20	Operational Safety Margins in Design of APWR	Sumio Fujii (Mitsubishi)
14:20 – 14:35	Reactor Vessel Manufacturing and Welding of APWR	Sumio Fujii (Mitsubishi)
14:35 – 14:50	Simplicity and Reliability in Safety System Design of APWR	Tesushi Yuasa (Mitsubishi)
14:50 – 15:05	Reliability Data of Equipment and CDF of APWR	Tesushi Yuasa (Mitsubishi)
15:05 – 15:15	Detection of Anomalous Condition and Safety Analysis of APWR	Tesushi Yuasa (Mitsubishi)
15:15 – 15:40	Integrated Full Digital I&C System and Advanced Operator Console of APWR	Sumio Fujii (Mitsubishi)
Break		
16:00 – 16:25	AES-2006 –New Design with VVER Reactor and INPRO Methodology	Denis Kolchinsky (Rosatom)
16:25 – 16:50	Margins and Robustness: Design Solutions Providing the V-491 Reactor Plant Reliability	Mikhail Bykov (Rosatom)
16:50 – 17:15	Safety System of Reactor Plant V-491 and Safety Analyses Results	Mikhail Bykov (Rosatom)
17:15 – 17:40	Discussion	*All*
17:40	Adjourn Welcome reception	 *All*
Wednesday 20th November		
09:00 – 09:25	Principles Providing Safety of the V-491 Type Reactor	Mikhail Bykov (Rosatom)
09:25 – 09:50	Release Into Containment in AES-2006 Design	Denis Kolchinsky

		(Rosatom)
09:50 – 10:15	Independence of DID Levels in AES-2006 Design	Denis Kolchinsky (Rosatom)
10:15 – 10:40	Overview of ABWR Safety Features	James A Beard (GE Hitachi Nuclear Energy International)
Break		
11:00 – 11:25	Reliability and Autonomy of ABWR Safety Functions	James A Beard (GE Hitachi Nuclear Energy International)
11:25 – 11:50	Improving the Operational Response of the ABWR	James A Beard (GE Hitachi Nuclear Energy International)
11:50 – 12:15	Frequency Reduction and Accident Management in the ABWR	James A Beard (GE Hitachi Nuclear Energy International)
12:15 – 12:40	Reducing the Operational Doses in the ABWR	Kenichi Yasuda (GE Hitachi Nuclear Energy International)
Lunch break		
14:00 – 14:25	Manufacturing and Construction Improvements	Kenichi Yasuda (GE Hitachi Nuclear Energy International)
14:25 – 14:50	Overview of APR1400 Design	Jong Tae Seo (KEPCO-E&C)
14:50 – 15:15	I&C and Inherent Characteristics	Jong Tae Seo (KEPCO-E&C)
15:15 – 15:40	Reliability of Engineered Safety Features	Jong Tae Seo (KEPCO-E&C)
Break		
16:00 – 16:25	In-Plant Severe Accident Management	Jong Tae Seo (KEPCO-E&C)
16:25 – 16:50	Design Characteristics of APR1400 Safety Injection System: RD&D and Scaling	Han Gon Kim (KHNP)
16:50 – 17:15	Code Validation for APR1400 LBLOCA Analysis	Han Gon Kim (KHNP)
17:15 – 17:40	Discussion	*All*
17:40	Adjourn	
Thursday 21st November		
09:00 – 09:25	Application of INPRO Comparative Assessment for Enhanced CANDU 6 ® Pertaining to Lessons Learned from the Fukushima Dai-ichi Nuclear Event	Sabine Boyle (Candu Energy Inc.)
09:25 – 09:50	Evolution of EC6 design from CANDU 6	Albert G Lee (Candu Energy Inc.)
09:50 – 10:15	INPRO assessment in the area of safety of EC6 Safety Margins	Albert G Lee (Candu Energy Inc.)
10:15 – 10:40	INPRO assessment in the area of safety of EC6 Robustness	Sabine Boyle (Candu Energy Inc.)
Break		
11:00 – 11:25	INPRO assessment in the area of safety of EC6 Safety Features	Albert G Lee (Candu Energy Inc.)
11:25 – 11:50	INPRO assessment in the area of safety & EC6 Safety Goals	Albert G Lee (Candu Energy Inc.)
11:50 – 12:15	Overview of ESBWR Safety Features	James A Beard (GE Hitachi Nuclear

		Energy International)
12:15 – 12:40	Simplicity of ESBWR Passive Safety	James A Beard (GE Hitachi Nuclear Energy International)
Lunch break		
14:00 – 14:25	Redundancy and Reliability of Passive Safety Features	James A Beard (GE Hitachi Nuclear Energy International)
14:25 – 14:50	Risk-Informing ESBWR Design with PSA	Gary L Miller (GE Hitachi Nuclear Energy International)
14:50 – 15:15	ESBWR Severe Accident Prevention and Mitigation	Gary L Miller (GE Hitachi Nuclear Energy International)
15:15 – 15:40	Confidence in ESBWR Safety Features	Gary L Miller (GE Hitachi Nuclear Energy International)
Break		
16:00 – 16:25	Presentation of the EPR™ reactor	Franck Lignini (AREVA)
16:25 – 16:50	Robustness of Design Part 1: Margins and Simplicity of Design	
16:50 – 17:15	Robustness of Design Part 2: Quality of Manufacture and Construction, Quality of Materials, and Redundancy of Systems	
17:15 – 17:40	Discussion	*All*
17:40	Adjourn	
Friday 22nd November		
09:00 – 09:25	Robustness of Design Part 3: Robustness against External Hazards	Franck Lignini (AREVA)
09:25 – 09:50	Release into the Containment: Major Release, Processes, and Accident Management	
09:50 – 10:15	Occupational Dose	
10:15 – 10:40	Q&A, comments and discussion	*All*
Break		
11:00 – 11:50	Q&A, comments and discussion (continued)	*All*
11:50 – 12:40	Statements from Member States	*Member States*
Lunch break		
14:00 – 15:40	Statements from Member States (continued)	*Member States*
Break		
16:00 – 16:50	Meeting Summary by Chairman Closing Remarks	Juergen Kupitz
16:50	Adjourn	

ANNEX III. TABULATED RESULTS OF QUESTIONNAIRE

TABLE IIII.1. Rate of familiarity (1=None; 2=Little; 3=Average; 4=Good; 5=Very good)

#	Country	INPRO Methodology	NESA
INPRO Members			
1	Algeria	3	3
2	Argentina	4	4
3	Armenia	5	5
4	Belarus	4	4
5	Brazil	3	3
6	Bulgaria	4	4
7	Chile	2	2
8	China	3	3
9	Egypt	2	2
10	Germany	3	1
11	India	4	3
12	Indonesia	4	4
13	Indonesia(2)	4	4
14	Indonesia(3)	4	4
15	Japan	4	4
16	Jordan	1	1
17	Kazakhstan	2	2
18	Kenya	4	2
19	Korea, Rep of	3	3
20	Malaysia	2	2
21	Pakistan	4	3
22	Poland	2	2
23	Romania	4	4
24	Russian Federation	3	2
25	South Africa	3	3
26	Ukraine	2	2
27	United States of America	5	5
28	Vietnam	4	4
INPRO Observers			
29	Hungary	2	2
30	Nigeria	4	4
31	Nigeria(2)	2	2
32	Thailand	2	2
33	Tunisia		
	Other Member States		
34	Albania	2	2
35	Bangladesh		
TALLY	**INPRO Member (%)**	3.286	3.036
	INPRO Observers (%)	2.5	2.5
	Other Member States (%)	2	2
	TOTAL (%)	3.152	2.939

TABLE III.2. The extent the topic of sustainability has been discussed in the respondent's country, considering INPRO Methodology is targeting at sustainability of a nuclear energy system, whether a current or future / planned system.

#	Country	None	Not so well	Well	Very well
INPRO Members					
1	Algeria			√	
2	Argentina			√	
3	Armenia			√	
4	Belarus				√
5	Brazil			√	
6	Bulgaria				√
7	Chile		√		
8	China		√		
9	Egypt				
10	Germany				
11	India				√
12	Indonesia				√
13	Indonesia(2)				√
14	Indonesia(3)				√
15	Japan				√
16	Jordan	√			
17	Kazakhstan		√		
18	Kenya		√		
19	Korea, Rep of				√
20	Malaysia			√	
21	Pakistan			√	
22	Poland				√
23	Romania				√
24	Russian Federation				√
25	South Africa				√
26	Ukraine		√		
27	United States of America			√	
28	Vietnam			√	
INPRO Observers					
29	Hungary		√		
30	Nigeria		√		
31	Nigeria(2)				√
32	Thailand	√			
33	Tunisia				
	Other Member States				
34	Albania		√		
35	Bangladesh				
TALLY	**INPRO Member**	1	6	8	12
	INPRO Observers	1	2	0	1
	Other Member States	0	1	0	0
	TOTAL	2	9	8	13

TABLE IIII.3. Rank the importance of benefits which can be obtained from performing an assessment of a nuclear energy system using the INPRO Methodology, with respect to respondent's country. (1=Not important; 2=Less important; 3=Important; 4=More important; 5=Very important)

#	Country	Identify gaps in national nuclear power program	Identify potential for regional and international arrangement or collaboration	Develop technical document	Develop capacity building of national experts
INPRO Members					
1	Algeria	5	4	4	5
2	Argentina	3	3	3	3
3	Armenia	4	2	5	5
4	Belarus	5	5	4	3
5	Brazil	2	2	2	2
6	Bulgaria	5	4	3	2
7	Chile	5	4	3	3
8	China	3	2	3	2
9	Egypt	5	2	3	4
10	Germany				
11	India	5	3	1	3
12	Indonesia	4	3	3	5
13	Indonesia(2)	5	4	2	2
14	Indonesia(3)	5	4	3	2
15	Japan	3	5	2	2
16	Jordan	4	3	4	5
17	Kazakhstan	3	3	4	3
18	Kenya		5	5	5
19	Korea, Rep of	4	3	2	1
20	Malaysia	5	5	5	5
21	Pakistan	2	2	3	2
22	Poland	3	5	3	4
23	Romania	5	4	5	4
24	Russian Federation	5	4	5	3
25	South Africa	4	4	5	5
26	Ukraine	3	2	3	4
27	United States of America	2	4	3	4
28	Vietnam	3	3	4	4
INPRO Observers					
29	Hungary	3	5	5	3
30	Nigeria	5	3	3	5
31	Nigeria(2)	5	5	5	5
32	Thailand	2	3	4	5
33	Tunisia				
	Other Member States				
34	Albania		3	4	4
35	Bangladesh				
TALLY	**INPRO Member (%)**	3.923	3.481	3.407	3.404
	INPRO Observers (%)	3.75	4	4.25	4.5
	Other Member States (%)	0	3	4	4
	TOTAL (%)	3.9	3.531	3.531	3.562

TABLE III.4. Areas in the INPRO Methodology that may be of interest to Member States for a partial / full NESA.

#	Country	Economics	Infra-structure	Waste Management	Proliferation Resistance	Physical Protection	Environ -ment	Safety
INPRO Members								
1	Algeria	√	√	√	√	√	√	√
2	Argentina			√			√	√
3	Armenia	√	√	√				√
4	Belarus	√	√	√	√	√	√	√
5	Brazil	√		√				√
6	Bulgaria	√		√				√
7	Chile	√	√	√	√	√	√	√
8	China	√		√	√			√
9	Egypt	√	√			√		√
10	Germany						√	√
11	India	√		√			√	√
12	Indonesia	√	√					√
13	Indonesia(2)	√					√	√
14	Indonesia(3)	√					√	√
15	Japan	√		√	√	√	√	√
16	Jordan	√	√	√	√	√	√	√
17	Kazakhstan	√	√	√				
18	Kenya	√	√	√	√	√	√	√
19	Korea, Rep of	√		√			√	√
20	Malaysia	√	√	√	√	√	√	√
21	Pakistan	√		√				√
22	Poland	√	√	√				√
23	Romania	√	√	√				√
24	Russian Federation	√	√	√			√	√
25	South Africa	√						√
26	Ukraine	√	√	√				√
27	United States of America	√	√	√				√
28	Vietnam	√	√	√	√	√	√	√
INPRO Observers								
29	Hungary			√				
30	Nigeria		√	√		√		√
31	Nigeria(2)	√	√	√	√	√	√	√
32	Thailand	√	√					√
33	Tunisia							
	Other Member States							
34	Albania	√	√				√	√
35	Bangladesh							
TALLY	**INPRO Member**	26	16	22	9	9	15	27
	INPRO Observers	2	3	3	1	2	1	3
	Other Member States	1	1	0	0	0	1	1
	TOTAL	29	20	25	10	11	17	31

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ANNEX V. OPENING REMARKS

Alexander Bychkov

IAEA Deputy Director for Nuclear Energy and INPRO Project Manager

Dear Participants, Ladies and Gentlemen,

On behalf of the International Atomic Energy Agency, I welcome you to the seventh INPRO Dialogue Forum on the Sustainability of Nuclear Energy System Based on Evolutionary Reactors.

The meeting is designed to foster a dialogue between technology developers and technology users from all IAEA interested Member States. It serves as a platform to discuss issues related to sustainable development and deployment of nuclear energy. The forum facilitates a mutual understanding of the needs and potential roles of technology users, and the possibilities and limitations of technology developers.

This particular forum is intended to familiarize participants with the INPRO Methodology, focused on the sustainability assessment of nuclear energy systems. Previous experience shows that in selecting certain reactor technology for the NES sustainability assessment (NESA), the assessor is dependent on the information available for each particular design of interest. This INPRO Dialogue Forum will provide an opportunity for Member States to meet and discuss with nuclear technology developers on potential cooperation.

Ladies and Gentlemen,

INPRO Methodology is an internationally validated tool for the assessment of nuclear energy system. It serves to assist Member States in assessing the NES sustainability based on a comprehensive set of internationally agreed basic principles, requirements and criteria in the areas of Economics, Infrastructure, Waste management, Proliferation Resistance, Physical Protection, Environment, and Safety.

In this forum, in addition to the IAEA and INPRO Group, we have representatives from the reactor vendors who will demonstrate the use of INPRO Methodology to assess their evolutionary reactor designs. The vendors include Candu Energy Incorporated, Mitsubishi Heavy Industry, KEPCO and KHNP, GE Hitachi Nuclear Energy International, AREVA and Rosatom. The vendors have selected the area of safety of nuclear reactors, one of the seven INPRO areas essential to the sustainability of a nuclear energy system and the most important area as we all have learned from the past Fukushima accident. They will perform evaluation of progress that has been achieved in their evolutionary reactor designs compared against operating reactors.

The assessments performed by the vendors also serves to further validate the INPRO Methodology as a tool that can be used by the Member States in their long range and strategic planning of nuclear energy programmes. The expected output shall be very useful for current and future NESA assessors and for the on-going efforts in updating the INPRO methodology.

Hereby I would like to acknowledge and thank the nuclear industry for their participation.

Ladies and Gentlemen,

In conclusion, INPRO is part of the IAEA integrated services provided to Member States considering initial development or expansion of their nuclear energy programmes. And INPRO cooperates with Member States to ensure that sustainable nuclear energy is available to help meet the energy needs of the 21st century. The INPRO Methodology serves as the tool for this purpose.

I wish to thank all participants, presenters and colleagues for their significant interest and contribution to this important event.

I also wish to thank the INPRO Group for organizing this dialogue forum, for bringing in the reactor vendors to the event.

I wish you all an active, successful and fruitful meeting.

Thank you.

ABBREVIATIONS

AB	:	Auxiliary Building
AC	:	Alternating Current
ACC	:	Advanced Accumulator
ACIWA	:	AC Independent Water Addition
ADS	:	Automatic Depressurization System
AHWR	:	Advanced Heavy Water Reactor
AOO	:	Anticipated Occupational Occurrence
APC	:	Airplane Crash
APC-S	:	Auxiliary Process Cabinet – Safety
APR1400	:	Advanced Power Reactor 1400
APWR	:	Advanced Pressurized Water Reactor
ASD	:	Adjustable Speed Drive
ATLAS	:	Advanced Thermal-hydraulic Test Loop
ATWS	:	Anticipated Transient Without Reactor Scram
BAMP	:	Boric Acid Makeup Pump
BiMAC	:	Basemat internal Melt Arrest and Coolability
BDBA	:	Beyond Design Basis Accident
BOC	:	Beginning of Cycle
BOP	:	Balance of Plant
BWR	:	Boiling Water Reactor
CANDU	:	CANada Deuterium Uranium
CAR	:	Control Absorber Rod
CB	:	Compound Building
CCF	:	Common Cause Failure
CCI	:	Core Concrete Interaction
CCU	:	Containment Cooling Unit
CCWP	:	Component Cooling Water Pump
CCWS	:	Component Cooling Water System
CDF	:	Core Damage Frequency
CEDM	:	Control Element Drive Mechanism
CFS	:	Cavity Flooding System
CHRS	:	Containment Heat Removal System
CHTR	:	Compact High Temperature Reactor
CLDS	:	Coolant Leak Detection System
CLI	:	Cold Leg Injection
CMF	:	Common Mode Failure
COPS	:	Containment Overpressure Protection System
CPCS	:	Core Protection Calculator System
CPM	:	Control Panel Multiplexer
C-PHRS	:	Containment Passive Heat Removal System
CRD	:	Control Rod Drive
CS	:	Containment Spray
CSA	:	Canadian Standards Association
CSAS	:	Containment Spray Actuation Signal
CTG	:	Combustion Turbine Generator
CVR	:	Coolant Void Reactivity
DAS	:	Diverse Actuation System
DBA	:	Design Basis Accident
DBC	:	Design Basis Conditions
DBE	:	Design Basis Earthquake
DCD	:	Design Control Document
DCH	:	Direct Containment Heating
DEB	:	Direct ECC Bypass
DEC	:	Design Extension Conditions

DF	:	Defence In Depth
DID	:	Decontamination Factor
DIS	:	Diverse Indication System
DIVA	:	Downcomer Injection Visualization and Analysis
DNB	:	Departure from Nucleate Boiling
DNBR	:	Departure from Nucleate Boiling Ratio
DOBO	:	Downcomer Boiling
DPS	:	Diverse Protection System
DVI	:	Direct Vessel Injection
EBIS	:	Emergency Boron Injection System
EBS	:	Emergency Borated System
EC6	:	Enhanced CANDU 6
ECC	:	Emergency Core Cooling
ECCS	:	Emergency Core Cooling System
ECSBS	:	Emergency Containment Spray Back-up System
EDG	:	Emergency Diesel Generators
ENSREG	:	European Nuclear Safety Regulators Group and Commission
EFCVS	:	Emergency Filtered Containment Venting System
EFWP	:	Emergency Feedwater pump
EFWS	:	Emergency Feed Water System
EHRS	:	Emergency Heat Removal System
EHM	:	Equipment Health Monitoring
EOC	:	End of Cycle
EOF	:	Emergency Operations Facilities
EPRI	:	Electric Power Research Institute
EPS	:	Emergency Power Supply
EPP	:	Entry Point Project
ERVC / ERVCS	:	External Reactor Vessel Cooling / ERVC System
ESBWR	:	Economic Simplified Boiling Water Reactor
ESF	:	Engineered Safety Features
ESFAS	:	ESF Actuation System
ESF-CCS	:	ESF-Component Control System
ESW	:	Electro Slag Welding
ESWS	:	Essential Service Water System
ETP	:	Economic Transformation Programme
FA	:	Fuel Assembly
FBR	:	Fast Breeder Reactor
FDD	:	Flow Distribution Device
FLB	:	Feed Line Break
FMCRD	:	Fine Motion Control Rod Drive
FOAK	:	First-of-a-Kind
GDCS	:	Gravity Driven Cooling System
GIST	:	GDCS Integrated System Testing
GSS	:	Guaranteed Shutdown State
GTG	:	Gas Turbine Generator
HCLPF	:	High Confidence of Low Probability of Failure
HFE	:	Human Factor Engineering
HHSI	:	High Head Safety Injection
HMI	:	Human-Machine Interface
HPCF	:	High Pressure Core Flooder
HPECC	:	High Pressure Emergency Core Cooling
HPME	:	High Pressure Melt Injection
HST	:	Hot wire Switching Tungsten inert gas
HVAC	:	Heating Ventilation and Air Conditioning
IBA	:	Inner Barrel Assembly
I&C	:	Instrumentation and Control

IAEA	:	International Atomic Energy Agency
ICPCCS	:	Isolation Condenser Passive Cooling Containment System
ICIS	:	In Core Instrumentation System
ICS	:	Isolation Condenser System
IE	:	Initiating Event
IHA	:	Integrated Head Assembly
IHTR	:	Innovative High Temperature Reactor
ILW	:	Intermediate Level Waste
INIG	:	Integrated Nuclear Infrastructure Group
INIR	:	Integrated Nuclear Infrastructure Review
INPRO	:	International Project on Innovative Nuclear Reactors and Fuel Cycles
IRWST	:	In-containment Refueling Water Storage Tank
ISI	:	In-Service Inspection
ISLOCA	:	Inter-system Loss of Coolant Accident
IVR	:	In-Vessel Retention
KEPCO-E&C	:	Korea Electric Power Corporation Engineering & Construction
KHNP	:	Korea Hydro & Nuclear Power
KINS	:	Korean Institute for Nuclear Safety
LAC	:	Local Air Cooler
LBB	:	Leak Before Break
LBLOCA	:	Large Break Loss of Coolant Accident
LCO	:	Limiting Conditions of Operation
LCS	:	Local Control Station
LHSI	:	Low Head Safety Injection
LLW	:	Low Level Waste
LOCA	:	Loss of Coolant Accident
LOHS	:	Loss of Heat Sink
LOOP	:	Loss of Off-site Power
LPD	:	Local Power Density
LPDS	:	Loose Parts Detection System
LPSI / LPSIP	:	Low Pressure Safety Injection / LPSI Pump
LRF	:	Large Release Frequency
LUHS	:	Loss of Ultimate Heat Sink
MCCI	:	Monitoring, Control and Diagnostic System
MCDS	:	Molten Core Concrete Interaction
MCR	:	Main Control Room
MIDAS	:	Multi-dimensional Investigation in Downcomer Annulus Simulation
MHI	:	Mitsubishi Heavy Industries
MIG	:	Metal Inert Gas
MMIS	:	Man-Machine Interface System
MSIV	:	Main Steam Isolation Valve
NES	:	Nuclear Energy System
NESA	:	Nuclear Energy System Assessment
NIMS	:	NSSS Integrity Monitoring System
NPA	:	Nuclear Plant Analyzer
NPCS	:	NSSS Process Control System
NPIDP	:	Nuclear Power Infrastructure Development Plant
NPIRDP	:	Nuclear Power Infrastructure and Regulatory Development Plant
NPP	:	Nuclear Power Plant
NPSH	:	Net Positive Suction Head
NSSS	:	Nuclear Steam Supply System
OBE	:	Operating Basis Earthquake
OPR1000	:	Optimized Power Reactor

PAM	:	Post Accident Monitoring
PAR	:	Passive Autocatalytic Recombiner
PAS	:	Process Automation System
PCCS	:	Passive Containment Cooling System
P-CCS	:	Process-Component Control System
PCCV	:	Pre-stressed Concrete Containment Vessel
PCS	:	Power Control System
PCT	:	Peak Cladding Temperature
PCV	:	Primary Containment Vessel
PICS	:	Process Information and Control System
PI/PA	:	Public Information and Acceptance
PIRT	:	Phenomena Identification and Ranking Table
PGA	:	Peak Ground Acceleration
PGAP	:	Performance Assessment of Passive Gaseous Provisions
PHWR	:	Pressurized Heavy Water Reactor
PHRS / PHRS-SG	:	Passive Heat Removal System / PHRS via Steam Generator
POSRV	:	Pilot Operated Safety Relief Valve
PPS	:	Plant Protection System
PRA	:	Probabilistic Risk Assessment
PSA	:	Probabilistic Safety Analysis
PTU	:	Pressurized Water Reactor
PWR	:	Protection Tube Unit
Q-DCIS	:	Safety-related Distributed Control and Information System
QIAS	:	Qualified Indication and Alarm System
RCB	:	Reactor Containment Building
RCCA	:	Rod Control Cluster Assemblies
RCIC	:	Reactor Core Isolation Cooling
RCL	:	Reactor Coolant Line
RCP	:	Reactor Coolant Pump
RCPB	:	Reactor Coolant Pressure Boundary
RCS	:	Reactor Coolant System
RCIC	:	Reactor Core Isolation Cooling
RCSL	:	Reactor Control Surveillance and Limitation system
RCW	:	Recirculated Water
RD&D	:	Research, Development and Demonstration
RFID	:	Radio Frequency In Identification
RHR	:	Residual Heat Removal
RIP	:	Reactors Internal Pump
RLACS	:	Remaining Life Automated Control System
Rosatom	:	Rosatom State Atomic Energy Corporation
RP	:	Reactor Plant
RPS	:	Reactor Protection System
RPV	:	Reactor Pressure Vessel
RRS	:	Reactor Regulating System
RSC	:	Remote Shutdown Console
RSR	:	Remote Shutdown Room
RSW	:	Raw Service Water
RTD	:	Resistance Temperature Detector
RTSS	:	Reactor Trip Switch gear System
RWCU	:	Reactor Water Cleanup Unit
RWST	:	Refueling Water Storage Tank
RWSP	:	Refueling Water Storage Pit
RWT	:	Reserve Water Tank
SA	:	Severe Accident
SAR	:	Safety Analysis Repot
SAM	:	Severe Accident Management
SAMG	:	Severe Accident Management Guideline

SAMS	:	Severe Accident Management Simulation Module
SARHRS	:	Severe Accident Recovery and Heat Removal System
SBLOCA	:	Small Break Loss of Coolant Accident
SBO	:	Station Blackout
SCC	:	Stress Corrosion Cracking
SCD	:	Severe Core Damage
SCP	:	Shutdown Cooling Pump
SDC	:	Shut Down Cooling
SDS1/SDS2	:	Shutdown System 1 / Shutdown System 2
SFB	:	Spent Fuel Bay
SG / SG PHRS	:	Steam Generator / Steam Generator PHRS
SIA	:	System of Integrated Analysis
SIAS	:	Safety Injection Actuation Signal
SICS	:	Safety Information and Control System
SID	:	System Integrated Diagnostics
SIP	:	Safety Injection Pump
SIS	:	Safety Injection System
SIT	:	Safety Injection Tank
SLB	:	Steam Line Break
SLC	:	Standby Liquid Control
SMR	:	Small and Medium-sized Reactor
SOR	:	Shut-Off Rods
SPACE	:	Safety and Performance Analysis CodE
SQed	:	Seismically Qualified
SSC	:	System, Structure and Component
SSE	:	Small Release Frequency
SRF	:	Safe Shutdown Earthquake
SRV	:	Safety Relief Valves
TB	:	Turbine Building
TIG	:	Tungsten Inert Gas
TLOFW	:	Total Loss Of Feed Water
TSC	:	Technical Support Centre
UGS	:	Upper Guide Structure
UHS	:	Uniform Hazard Spectra
URD	:	Utility Requirements Document
VAPER	:	Valve Performance Evaluation Rig
VDS	:	Vibration-noise Diagnostics System
V&V	:	Verification & Validation
WENRA	:	Western European Nuclear Regulators Association
WHO	:	World Health Organization