TOPSAFE

Dubrovnik, Croatia, 30.09 - 3.10.2008

TopSafe 2008
Transactions

Dubrovnik, Croatia
30.9. - 3.10.2008

HND

AEN NEA

IAEA
International Atomic Energy Agency
Atoms For Peace

European Nuclear Society
Rue Belliard 65
1040 Brussels, Belgium
Phone + 32 2 505 30 54
Fax +32 2 502 39 02
E-mail ens@euronuclear.org
Internet www.euronuclear.org

ISBN 978-92-95064-06-5

These transactions contain all contributions submitted by 30 September 2008.

Safety of Forthcoming Reactors

TOPSAFE
Dubrovnik, Croatia, 1 - 3 October 2008
International Topical Meeting on Safety of Nuclear Installations

AP1000: The PWR Revisited

Paolo Gaio
Westinghouse Electric Company LLC
gaio-p@notes.westinghouse.com

ABSTRACT

The distinguishing features of Westinghouse's AP1000 advanced passive pressurized water reactor are highlighted. In particular, the AP1000's passive safety features are described as well as their implications for simplifying the design, construction, and operation of this design compared to currently operating plants, and significantly increasing safety margins over current plants as well. The AP1000 design specifically incorporates the knowledge acquired from the substantial accumulation of power reactor operating experience and benefits from the application of the Probabilistic Risk Assessment in the design process itself. The AP1000 design has been certified by the US Nuclear Regulatory Commission under its new rules for licensing new nuclear plants, 10 CFR Part 52, and is the subject of several combined Construction and Operating License for USA utilities applications. Currently the AP1000 design is being assessed against the EUR Rev C requirements for new nuclear power plants in Europe.

1 BACKGROUND

For nearly two decades, Westinghouse has pursued an improved pressurized water reactor (PWR) design. The result of this commitment is the AP1000, a simpler and more economical PWR. The design began to develop in the late 1980s in conjunction with the development of the "Advanced Light Water Reactor Utility Requirements Document (URD)." The URD, drafted under the direction of the Electric Power Research Institute (EPRI), came to embody the policy and design requirements of US power utilities for the next generation of nuclear power plants in the US. These requirements were also endorsed by the US Nuclear Regulatory Commission (NRC). In Europe the corresponding body of design requirements and expectations developed as the European Utility Requirements (EUR).

The URD addresses evolutionary and passive light water reactors. The two classifications have different requirements. Expectations are much higher for passive designs. Indeed, more should be expected from designs that are not constrained to follow the existing models. For example, passive designs are expected to be able to achieve and maintain safe shutdown for 72 hours following the initiation of a design basis event without needing operator action. The corresponding expectation for an "evolutionary" plant is 30 minutes before the operator must take action to protect the core. As defined by the URD, a passive reactor is also "simpler, smaller and much improved..."

Simplification is a major requirement of the URD and a major characteristic of the AP1000.

2 THE AP1000 OVERVIEW

AP1000 is designed around a conventional 2-loop, 2 steam generator primary system configuration that is improved in several details. AP1000 is rated at 3400 MW(t) core power and, depending on site conditions, nominally 1117 MW(e). The core contains 157 fuel assemblies, similar to Doel 4 and Tihange 3. AP1000 features passive emergency core cooling and containment cooling systems. This means that active systems required solely to mitigate design basis accident conditions have been replaced in AP1000 by simpler, passive systems relying on gravity, compressed gases, or natural circulation to drive them instead of pumps. AP1000 also does not require safety-grade sources of ac power. Class 1E batteries provide for electrical needs during the unlikely scenario requiring the activation of the passive emergency system.

Compared to a standard plant of similar power output, AP1000 has 35% fewer pumps, 80% less safety- class piping, and 50% fewer ASME safety class valves. There are no safety-grade pumps. This allows AP1000 to be a much more compact plant than earlier designs. With less equipment and piping to accommodate, most safety equipment is installed within the containment. Because of this, AP1000 has approximately 55% fewer piping penetrations in the containment than current generation plants. Seismic Category I building volume is about 45 % less than earlier designs of comparable power rating. Figure 1 depicts the compact AP1000 station. Figure 2 compares the essential nuclear island building footprints to a typical, currently operating PWR. Seismic Category I buildings are shown in bold outline.

Here is a comparison of AP1000 safety margins to those of a currently operating plant.

Table 1: AP1000 safety margins

	Watts Bar	AP1000
Margin to DNBR, Loss of flow, %	14	16
SG tube rupture	Operator action required in 15 minutes	No operator action required
Small break LOCA Peak clad temperature, C	10 mm break Core uncovered PCT = 608C	20 mm break Core stays covered
Large break LOCA peak clad temperature, C	977	< 871

With a relatively large pressurizer, the AP1000 is more accommodating to transients and is, therefore, a more forgiving plant to operate.

The AP1000 is designed in accordance with the principles of ALARA to keep worker dose As Low As Reasonably Achievable. Features such as an integrated reactor vessel head package for quicker removal reduce the time required to do the job, and, therefore, reduce worker exposure. Attention to shielding, establishing distance from radiation sources, using low cobalt alloys, and using remote tooling or controls, are among the approaches that will minimize exposure throughout the plant. This is an area that has greatly benefited from operating plant experience.

Before delving into the further details of the AP1000 and how it is constructed, let us first review the regulatory status of this design.

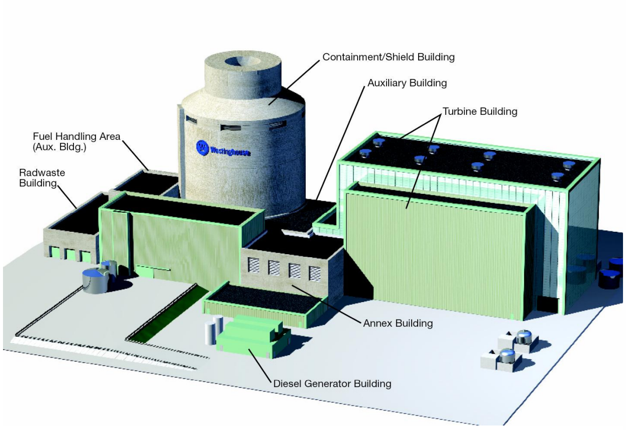

Figure 1: AP1000 Station

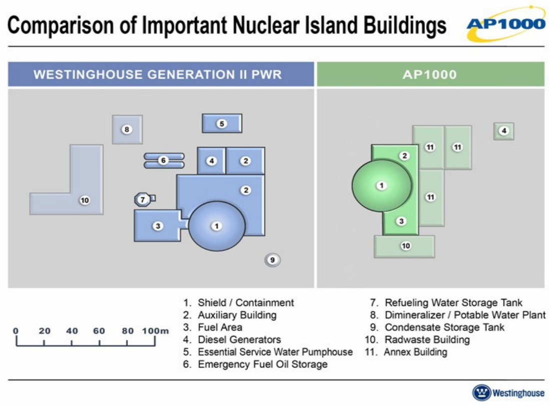

Figure 2: Seismic Category I Building Comparisons

3 AP1000 LICENSING AND REGULATORY STATUS

Nuclear power plants currently operating in the US were licensed under Title 10 CFR Part 50. In 1989 the US Nuclear Regulatory Commission (NRC) established alternative licensing requirements under 10 CFR Part 52. Prior to 1989 and under Part 50, all aspects of licensing from the design of the nuclear steam supply system to site-related topics remained open until after the plant was constructed. This left all aspects of a plant license application unsettled – and at risk - until virtually the entire plant capital investment was made. The current regulations under Part 52 ensure that all significant licensing issues have been resolved early in the process and with a high degree of finality.

Under Part 52 regulations, a plant design can be submitted for NRC Design Certification. The applicant is the plant design organization and the certification is generic and independent of any particular plant site. **NRC approved and certified the AP1000 design** under 10 CFR Part 52 in December 2005. The certification is valid for 15 years. Westinghouse submitted the AP1000 application in March, 2002.

Similarly, individual plant sites can be generally approved for construction of a nuclear plant through the Early Site Permit process under 10 CFR Part 52. This approval covers all elements affecting site suitability except for the specific effects of a particular plant design. These permits are valid for 10 to 20 years and can be extended for an additional 10 to 20 years.

With a design approved and certified and with a site that has received a permit, it then remains to merge these in order to actually proceed to construct and operate a specific nuclear power plant design at a specific site. This marriage of the two is the combined Construction and Operating License (COL) application. This application is made to the NRC by the site owner. Once the COL is granted by NRC, construction at the site may proceed.

This leaves the final step in the licensing process which is a verification that the plant has been constructed and will operate in conformance with the previously issued COL. This is accomplished by the Inspection, Tests, And Acceptance Criteria (ITAAC). Specific requirements for ITAACs for a particular case are established along the way in conjunction with the Final Design Certification and the COL applications.

Figure 3 summarizes all of this and identifies the US utilities that have declared that they will pursue a COL application. With the design certified for AP1000, preparing applications for COLs based on the AP1000 design can proceed directly

- 10 CFR Part 52 (operating plants licensed under earlier 10 CFR Part 50)
- Resolve licensing issues early in the process and with high degree of finality

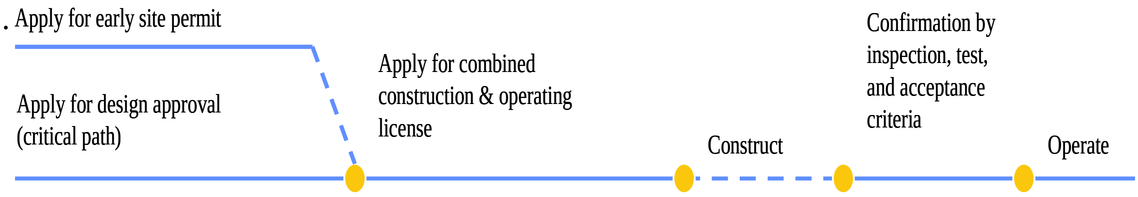

Figure 3: Licensing and Regulatory Status

4 AP1000 PASSIVE SAFETY SYSTEMS

What is meant by passive safety systems, the major differentiating feature of the AP1000? Let us start with the emergency core cooling system. This system comes into play only during transients or accidents which cannot be handled by the first-line of defense: the non-safety grade systems. In the current Generation II plants, the emergency core cooling system consists of redundant trains of high pressure and low pressure safety injection systems driven by pumps. These pumps force water into the primary system to replace core coolant in the event of a loss of coolant accident. Such pump-driven systems are termed “active” systems. The pumps take suction from tanks of borated water, valves are opened, and water is sent to the reactor vessel to cool the fuel rods. To increase reliability, multiple redundant trains may be installed. The net result is a substantial amount of machinery standing by for a call to action that designers and operators work very hard to never need.

By contrast, the AP1000 passive core cooling system uses staged reservoirs of borated water that are designed to discharge into the reactor vessel at various threshold state points of the primary system. To begin the description, let us first see the configuration of the AP1000 reactor primary coolant system shown in Figure 4. Now we can attach the essentials of the passive emergency core cooling system, as illustrated in Figure 5. There are three sources of borated replacement coolant and three different means of motivating the injection in AP1000:

1) Two core makeup tanks (CMT). Each CMT is directly connected to a RCS cold leg by an open “pressure balance” line. The balance line enters the CMT at the top of the tank, as shown in the figure. With outlet valves closed, the system is static. When actuated and check valves opened, water is forced out of these tanks and into the reactor vessel depending on and motivated by conditions in the cold leg via the always open balance line. Water from the RCS cold leg, which is hotter than water in the CMTs, will force the injection by its expansion into the CMT. If the cold leg is full of steam, steam will force the injection. CMTs are the first to actuate for smaller primary system breaks.
2) Two accumulators (ACC). These spherical tanks are 85% full of borated water and pressurized to 700 psig with nitrogen. Check valves open when pressure in the reactor vessel drops below 700 allowing the water in the tanks to flow into the reactor vessel. Large break LOCAs, which cause rapid system de-pressurization, will result in the accumulators being the first to respond.

3) The in containment refueling water storage tank (IRWST). Located above the RCS piping, the IRWST will discharge by gravity to the reactor vessel after the RCS has been de-pressurized by a break or by the automatic depressurization system, also shown in Figure 5. Flow is initiated by a depressurization signal which activates squib valves which open using an explosive charge. The squib valves are in series with check valves in the injection lines.

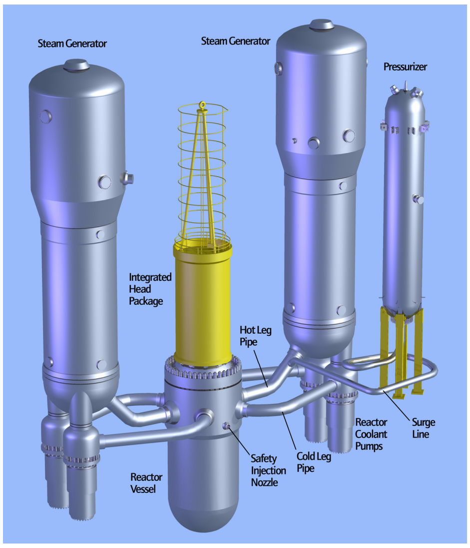

Figure 4: Primary System

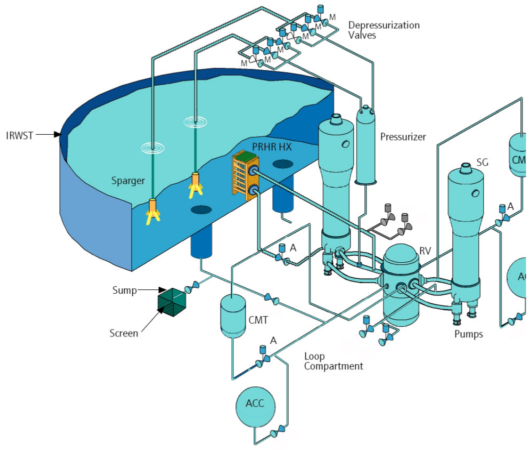

Figure 5: AP1000 Passive Core Cooling

These injection sources are connected to two Direct Vessel Injection nozzles on the reactor vessel dedicated solely for this purpose. The passive emergency core cooling system components are all located within the containment vessel. Without pumps to run, there is no need for emergency ac electrical power to maintain operation during an event. Any electrical power needed for the few safety valves and actuators that require it comes from 1E dc power, backed up by 1E batteries.

The injection system is enabled by an automatic depressurization system which executes a staged depressurization of the primary system initiated from any actuation of the CMTs that reaches pre-set water levels in those tanks.

The IRWST is part of the passive decay heat removal system. A heat exchanger inside the IRWST has an inlet from the reactor coolant system (RCS) hot leg and an outlet into the RCS cold leg. In the event of loss of RCS heat removal from the steam generators, the IRWST will absorb heat from the heat exchanger while primary system coolant circulates through the exchanger by natural circulation. After several hours of operation, the IRWST water will begin to boil. Steam from IRWST will begin to condense on the containment walls. The condensate will then be directed by a safety-grade guttering system back to the IRWST to continue the cycle.

The steel containment vessel located inside the concrete shield building provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Heat is removed from the containment vessel by the continuous natural circulation of air within the shield building/containment vessel annulus. During a design basis accident, the air cooling is supplemented by evaporation of water. This cooling water drains by gravity from a tank located on top of the containment shield building. The water runs down over the steel containment vessel, thereby enhancing heat transfer. This passive containment cooling system design eliminates the safety-grade containment spray and fan coolers required for a conventional plant.

ey elements of this system were extensively tested and documented as part of the basis for receiving NRC‘s Final Design Certification. Figure 6 indicates the kind of simplification that results from AP1000’s passive system versus a standard PWR emergency system.

5 SEVERE ACCIDENT MITIGATION

The AP1000 is designed to retain melted core debris within the reactor vessel. To start with, the reactor vessel has no penetrations in the bottom head. In case of a severe accident, cooling water from the large IRWST can be used to flood the reactor cavity and cool the outside of the reactor vessel. The arrangement is shown in Figure 7. Specially designed reactor vessel insulation forms an annulus that allows cooling water to directly contact the vessel. Vents are provided for steam to escape the annulus. To complete the description, the vented steam will condense on the containment walls and be directed back to the cavity.

Conventional **AP1000**

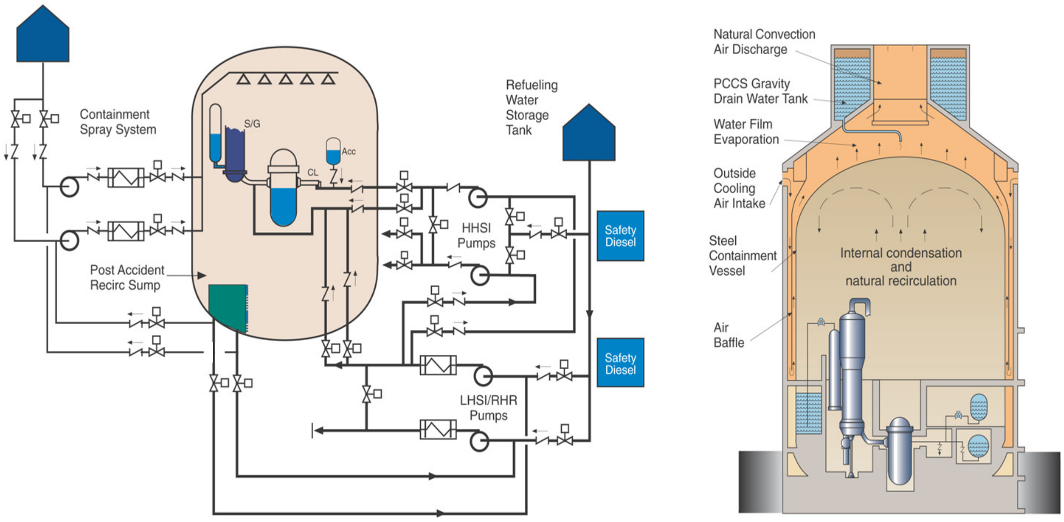

Figure 6: Reduced Complexity

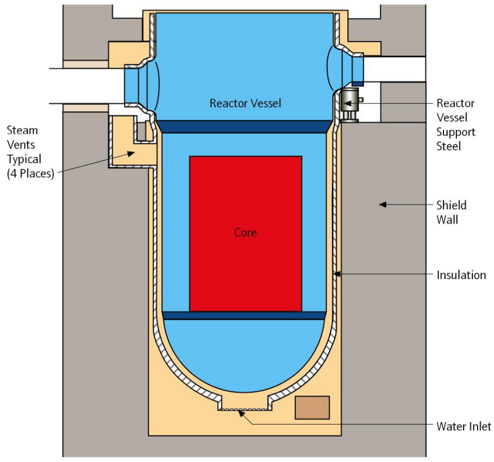

Figure 7: Severe Accident Design

6 PROBABILISTIC RISK ASSESSMENT

One of the advancements that benefits the AP1000 is the further development of probabilistic risk assessment tools (PRA) and the application of these tools to the design process itself. The result for AP1000 has been a more effective combination of redundancy and diversity. This includes the defense-in-depth design that utilizes non-safety controls and systems as the first line of defense. If the first line systems are not capable of handling the event, the passive safety systems come into play. As revealed by the PRA, the risk of core damage and large radioactive release for AP1000 is extremely low. Here are the results for combined conditions of power, shutdown, and internal events, as well as fire and flood events:

- Core damage frequency, 5x10-7
- Large release frequency, 6x 10 -8.

For some perspective, here are some comparative results for core damage frequency:

Table 2: Comparative results for core damage frequency

US NRC requirement	1×10^{-4}
Current plants	5×10^{-5}
URD requirement	$<1 \times 10^{-5}$
AP1000	5×10^{-7}

The AP1000 PRA led to the following statement by the US Advisory Committee for Reactor Safeguards in their report on AP1000 certification:

"This PRA was well done and rigorous methods were used…The fact that the PRA was an integral part of the design process was significant to achieving this estimated low risk."

7 AP1000 REACTOR COOLANT PUMPS

Among the improvements embodied in the AP1000 are the reactor coolant pumps. AP1000 employs four canned motor pumps, two in each loop, as can be seen in Figure 4. Although such pumps have been used for decades in naval nuclear power plants, commercial PWRs have not employed them recently because the capacities required for Generation II nuclear plants began to exceed the capacity range of canned pumps prevailing at that time. However, in the meantime, the capacity of canned motor pumps has increased. The advantages of the canned motor design over conventional reactor coolant pumps are:

- Elimination of the shaft seal and the system needed to maintain seal injection
- By eliminating this seal and seal injection, a potential leakage path of primary coolant and a source of small break LOCA are also eliminated
- Canned motor pumps require very little or no maintenance and thereby also help lower worker dose.

8 AP1000 INSTRUMENTATION AND CONTROL SYSTEMS

The Westinghouse AP1000 instrumentation and control (I&C) system is comprised of the following subsystems:

Operation and control centers (OCS)
Data display and processing (DDS)
Protection and safety monitoring (PMS)

Plant control (PLS)
Main turbine control and diagnostics (TOS)
Incore instrumentation (IIS)
Special monitoring (SMS)
Diverse actuation (DAS)
Radiation Monitoring (RMS)
Seismic Monitoring (SJS).

Following are highlights of some of these systems:

The OCS provides the human interface control facilities: the main control room, the technical support center, the remote shutdown workstation, the emergency operations facility, local control stations, and the associated workstations for each of these centers. The main control room, for example, is environmentally controlled and designed in conjunction with a comprehensive human factors engineering program conducted at Westinghouse. This program included an extensive operating experience review. Figure 8 shows a representative main control room layout for the AP1000.

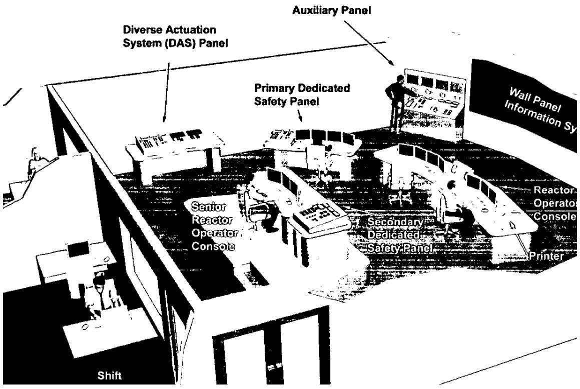

Figure 8: Control Room

The plant control system (PLS) provides for control rod motion and position monitoring and controls the transport of heat energy from the nuclear reactor to the main steam turbine by means of the following major control functions:
Pressurizer pressure and level
Steam generator water level
Steam dump (turbine bypass)
Rapid power reduction
Various component controls (pumps, motors, valves, breakers, etc.)

The system provides for automatic and manual control.

The special monitoring system (SMS) is a non-safety-related system comprised of subsystems that interface with the I&C architecture to provide specialized diagnostic and long-term monitoring functions for detection of metallic debris in the reactor coolant system, core barrel vibration, and reactor coolant pump monitoring.

The diverse actuation system (DAS) provides I&C functions necessary to reduce the risk associated with a postulated common-mode failure in the PMS. The types of common-mode failures addressed by the DAS include software design errors, hardware design errors, and test and maintenance errors.

9 CONCLUSION

The AP1000 is a PWR design that offers power generating companies a clear and practical alternative for new generating capacity. It was designed to be competitive with fossil fuel plants and will be overwhelmingly so as actions are implemented to reduce greenhouse gas emissions. With decades of operating experience to draw on, AP1000 incorporates proven technologies in a new combination to consolidate the advantages of nuclear power units while reducing their cost and complexity. It is important to recognize that among all the advantages of AP1000, it is also a demonstrably safer plant and an advanced design that has already been certified by the US

TOPSAFE
Dubrovnik, Croatia, 30.09 - 3.10.2008

The Safety Concept of the SWR 1000 with Active and Passive Safety Systems

Werner Brettschuh, Doris Pasler
AREVA NP GmbH
Strahlenbergerstraße 17, 63067 Offenbach
werner.brettschuh@areva.com, doris.pasler@areva.com

ABSTRACT

The SWR 1000 blends years of experience in design, construction and operation of BWRs with new concepts to achieve an optimum blend of increased safety and reduced costs. It has been developed to provide a reliable source of economically competitive and safe electricity.
This is achieved through the use of passive safety systems which function solely according to basic laws of nature such as gravity, natural convection and heat transfer induced by temperature differentials. In keeping with their principles of operation, the passive systems are of simple design and perform their intended functions without any need for instrumentation and control (I&C) equipment or an active supply of energy, such as electric power.
A key feature of the safety concept of the SWR 1000 is the fact that all design basis accidents can be controlled using just passive systems alone. Nevertheless, service-proven active safety systems are still intended to operate, if possible, before passive safety equipment takes over. The functional scope and degree of redundancy of these active systems can, however, be reduced.
This safety concept is supplemented by systems and actions for controlling a postulated core melt accident; i.e. for retaining the molten core inside the reactor pressure vessel (RPV) by external cooling.
By this way the consequences of this type of severe accident will remain restricted to within the plant itself and no emergency response actions will be necessary in the plant environs. Compared to existing plants, this capability for core melt retention represents an additional level of safety within the overall concept known as "defense-in-depth".

1 INTRODUCTION

The overall safety concept of nuclear power plants is basically built up around three safety objectives:

1. Attainment and maintenance of sub-criticality
2. Assurance of core cooling, and
3. Confinement of radioactivity.

To meet these objectives both during normal operation and in the event of off-normal operating conditions and accidents, safety systems are required to perform certain protective functions such as:

• Reactor shutdown
• Reactor pressure relief
• Core cooling
• Removal of residual heat from the containment
• Isolation of the reactor coolant pressure boundary
• Containment isolation.

To guarantee the necessary degree of functional reliability, these systems must satisfy stringent requirements in terms of redundancy, diversity, physical separation and component quality.

Until now, the safety systems have mainly comprised active systems; i.e. systems that can only perform their designated safety functions with the aid of a secure external energy supply, whether in the form of electric power or actuating fluids. Of course, the auxiliary systems supplying this energy have to satisfy the same requirements as the safety systems themselves.

The aim pursued in developing a new safety concept for the SWR 1000 was to guarantee the functional capability and reliability of safety equipment by providing simple and robust system designs and thus to increase plant safety as well as economic efficiency even further.

This is achieved through the use of passive safety systems which function solely according to basic laws of nature such as gravity, natural convection and heat transfer induced by temperature differentials. In keeping with their principles of operation, the passive systems are of simple design and perform their intended functions without any need for instrumentation and control (I&C) equipment or an active supply of energy, such as electric power.

A key feature of the safety concept of the SWR 1000 is the fact that all safety functions can be performed by passive systems. Design basis accidents can be controlled using just passive systems alone. Nevertheless, service-proven active safety systems are still intended to operate, if possible, before passive safety equipment takes over. The functional scope and degree of redundancy of these active systems can, however, be reduced.

This safety concept is supplemented by systems and actions for controlling a postulated core melt accident; i.e. for retaining the molten core inside the reactor pressure vessel (RPV) so that even the consequences of this type of severe accident will remain restricted to within the plant itself and no emergency response actions will be necessary in the plant environs. Compared to existing plants, this capability for core melt retention represents an additional level of safety within the overall concept known as "defences-in-depth".

All new passive systems have been tested either in full scale or in a scaled configuration to ensure the proper function of these systems. They will be tested again with full-scale, prototype components. In addition the external cooling of RPV has been tested successful [3].

2 SAFETY SYSTEMS

In line with the philosophy underlying the new safety concept of the SWR 1000, both active and passive systems are available for all safety functions. The passive systems are – just like the active systems – of multiple redundancy. In addition, active and passive systems provide functional diversity for one another. In the postulated event of failure of all active systems, accident control is still possible using only passive systems that need no I&C signals or power supplies to operate.

An overview of the safety systems of the SWR 1000 is given below, broken down according to the various safety functions listed above. The individual systems are not, however, described in detail. The active safety systems planned for the SWR 1000 are functionally equivalent to the systems already familiar from existing plants such as Gundremmingen B+C.

2.1 Systems for Reactor Shutdown

Diverse systems are available for shutdown of the reactor. These include the control rods, with their diverse drive systems:

- Electric motor drive for operational shutdown processes (control rod drive (CRD))
- Hydraulic drive for reactor scram

The SWR 1000 is also equipped with a fast acting boron injection system, which causes reactor shutdown independent from the control rods and is completely independent of control rod operations (Fig. 1).

The scram system is based largely on the accumulator tank concept implemented in German BWR plants, whereby the energy required for fast control rod insertion by hydraulic means is stored in tanks under nitrogen pressure. The tank pressure for the SWR 1000 is provided by steam pressure, much like a PWR pressurizer, to provide the required driving head. The water-filled tanks' steam pressure blanket is generated by electric heaters in the upper area of each tank. This modification enables the reduction of the tank size, and prevents nitrogen from entering the RPV in the event of a malfunctioning tank isolation valve.

The scram system of the SWR 1000 also differs from existing collector tank systems because the two ring lines are not interconnected. Each ring line is supplied by two scram tanks, and supplies half of the control rod drives. Insertion of half of the control rods is sufficient to bring the core rapidly to the 'hot sub-critical' condition. This means that the connections between the two ring lines are eliminated, and the system remains separated into two redundant subsystems. Each scram tank has the capacity to insert half of the control rods within the required time. During power operation the tanks are each isolated via a quick-opening valve. Tripping of the scram function is initiated via solenoid pilot valves or via parallel diaphragm pilot valves that are actuated by Passive Pressure Pulse Transmitters (PPPTs).

The fast acting boron injection system (FABIS) is also based on the pressure tank concept: The quantity of pentaborate solution required for hot and cold sub-criticality is stored in a tank and injected into the RPV.

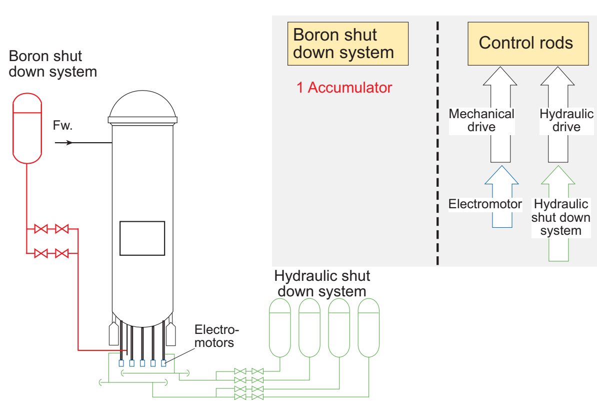

Figure 1: Shutdown systems

2.2 Systems for Core Cooling

The active system provided for core cooling in the event of a loss-of-coolant accident (LOCA) is the two-train residual heat removal (RHR) system operating in the low-pressure coolant injection (LPCI) mode ("active core flooding"). The RHR trains are connected to the emergency power supply system. The RHR system is designed to pump water from the pressure suppression pool into the RPV once reactor pressure has dropped below 10 bar. The pumping capacity of just one RHR train is sufficient for controlling such events. During flooding, heat is also removed from the containment to the plant's ultimate heat sink via the RHR heat exchangers.

Reactor depressurization in the event of a LOCA is assured by eight safety-relief valves (SRVs), four of which operate according to the depressurization principle and four according to the pressurization principle. Each main valve is provided with redundant and diverse means of actuation in the form of battery-backed solenoid pilot valves as well as pilot valves operated by PPPTs. The main valves are latched in the open position to ensure that they stay open when the reactor has been depressurized. The high degree of redundancy and diversity of the safety-relief valve system together with its capabilities for active and passive actuation guarantee a high level of system availability.

Passive removal of residual heat from the reactor is performed by the emergency condenser system (Fig. 2). The emergency condensers (ECs) are tubular heat exchangers which are submerged in the core flooding pools and are connected to the RPV by non-isolatable inlet and outlet lines. They are arranged such that if reactor water level should drop, steam will flow into the heat exchanger tubes where it condenses. Thus they perform two functions at once: heat removal from the RPV (by transferring the residual heat to the water of the core flooding pools through condensation) and core flooding (by returning the resulting condensate to the RPV).

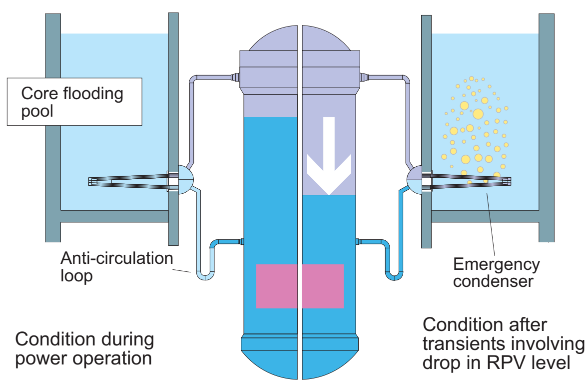

Figure 2**:** Emergency condenser

2.3 Systems for Removal of Residual Heat from the Containment

In the case of all accidents which lead to the reactor becoming isolated from the main heat sink (e.g. loss of the main heat sink, LOCAs outside containment or natural and external man-made hazards), the two-train RHR system operating in the "flooding pool/pressure suppression pool cooling" mode is used to remove heat from the containment to an ultimate heat sink.

The containment cooling condenser (CCC) system, which mainly consists of four tubular heat exchangers, is provided for passive heat removal from the containment (Fig. 3).

The CCCs condense steam released inside the containment in the event of an accident and return the condensate to the core flooding pools. Condensation is effected by water from the shielding/storage pool located directly above the containment which flows by natural circulation through the tubes of the CCCs, thus removing the heat from the containment to the pool above. The water inventory of the shielding/storage pool is sized such that passive containment heat removal by the CCCs can continue for a prolonged period of time (without any need for further actions). If makeup water is supplied to the pool – e.g. via connected systems or external pumps – the grace period available before the entire pool water inventory has evaporated can be extended indefinitely.

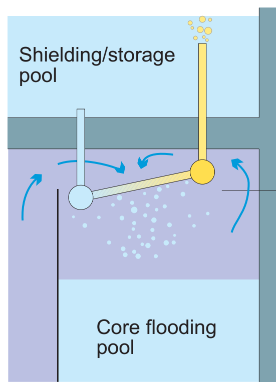

Figure 3: Containment cooling condenser

2.4 Systems for Reactor Pressure Relief

Following isolation of the RPV from the main heat sink, reactor pressure relief is performed by the eight SRVs (see below). Active actuation of the system-fluid-operated main valves is effected by the solenoid pilot valves. Passive actuation of the SRVs for the pressure relief function is performed by spring-loaded pilot valves (Fig. 4).

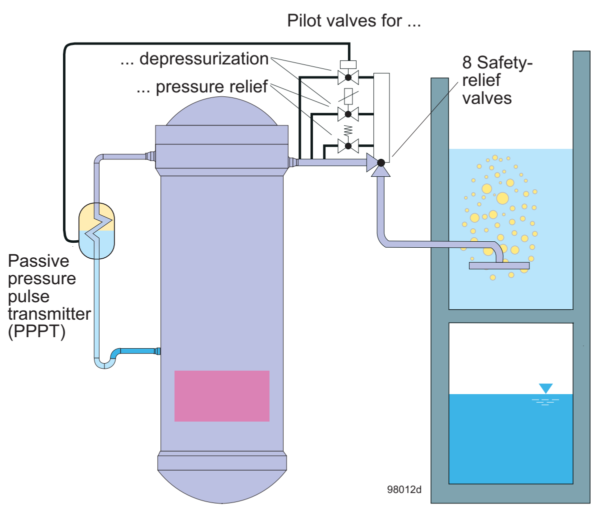

Figure 4: Safety relief valve system

2.5 Systems for Reactor Coolant Pressure Boundary and Containment Isolation

Active isolation of the reactor coolant pressure boundary (RCPB) and of the containment is initiated by I&C equipment according to a concept comprising containment isolation at the main steam line penetrations, isolation at the feedwater line penetrations, and isolation at the penetrations of auxiliary system piping.

Two containment isolation valves (main steam isolation valves (MSIV)) are provided in each of the three main steam lines: one inboard gate valve and one outboard globe valve. These system-fluid-operated isolation valves are each actuated by two pilot valves (one solenoid valve and one pneumatic valve operated by PPPTs). The two feedwater lines are each provided with two containment isolation valves: one non-piloted check valve (inboard) and one system-fluid-operated gate valve outside the containment. The gate valve is actuated by two pilot valves (one solenoid valve and one pneumatic valve operated by PPPTs). The inclusion of the system-fluid-operated gate valve – serving as a second isolation valve of diverse design – has enabled the probability of loss of reactor coolant inventory due to a feedwater line break occurring outside the containment to be considerably reduced even further. Finally, appropriate means are likewise provided to isolate the RCPB from all other systems that penetrate the containment, except for those required to perform safety-related functions.

According to the design concept of the SWR 1000, there are no other high-energy piping systems conveying reactor coolant that are situated outside the containment apart from the main steam and feedwater lines. Hence only these lines are equipped with passive isolation devices.

2.6 Systems for In-Vessel Core Melt Retention in the Event of a Severe Accident

The new safety concept of the SWR 1000 makes the probability of a core melt accident even lower than at existing plants. Provisions have nevertheless been made to ensure control of this hypothetical event.

If there should be an impending risk of core melt, water from the core flooding pools is allowed to flow down by gravity into the lower section of the drywell surrounding the RPV through a special drywell flooding line. The RPV then becomes surrounded by water up to the elevation of its support skirt. If a pool of molten core material should collect in the bottom head of the RPV, enough heat can be removed to this water through the RPV wall without causing melting of the wall [2]. Thus it is possible to retain the melt inside the RPV. The steam produced by evaporation of the water on the RPV exterior is condensed by the CCCs and the resulting condensate circulates back to the drywell by way of the core flooding pools (Fig. 5).

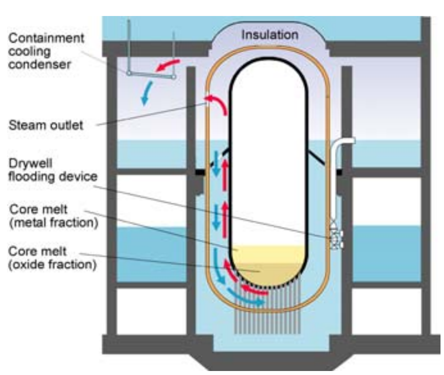

Figure 5: Severe accident control

3 REDUNDANCY CONCEPT

Active and passive systems together fulfil the requirements for redundancy according to the single failure concept. In other words, the specified safety functions will still be performed even if systems should be unavailable on demand as a result of:

- Consequential failure
- Single failure
- Maintenance.

For the passive systems, the scope of failures that need to be postulated can be reduced since they do not contain any active components for which a single failure would have to be assumed and also since unavailability due to maintenance work can be disregarded because they are located inside an inerted containment which is not accessible during power operation of the plant. Single failures in passive components such as piping can be ruled out due their high quality of fabrication. For these reasons, the redundancy concept presented below still ensures that the specified safety functions can be fulfilled by just passive systems alone.

The requirement for diversity, aimed at avoiding common cause failures, is met by employing different passive systems, or active systems together with passive systems of diverse design, for all safety functions. In addition, the systems are installed with physical separation in order to protect them against the effects of flooding or fire, etc.

Fig. 6 provides a summary of the active and passive systems provided to fulfil the designated safety functions.

Safety Function		Passive Systems (classification acc. To IAEA TECDOC 626)		Active Systems
Reactivity control		**CRD with scram system** Activation: • PPPT (Passive C) • I & C (Passive D) **Boron injection system** (Passive D)		**157 fine motion control rod drives** (FMCRD)
Containment isolation at main steam lines		**2 MSIVs per steam line (diverse design)** (system fluid operated) Activation: • PPPT (Passive C) • I & C (Passive D)		
Reactor pressure relief		**8 SRVs**	Activation: • SLPV (Passive C) • I & C (Passive D)	
Reactor depressurization		**8 SRVs**	Activation: • PPPT (Passive C) • I & C (Passive D)	
RHR from RPV	HP	**4 Emergency condensers** (Passive B)		
RHR from RPV	LP	**4 Emergency condensers** (Passive B)		**2 RHR and LPCI system trains** (I & C and EPS)
Core flooding	LP	**4 Core flooding lines** (Passive C)		**2 RHR and LPCI system trains** (I & C and EPS)
Heat removal from containment		**4 Containment cooling condensers** (Passive B)		**2 RHR and LPCI system trains** (I & C and EPS)

Figure 6: Passive and active systems for accident control

The redundancy provided for the safety functions of heat removal from the RPV, core flooding and heat removal from the containment, largely fulfilled by innovative equipment, is as follows:

	Active Systems	Passive Systems				Effective Capacity
Heat removal from RPV, NON-LOCA (High pressure)	8 Safety Relief Valves + (1 of 2 Feedwater pumps or 2 of 2 RWCS return pumps)	4 Emergency condensers				
		50%	50%	50%	50%	
	Not considered	✓	✓	✓	✓	200%

Figure 7: Capacities for heat removal from RPV

SF ... Single failure CF ... Conseq. failure	Active Systems		Passive Systems				Effective Capacity
Core flooding upon LOCA	2 RHR and LPCI systems		4 Flooding lines				
	100%	100%	100%	100%	100%	100%	
Feedwater line break	CF	Maintenance	SF	✓	✓	✓	300%
Core flooding line break	Maintenance	SF	CF	✓	✓	✓	300%
	Maintenance	✓	CF	SF	✓	✓	300%
	100%	100%	50%	50%	50%	50%	
Leak below core (15 cm^2)	Maintenance	SF	✓	✓	✓	✓	200%

Figure 8**:** Capacities for core flooding in the event of a LOCA

SF ... Single failure	Active Systems		Passive Systems				Effective Capacity
Heat removal from containment	2 RHR systems		4 containment cooling condensers				
	100%	100%	50%	50%	50%	50%	
Failure assumptions	SF	Maintenance	✓	✓	✓	✓	200%

Figure 9**:** Capacities for heat removal from the containment

It can thus be seen that the diverse and redundant safety systems planned for the SWR 1000 provide adequate functional capacity, i.e. ≥ 100%, even when failures postulated according to the single failure concept are taken into account. The degree of redundancy of the passive systems has been selected such that accidents can be controlled by just the passive systems alone, even upon loss of the active RHR systems due to a consequential failure combined with maintenance or a single failure.

4 ELECTRICAL AND I&C SYSTEM DESIGN

The design of the electrical and I&C systems for the SWR 1000 is based on the requirements associated with normal plant operation as well as on fulfillment of the safety objectives for the reactor and its auxiliary systems. Whereas the objectives applying to normal operation (e.g. feeding of power into the main offsite power system, supply of power to electrical loads, automation for reducing the workload on operating personnel, and monitoring and display of plant conditions) are centered on achieving maximum system and component availability, and thus maximum availability of the overall plant, different objectives are pursued in the case of safety-related tasks, as already described above.

A particularly unusual aspect encountered in designing the electrical and I&C systems was the fact that they have to perform their tasks in conjunction with passive systems – systems that do not actually exist from an electrical or I&C viewpoint because they require neither power supplies nor I&C signals to operate.

Since, according to the SWR 1000's safety concept, all postulated accidents can be controlled by passive systems alone, this means that it is possible to limit the degree of redundancy of the electrical and I&C systems to 2 x 100%, or dual redundancy. The degree of redundancy generally prescribed in codes and standards for accident control assuming single failures and repairs is thus provided jointly by active and passive systems.

Naturally, for safety-related tasks for which no passive systems are available (e.g. monitoring), the necessary degree of redundancy is provided only at the active equipment.

The configuration of the plant electrical systems can be described, in simple terms, as follows. The generator feeds power to the 400 kV main offsite power system connection via a generator transformer. Auxiliary power is tapped off between the generator breaker and the generator transformer and fed to the 10 kV switchgear via two auxiliary power transformers. In the event of loss of the normal power supply via the generator and main offsite power system, as well as loss of the standby offsite power system, certain electrical loads must remain in operation or come into operation in order for safety functions to be performed. In terms of power supply requirements, a distinction is made between two categories of electrical loads:

• Loads for which a period without power is permissible

• Loads which must remain in operation without interruption or which must be immediately started up.

The first group is connected to the three-phase AC distribution boards of the two-train emergency power supply system, while the second is supplied with power either from the 220 V DC system, via inverters or from distributed uninterruptible power supply equipment.

The I&C concept planned for operator control, monitoring and closed- and open-loop control is based on the following principles.

The technology implemented in the control rooms of today's nuclear power plants as well as in control rooms of the future is and will be characterized to a large extent by screen-based displays and screen-based control equipment. The design concept of the control room is therefore based on a powerful process information system for operating data analysis and storage, trend analyses, preventive maintenance, process optimization, support of the operating personnel in fault analysis and other tasks.

Widespread use of field bus systems can minimize cabling requirements. This is made possible by the availability today of bus systems with high data-transfer capacities, as well as distribution of I&C equipment throughout the plant (ensuring sufficiently short response times). Bus systems provided for safety systems are designed taking single failures into account. In non-safety-related I&C systems the bus systems can be configured to tolerate certain single failures.

The safety I&C with its programmable logic control systems for actuation of the active safety systems primarily comprises the reactor protection system which is configured in multiply redundant subsystems that are decoupled to provide non-interactive separation of the redundant trains at their interfaces. Measured data acquisition is performed with quadruple redundancy for limit signal generation in the data acquisition computers and for two times (2 out of 4) logic gating in the processing computers. The output signals are gated once more in 1 out of 2 logics and actuation signals are generated for controlling the active safety system components.

AREVA NP’s digital safety I&C platform TELEPERM XS is to be used for the reactor protection system of the SWR 1000. Analyses of this safety I&C equipment's expected unavailability under SWR 1000 boundary conditions have shown, for example, that an unavailability for reactor scram of 0.5 x 10-8 per demand can be achieved. This provides significant margins to the requirements specified in pertinent codes and standards.

5 PROBABILISTIC SAFETY ASSESMENT

By combining well-known active safety equipment with passive safety systems of diverse design [1], the effects of Common Cause Failures are significantly reduced and the frequency of core damage states caused by plant-internal events is two orders of magnitude lower than that of contemporary plants. In fact, the integral frequency of core damage states calculated by proven methods for initiating events occurring during power operation and plant shutdown is only 8.4 x 10-8 per year.

6 PROTECTION OF BUILDINGS AGAINST NATURAL AND MAN MADE HAZARD

The plant is designed in accordance with the European Utility Requirements and the Finnish requirements to withstand the effects of natural and external man-made hazards such as seismic events, aircraft crash (military fighter and large passenger aircraft) and explosion pressure waves. One of the goals in designing the plant was to accommodate the systems and components that require protection against these hazards in such a way inside the plant buildings that as few buildings as possible would have to be designed to withstand the loads from such events. Since all safety-related systems and components as well as those containing a high activity inventory are housed in the reactor building – except for the redundant emergency diesels along with their switchgear and the two safety-related closed cooling water and service water systems – the concept implemented for building protection is as follows:

The reactor building is the only building protected against all three major postulated hazards (seismic events, aircraft crash and explosion pressure waves). The buildings containing the emergency diesels and safety-related cooling water systems are protected against the effects of aircraft crash through physical separation (diesel buildings 120 m) and are designed to safely accommodate the loads imposed by a seismic event or an explosion pressure wave. An emergency control room building (bunker) is protected in the same way and is physically separated from the reactor supporting systems building housing the main control room. Since none of the other buildings contains safety-related equipment or components with a high activity inventory, they are only designed to withstand seismic loading according to standard industrial practices. These simplifications in system and component design reduce plant construction cost as well as, of course, plant operating cost since any decrease in construction cost always means less expenditure on inspection and maintenance.

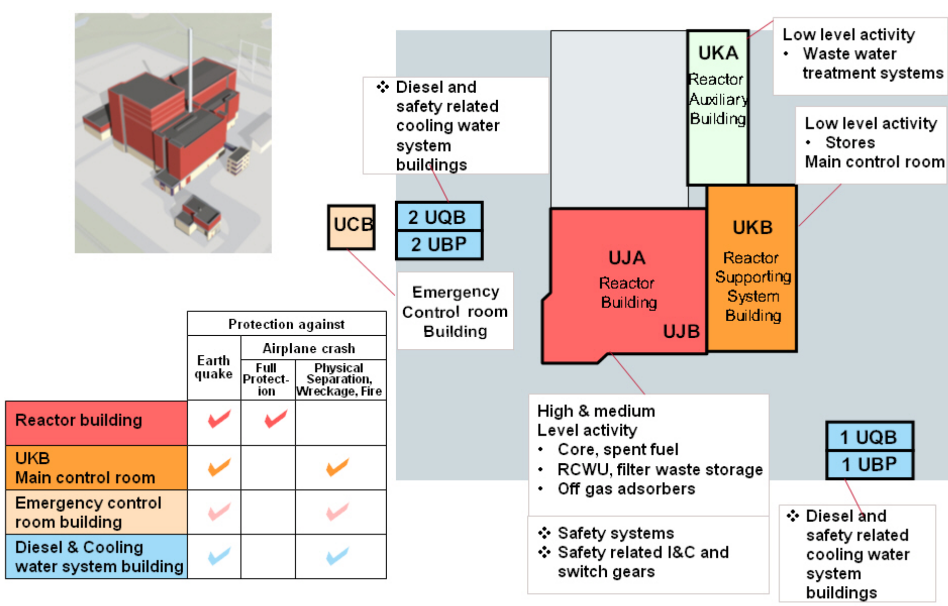

Figure 10: Protection against external events

7 CONCLUSION

Through the consistent deployment of passive safety systems to provide redundancy and diversity for active systems, the safety level of the SWR 1000 has been able to be significantly increased compared to existing plants. The advantages of the new safety concept are:

- Reduced susceptibility of safety systems to failures
- Larger safety margins
- Good plant behavior in the event of accidents due to the fact that conditions change at a slower rate
- Grace periods of several days after an accident before operator intervention is required
- Significantly reduced impact of operator error on reactor safety
- No need for large-scale emergency response actions such as temporary evacuation or relocation of the neighboring population following a core melt accident.

The use of passive systems of simple design combined with a corresponding reduction in the number of active safety systems has enabled not only plant safety but also the economic efficiency of the plant as a whole to be considerably increased.

The need to comply with nuclear codes and standards provided the framework for development of the new safety concept. In accordance with the original design development contract, the SWR 1000 was based on German codes and standards for nuclear power plant construction and operation. In the meantime, however, the design concept has also been assessed with respect to the European Utility Requirements (EUR), a set of guidelines issued by Europe's major nuclear power plant operators. Here, too, compliance has been verified. Furthermore, as part of preparations to submit a proposal to build Finland’s fifth nuclear power plant, the Finnish Radiation and Nuclear Safety Authority, STUK, has certified that the SWR 1000 is basically licensable according to the Finnish codes YVL [4].

REFERENCES

[1] ARVA NP GmbH, “SWR 1000 – General Plant Description”, Offenbach (2003)

[2] H. SCHMIDT, “Large Scale Verification of External RPV Cooling in Case of Severe Accident”, Proc. of ICAPP’04, Pittsburgh, PA USA, June 13-17, 2004.

[3] J. MESETH, “Experimental Verification of SWR 1000 Passive Components and Systems”, Proc. of ICAPP’04, Pittsburgh, PA USA, June 13-17, 2004.

[4] W. BRETTSCHUH, “Entering the Market”, Nuclear Engineering International, p. 28-30, March 2003.

TOPSAFE
Dubrovnik, Croatia, 30.09 - 3.10.2008

ACR-1000©*: Advanced CANDU Based on Proven Safety of CANDU Reactors

N. Popov, R. Ion, S. Yu, R. Duffey
Atomic Energy of Canada Limited
2251 Speakman Drive, Mississauga, Ontario, Canada, L5M 1B2

Corresponding author: Nik Popov popovn@aecl.ca

ABSTRACT

This paper provides an overview of the Advanced CANDU Reactor®-1000 (ACR-1000®) design focusing on the safety systems and functions that are based on decades of design development and R&D of different CANDU reactor designs in Canada.

The ACR-1000 developed by Atomic Energy of Canada Limited (AECL) is a 1200 MWe-class light-water-cooled, heavy-water-moderated pressure-tube reactor, which has evolved from the well-established CANDU® line of reactors. The ACR-1000 design retains the basic, proven, CANDU design features while incorporating innovations and state-of-the-art technologies to ensure fully competitive safety, operation, performance and economics. Improvements include greater operating and safety margins plus adherence and compliance with the latest safety thinking regarding external events and risk assessment.

The ACR-1000 design complies with all applicable Canadian Nuclear Safety Commission (CNSC) regulatory requirements. Although not mandatory in Canada, the ACR-1000 design takes into account all applicable international requirements as appropriate. Moreover, IAEA's safety standard "Safety of Nuclear Power Plants: Design Requirements", NS-R-1, has been used in the ACR-1000 design.

AECL has recently issued the ACR-1000 Generic Safety Case Report (GSCR) that provides a site-independent overview of the design, safety characteristics, and bounding safety analysis of the ACR-1000, which demonstrates design readiness and licensability in Canada and abroad.

1 INTRODUCTION

This paper provides an overview of the Advanced CANDU Reactor®-1000 (ACR-1000®) design, with a focus on the safety systems and functions. The paper demonstrates a good design balance by taking advantage of proven traditional CANDU features with a number of innovations that enhance the safety, operability and maintainability of the reactor. The ACR-1000 design is based on decades of design development and R&D of different CANDU reactor designs in Canada and internationally. The ACR-1000 complies with all applicable Canadian regulatory requirements and the IAEA safety guidelines.

1.1 Background

The ACR-1000 developed by Atomic Energy of Canada Limited (AECL) is a 1200 MWe-class light-water-cooled, heavy-water-moderated pressure-tube reactor, which has evolved from the well-established CANDU® line of reactors. CANDU 6 units have already been licensed since the early 1980s until 2007 in a number of countries around the world: Canada, Argentina, Republic of Korea, Romania, and China. There are 11 CANDU 6 units

* ACR-1000© and CANDU are registered trademarks of Atomic Energy of Canada Limited (AECL)

currently in successful operation, which have exceptional lifetime operating performance records. The ACR-1000 design retains the basic, proven, CANDU design features while incorporating innovations and state-of-the-art technologies to ensure fully competitive safety, operation, performance and economics. Improvements include greater operating and safety margins plus adherence and compliance with the latest safety thinking regarding external events and risk assessment.

With a 60-year design life, the ACR-1000 is based on the use of a modular horizontal fuel channel surrounded by a heavy water moderator, the same feature as in all CANDU reactors. Each unit has a nominal gross output of 1165 MWe with a net output of approximately 1085 MWe. The unit uses low enriched uranium fuel, and the CANFLEX® ACR fuel bundle, which has a lower linear rating and higher critical heat flux relative to the 37-element bundles used to date as the standard CANDU 6 fuel.

The major nuclear systems of the ACR-1000 are located in the Reactor Building (RB) and the Reactor Auxiliary Building (RAB). Safety enhancements made in the ACR-1000 encompass improved safety margins and reliability of Safety Systems, which include two Shutdown Systems, enhanced Emergency Core Cooling System, Emergency Feedwater System (defined as the Emergency Heat Removal System), Containment System, and the associated safety support systems.

The development of the ACR-1000 safety case is largely based on the ACR-1000 Design Program, which consists of three distinct project phases: the Product Definition phase, Basic Engineering Program, and Project Final Design phase. The Generic Safety Case Report (GSCR), that was issued in June 2008, reflects the Basic Engineering Program (BEP), with the major structures, systems and components designed, developed, refined and integrated to a more precise level of detail on a generic basis (meaning non site-specific, non project-specific) to allow speedy and timely adaptation for specific customers and multiple project implementations. Generic system documentation covers system design requirements and descriptions, detailed flow sheets, system assessments, etc. The ACR confirmatory R&D program is designed to support reactor development and licensing. The use of PSA in design has been extensive, including using the knowledge and experience gained from operating CANDU plants and in defining reliability and risk targets.

The ACR-1000 design complies with all applicable Canadian (CNSC) regulatory requirements [1], while taking into account all applicable international requirements, as appropriate, such IAEA's safety standard "Safety of Nuclear Power Plants: Design Requirements", NS-R-1 [2].

The major proven safety features of the CANDU are, for comparison with other Gen III reactor systems: four-quadrant separation; modular core design with replaceable and inspectable pressure tubes; negative reactivity coefficients backed by multiple, diverse shutdowns systems; inherent built-in cooling of the core by moderator, shield tank and reactor vault water; and a robust containment design with diverse cooling and heat rejection systems. These contribute not only to an ultra-low Core Damage Frequency (CDF), but also provide inherent physical barriers against accident progression.

1.2 Generic Safety Case Report (GSCR)

The Generic Safety Case Report (GSCR) [1] has been prepared to provide an integrated, site-independent overview of the ACR-1000 safety design requirements, a demonstration of the design compliance with Canadian regulatory requirements, and document the Level 1 Probabilistic Safety Assessment (PSA) and the comprehensive bounding deterministic safety

® CANFLEX is a Registered Trade-mark of AECL and the Korea Atomic Energy Research Institute analysis. This 20 chapter, 3000 page report follows the scope and content of the Preliminary Safety Case Report (PSAR), and it consolidates key design information from ACR-1000 support documents, and presents the basis of the ACR-1000 safety case based on the detailed design completed during the basic ACR engineering program.

2 OVERVIEW OF THE ACR-1000 SAFETY DESIGN

2.1 Safety Design Approach

Consistent with the overall safety concept of defence-in-depth, the design of the ACR-1000 plant aims to, as far as practicable: prevent, and reduce challenges to the integrity of physical barriers; maintain integrity of any barriers when and if challenged; and obviate failure of a barrier as a consequence of the failure of another barrier. The objectives of this approach are to provide adequate means to maintain the plant in a normal operational state, to ensure proper short-term response immediately following an initiating event, to facilitate the management of the plant in and following any DBA, and in certain defined accident conditions, beyond the DBAs. This includes the design of the core to have a negative power coefficient and a small negative coolant void reactivity under nominal design conditions, improved performance of safety systems, and provision for a robust containment design meeting Canadian and international practice for the siting and licensing of new nuclear plants.

2.2 Defense in Depth

The ACR-1000 design has evolved from a proven CANDU line of products that has always used the defence-in-depth principle as a basis for design. In addition, the ACR-1000 design includes additional inherent and engineered safety features and incorporates the five major classic physical barriers to the release of radioactive materials to the environment, as follows:

a) The fuel matrix. The bulk of the fission products generated in the fuel are contained within the fuel grains or on the grain boundaries, and are not readily available to be released even if the fuel sheath fails.
b) The fuel sheath. There are large margins to fuel sheath failure under normal operating conditions.
c) The heat transport system (HTS). Even if fission products are released from the fuel during an accident, they will be contained within the HTS, which is designed to withstand the pressure and temperature loading resulting from the accident conditions. Figure 1 presents a schematic of the HTS.
d) Containment. Designed to withstand major internal and external forces, and retain its integrity. In the event of a DBA, automatic containment isolation will occur, ensuring that any subsequent release to the atmosphere is extremely small. Figure 2 presents a schematic of the containment building.
e) The exclusion zone. To provide an additional physical mechanism to limit doses to the public; this siting requirement ensures that even if fission products were released from containment they are dispersed in the atmosphere limiting any harm to any member of the public.

As a part of the inherent safety features, the moderator system in all CANDU designs provides a key additional heat sink, providing another means of core cooling and maintaining the barriers to the release of radioactive materials to the containment. Also, even following in-core LOCAs in which fission products are released from a channel, the fission products must pass through the moderator water, where the majority are retained.

This classic concept of defence-in-depth for physical systems is also extended and applied to all management activities, whether organizational, safety, behavioural, or design-related, thus in effect providing a sixth and overarching safety management barrier. By ensuring that all safety-related activities are subject to overlapping provisions, even if a failure occurs, it is detected and compensated for or corrected by appropriate measures. Application of the concept of defence-in-depth throughout design and operation provides a graded protection against a wide variety of postulated transients, AOOs, and DBAs, including those resulting from equipment failure or human action within the plant (internal events), and events which originate outside the plant (external events). Application of the concept of defence in depth in the ACR-1000 design provides a series of levels of defence (physical barriers, quadrant separation, safety management, inherent features, equipment, and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails, and ensuring not only low probabilities of occurrence but utilizing the reactor's redundant and diverse safety features as shown below.

2.3 Independence and Separation

Physical and functional separation of systems important to safety performing the same safety function provides independence to ensure that common cause events and functional interconnections do not impair performance of the required safety functions. The common cause events that tend to drive the requirements for independence typically include fires, flooding, earthquakes, explosions, missiles, pipe whip, electrical faults, electromagnetic or radio frequency interferences, and software errors.

ACR-1000 safety and support systems are designed in conformance with the philosophy and safety objective of physical and functional separation for SSCs important to safety required by References [2] and [3]. Basically, any two systems or system divisions or redundant components carrying out the same function need to be separated and treated as being independent in safety analyses or Probabilistic Safety Assessments (PSAs).

The philosophy is applied as follow:

a) Separation of safety systems from process and control systems;

b) Separation between safety systems;

c) Separation of redundant SSCs important to safety.

The four fundamental nuclear safety functions (Control, Cool, Contain and Monitor) are generally provided by at least two totally redundant systems or subsystems, and the trip signals to actuate these systems are provided by four redundant instrumentation channels that feed redundant actuation logic circuitry. This is done to ensure high reliability in the execution of these essential safety functions. Independence must be ensured between redundant systems and between redundant parts of a system.

To address important common cause events such as a fire requires both physical and functional separation. In addition to limiting the direct damage by the fire, and the adverse environmental conditions due to heat and smoke, physical boundaries prevent fire damage causing adverse electrical effects between connected systems, because of functional separation of redundant systems or subsystems.

Different philosophical approaches can be taken in the design and layout of the plant to achieve the goal of physical and functional separation. For the ACR-1000 design, independence is provided between redundant systems and between redundant divisions and components within those systems to mitigate the consequences of common cause events. The "Four Quadrant (4Q) Separation Philosophy" consists of four separate areas or "quadrants" of systems important to safety, and the associated quadruplicated instrumentation channels and power supply divisions for safety system instrumentation. The loss of one quadrant due to a common mode event still allows continued safe operation with the remaining three quadrants intact (but still only two are required to perform the fundamental safety functions, each generally being 4x50% capacity). This approach provides an added advantage with regard to on-line maintenance and helps achieve high capacity factors. The ACR-1000 quadrant layout is provided in Figure 2.

2.4 Safety Systems

The ACR-1000 *safety* systems are those designed to shut down the reactor, remove decay heat, and limit the radioactivity release subsequent to the failure of normally operating process systems. These consist of the Shutdown System 1 (SDS1), Shutdown System 2 (SDS2), Emergency Core Cooling (ECC) system, Emergency Feed Water (EFW) system, and Containment System. Safety support systems are those that provide services needed for proper operation of the safety systems (e.g., electrical power, cooling water, and instrument air).

Shutdown System 1: SDS1 is a mechanical rod design inserting into the low pressure moderator tank (so no rod ejection is possible) and quickly terminates reactor power operation and brings the reactor into a safe shutdown condition by dropping shut-off rods into the reactor core. Reactor operation is terminated when a certain neutronic or process parameter enters an unacceptable range. The measurement of each parameter is performed by channel and the system is initiated when any two of the four SDS1 trip channels are tripped by any parameter or combination of parameters.

Shutdown System 2: SDS2 provides a second diverse and independent chemical method of quickly terminating reactor power operation by injecting a strong neutron-absorbing solution (gadolinium nitrate) into the moderator when any two of the four SDS2 trip channels are tripped by any parameter.

Emergency Core Cooling System: The ECC system is designed to supply emergency cooling water to the reactor core to cool the reactor fuel in the event of a LOCA. The design basis accidents are LOCA events where ECC is required to fill and maintain the HTS inventory, and remove decay heat from the fuel.

The ECC function is accomplished by two sub-systems:

- The emergency coolant injection (ECI) system, which immediately injects high-pressure coolant into the HTS after a LOCA.
- The Long Term Cooling (LTC) system provides long-term injection including coolant recovery after a LOCA. The LTC system is also used for LTC after reactor shutdown following other accidents, and to allow for routine maintenance activities.

Emergency Feedwater System: The emergency heat removal function is accomplished by the EFW system which is designed to provide cooling water to the secondary side of the steam generators to enable the steam generators to remove the decay heat to the ultimate heat sink on a loss of normal feedwater supply (main feedwater and start-up feedwater). The EFW system is designed to meet the CNSC licensing requirements for an emergency heat removal system.

Containment System: The basic function of the containment system is to provide a continuous, pressure-retaining envelope around the reactor core and HTS. Following an accident, the containment system minimizes release of resultant radioactive materials to the external environment to well below regulatory limits.

The containment system includes the steel-lined, prestressed concrete Reactor Building (RB) containment structure, main and auxiliary airlocks, containment cooling spray for pressure suppression, and a containment isolation system consisting of valves or dampers in the ventilation ducts and certain process lines penetrating the containment envelope. The design ensures a low leakage rate and provides a pressure-retaining boundary for all DBAs causing high pressure and/or temperature inside containment.

The containment system automatically closes all penetrations open to the RB atmosphere when an increase in containment pressure or radioactivity level is detected. Measurements of containment pressure and radioactivity are quadruplicated and the system is actuated using two-out-of-four logic.

2.5 Safety Support Systems

Safety support systems provide services needed for proper operation of the safety systems for the ACR-1000 plant.

The ACR-1000 design includes a Reserve Water System (RWS) with a reserve water tank (RWT), located at a high elevation in the RB to provide an emergency source of water by gravity feed to the steam generators (back-up EFW), containment cooling spray, calandria vessel, reactor vault, and HTS, if required.

The Electrical Power Systems (EPS) supply all electrical power needed to perform safety functions under transient and accident conditions and non-safety functions for Normal Operation (NO). The essential (safety support) portions of the systems are seismically qualified and consist of redundant divisions of standby generators, batteries, and distribution to the safety loads.

The Essential Cooling Water System (ECW) system circulates demineralized cooling water to systems important to safety. The ECW system is seismically qualified and is comprised of four separated closed loops. All four loops operate during NO.

The Essential Service Water System (ESW) disposes heat from the ECW system to the ultimate heat sink. The ESW system is seismically qualified and comprised of four separated open loops. All four loops operate during NO.

The Compressed Air System (CAS) provides service air, instrument air, and breathing air to different safety systems and power production systems in the plant.

The Chilled Water System (CWS) supplies water to air conditioning and miscellaneous equipment, and provides sufficient cooling capacity during NO. The system includes two separate systems; one shared system (between two reactor units) serving loads during normal plant power production function, and one unitized system serving loads credited for safety support function.

3 ACR-1000 COMPLIANCE WITH CANADIAN AND INTERNATIONAL SAFETY REQUIREMENTS

The ACR-1000 design has been developed to ensure its compliance with the Canadian regulations and regulatory requirements, and with the nuclear series of standards that are prepared and issued by the Canadian Standards Association (i.e., the CSA N285 to N293 standards series, including the N286 series of standards on Quality Assurance), with the National Building Code of Canada subject to the exemptions identified in CSA N293, and with the National Fire Code of Canada. The design also follows relevant sections of the ASME Codes and Standards for boiler and pressure vessel nuclear components, with the sole exception of the pressure tubes which are subject to special inspection and licensing requirements (failure of a CANDU pressure tube is acceptable and manageable event with limited acceptable safety consequences in a CANDU reactor, whereas failure of a pressure vessel in an LWR is not acceptable).

AECL focus has been to design ACR-1000 to primarily meet the Canadian regulatory requirements, and thus place the initial emphasis on ACR-1000 design licensability and construction in Canada. However, as part of international marketing of the ACR-1000 designs, reviews that were carried out by the US NRC and in the United Kingdom, give high confidence that the ACR-1000 design is robust and that it will meet regulatory requirements in foreign jurisdictions. Both the US and the UK Nuclear Installations Inspectorate (NII) experience indicate that the ACR design is consistent with the regulators' expectations of a robust design that provides adequate protection against potential accidents in a manner that meets modern international good practice.

AECL has conducted a thorough review of CNSC's Generic Action Items (GAIs) and licensing-related OPEX issues for their applicability to and resolution by the ACR-1000 design. The improvements adopted for the current ACR-1000 reference design ensure that all of the issues raised in the past by the Canadian regulator have been addressed in the design.

In addition, AECL has completed a comprehensive review of the requirements in the IAEA document NS-R-1, "Safety of Nuclear Power Plants: Design Safety Requirements", and determined compliance of the ACR-1000 design with the IAEA requirements. This provides confidence that the ACR-1000 design will be licensable in other international jurisdictions.

4 ACR-1000 DESIGN FEATURES – ENHANCED SAFETY ROBUSTNESS

4.1 Plant Design Objectives

4.1.1 Power Generation Objectives

Each unit of the two-unit ACR-1000 plant is designed to have a core thermal output of 3200 MWth, with a nominal gross electric output is 1165 MWe. The capability criteria for power cycling of the ACR 1000 design are:

- Controlling the plant automatically in either the turbine-following-reactor mode or the reactor-following-turbine mode.
- Continuously compensating for grid fluctuations of plus or minus 2.5% full power, while operating in the range 90% to 97.5% of full power.
- Reducing reactor power quickly from steady state 100% power operation to 75%, remaining at that reduced power level for any length of time, and then returning to full power.
- Periodic load following down to 60% of full power.
- On loss of line, a safe transition to continued operation at house load.

4.1.2 Reliability and Availability Objectives

The ACR-1000 plant is designed so that the plant is readily maintainable over its full 60 year operating life, including pressure tube (PT) replacement, to provide a high confidence in achieving a lifetime capacity factor of greater than 90%. Noting that world experience shows major overhauls are required in all plants during their life, the planned and unplanned incapability has been minimized allowing an operating capacity factor of 95% to be achieved on a year-to-year basis. Plant reliability is improved by ensuring the following are implemented on an integrated basis:

- High unit capability (which is a measure of a plant's ability to stay on-line and produce electricity), by minimizing time lost due to unplanned outages and reducing duration of planned outages.
- 3-year duration between planned outages, with an outage duration time of 21 days.
- Major outages lasting less than 365 days for every 30 years.
- Annual forced outage rate of less than 1.5%.

- Low loss factor (a measure of the effects of equipment and component ageing), and unplanned shutdowns or outage extensions, by implementing equipment performance and material condition monitoring programs.
- System design changes incorporated to address lessons learned from operational experience, including ease of maintenance and operation.
- Time to enter and exit Guaranteed Shutdown State (GSS) is minimized.
- An enhanced maintenance program is used to maximize component life and minimize component replacement time, thereby minimizing radiation exposure, replacement costs, and the number of operating and maintenance personnel required.
- Information technology and configuration management systems and processes are used to support operations and maintenance (O&M).
- Retain proven technology and basic CANDU design features is used, including:
  - On-power refuelling.
  - Horizontal fuel channel design.
  - Heavy-water-moderated reactor core.
  - CANDU type fuel bundle (CANFLEX®-ACR fuel).
- Traditional CANDU design features enhanced to meet or exceed target reliability and performance of the systems important to plant availability.

4.1.3 Operability and Maintainability Objectives

Improved operability and maintainability in support of high plant reliability is assured by the following:

- Operational improvements allow increased on-line testing and maintenance. For example, the ACR-1000 plant uses four-quadrant separation concept for the safety and safety support systems to improve operability and safety.
- Modernized control room interface based on extensive feedback provided by licensed control room operators in current CANDU operating plants. The annunciation system is designed for improved event diagnosis.
- Systems, controls and procedures that minimize the potential for control room operator error. This is accomplished by providing a panel and console with careful assessment of the operator interface, automating frequent operations, validating normal and abnormal operating procedures, and providing a system that presents alarms in a sorted fashion to enable event diagnosis.
- Human factors principles and criteria are included in the design of systems, facilities, equipment, and procedures.
- To facilitate commissioning, performance tests are documented and provisions built into the design for ease of execution.
- Credited operator actions and procedures related to abnormal operations are practical to perform and validated during the design phase.
- The core physics and the fuel handling system are designed to allow and minimize maintenance of fuelling machines.
- Operational feedback and experience are incorporated into the design.
- Good access and lifting devices for maintenance purposes is provided in the design based on laydown and access needs.
- Facilities, specialized tools, flasks and other equipment, and procedures are included to allow for the inspection, disassembly and/or replacement of major pieces of

® CANFLEX is a Registered Trade-mark of AECL and the Korea Atomic Energy Research Institute equipment in the plant, including but not limited to fuel channels, fuelling machines, steam generators (SGs), large pumps and heat exchangers, and turbine generator.

- Equipment standardization is applied in order to reduce parts cost, and facilitate maintenance training and execution.

4.1.4 Safety Design Criteria

The safety design objective requires excellence in safety to protect the general public, plant personnel, and the environment, as well as plant investment.

As noted above, the ACR-1000 plant is designed based on the "defence-in-depth" safety philosophy applied to the all CANDU plants with enhancements to further improve the overall safety of the plant. This includes the core being designed to have a negative power coefficient of reactivity and a small negative coolant void reactivity (CVR) under nominal design conditions, improved performance of safety systems, and provision for a robust containment design to meet Canadian and international practice for new plants, as applicable.

The design basis events are defined, determined, and analyzed according to the internationally-applied safety analysis approach in which the worst single failure in a mitigating system is postulated along with each initiating event. The required analyses for the design basis events are performed with conservative computational methods and assumptions, and meet mandated regulatory acceptance criteria.

To ensure the safety design bases are met, a series of Safety Design Guides (SDGs) are applied in the ACR-1000 design process in order to ensure that the design complies with the appropriate safety and regulatory requirements. These SDGs identify systems important to safety, and provide requirements and guidance to designers for safety classification of structures, systems and components (SSCs); seismic qualification; environmental qualification; separation of systems and components; fire protection; and containment, radiation, and tornado protection. Appropriate design guides (DGs) are also developed and used in the system design and analysis.

Probabilistic Safety Assessments (PSAs) are conducted in parallel with the design to ensure that accident analysis frequencies meet regulatory acceptance criteria. Quantitative safety goals are defined and the PSAs demonstrate that they are met. The objective safety goal is a summed core damage frequency SCDF < 10^{-6} per reactor operating year, and a large release frequency LRF < 10^{-7} per reactor operating year. These goals are met with comfortable margin.

The ACR-1000 plant design ensures safety during construction, commissioning, start-up, and O&M by assuring the following:

- Ample thermal and safety margins for safe operation.
- Radiation exposure to plant personnel and the public is well below regulatory limits.
- Feedback from construction, commissioning, start-up, O&M experience is incorporated.
- The "defence-in-depth" safety philosophy is applied.
- Physical and functional separation of safety systems.
- Simplified, more reliable systems.
- Safety systems perform their required safety functions during design basis accidents (DBAs) without credit for active mitigation by the process systems.
- Human factors engineering principles and criteria are applied in the design of systems, facilities, equipment, and procedures.

4.2 Plant Design Features

4.2.1 Plant Description

Each ACR-1000 unit consists of the nuclear steam plant (NSP) and the balance of plant (BOP). Table 1 provides a summary of the ACR-1000 design features compared to other selected CANDU designs.

The major nuclear systems, which comprise each NSP portion, are located in the RB and Reactor Auxiliary Building (RAB). These systems include, but are not limited to, the following (noting major changes from existing CANDU practice):

- The reactor assembly, consisting of 520 channels in a reduced square lattice pitch, with larger-diameter calandria tubes (CTs) than current CANDU designs, contained within a calandria vessel. Figure 3 presents the ACR-1000 reactor assembly.
- The moderator system with a reduced volume of D_2O as compared to CANDU 6 on a per MWe output.
- The HTS with light water coolant operating at higher temperatures and pressures than current CANDU designs, in a two-loop, figure-of-eight configuration with four steam generators, four HTS pumps, four reactor outlet headers, and four reactor inlet headers.
- The fuel handling system, which consists of two fuelling machines, each mounted on a fuelling machine bridge and columns, located at both faces of the reactor to allow for on-line refuelling.
- The main steam supply system, with higher pressure and temperature conditions than the current CANDU designs, for improved turbine cycle efficiency.
- Safety systems, specifically, two shutdown systems, the emergency core cooling (ECC) system, the emergency feedwater (EFW) system (defined as the emergency heat removal system), and the containment system, and associated safety support systems.

The BOP consists of the Turbine Building (TB), the steam turbine, the generator and condenser, the feedwater heating system with associated auxiliaries, and electrical equipment. The BOP also includes the plant cooling and service water systems, water treatment facilities, auxiliary steam facilities, service and breathing air systems, BOP pumphouses and optional cooling towers, main switchyard, and other systems, equipment, and components credited for the plant power production function in the ACR-1000 two-unit plant. The TB does not contain any safety or safety support systems or components..

4.2.2 Fuel Channels and Calandria Assembly

The reactor assembly consists of 520 horizontally-aligned fuel channels arranged in a square pitch. The fuel channels are mounted in a calandria vessel containing the D_2O moderator. Each fuel channel assembly consists of a PT, two end fittings, and associated hardware. The PTs contain the LEU fuel and the high-pressure light water coolant. Individual CTs surround each individual PT. The end fittings are out of core extensions of the PT, and extend out of the end shields past the feeder cabinets. The end fittings provide connections to the fuelling machine head for on-line refuelling and to the feeder pipes.

The calandria vessel has end shields located at both ends. They are filled with shielding balls and water to provide shielding. The fuel channels are located by adjustable positioning assemblies on the two end shields and are connected by individual feeder pipes to the HTS. The calandria vessel is enclosed in a concrete vault (reactor vault) filled with light water for shielding. The reactor vault is closed at the top by the reactivity mechanisms deck. Both the moderator in the calandria vessel and light water in the reactor vault provide additional heat sink capability for beyond design basis accidents (BDBAs).

4.2.3 Fuel

The CANFLEX-ACR fuel (see Figure 4) represents the next evolution in fuel design beyond what is currently used in the Pickering, Bruce, and all of the CANDU 6 reactors. The fuel design is a modified CANFLEX-type fuel bundle similar to that already demonstrated in the CANDU 6 reactor at Point Lepreau. The fuel consists of 42 elements containing uranium dioxide fuel pellets plus a central element containing burnable poison in a zirconia matrix. The uranium dioxide pellets are made with LEU. The fuel element sheaths are made from zirconium alloy. The 43 elements are assembled between end plates to form a fuel bundle. Each of the 520 channels contains 12 bundles. The fuel enrichment of the reference core is 2.4%, and the average fuel burnup is 20,000 MWd/t. A lower fuel enrichment is used for the intermediate core fuel compared to the reference core fuel, during the transition from the fresh core for initial reactor start-up to equilibrium fuelling with the reference fuel.

4.2.4 Reactor Control Units

The reactivity control units (RCUs) are comprised of the in-reactor sensor and actuation portions of reactor regulating and shutdown systems. RCUs include neutron-flux measuring devices (vertical and horizontal flux detector units, ion chamber units, and fission chamber units), reactivity control devices (zone control units and control absorber units), safety shutdown devices (shutdown units and liquid injection shutdown units), and GSS devices. RCUs are designed to be simple, rugged, highly reliable, and require little maintenance.

Flux detectors are provided in and around the core to measure neutron flux, and reactivity control devices are located in the core to control the nuclear reaction.

In-core flux detectors are used to measure the neutron flux in different zones of the core. These are supplemented by fission chamber and ion chamber assemblies mounted in housings on the calandria shell. The signals from the in-core flux detectors are used to adjust the location of the zone control unit assemblies to compensate for changes in power levels and distribution. By varying the absorber position in these assemblies, the local neutron absorption in each zone of the reactor changes, thereby controlling the local neutron flux level.

Control absorber unit elements penetrate the core vertically. These are normally parked out of the reactor core and are inserted to control the neutron flux level at times when a greater rate or amount of reactivity control is required than can be provided by the zone control units.

Slow or long-term reactivity variations are controlled by the addition of a neutron-absorbing poison to the moderator. Control is achieved by varying the concentration of this “neutron-absorbent material” (i.e., gadolinium nitrate and boron) in the moderator. For example, the liquid neutron-absorbent material is used to compensate for the excess reactivity that exists with a full core of fresh fuel at first start-up of the reactor.

Two independent safety-grade reactor shutdown systems are provided. Each shutdown system, acting alone, is designed to shut down the reactor and maintain it in a safe shutdown condition. The safety shutdown systems are independent of the reactor regulating system and are also independent of each other. The first shutdown system, shutdown system no. 1 (SDS1), consists of shut-off units (absorber element, guide assembly, and drive mechanisms), which drop neutron-absorbing elements into the core by gravity and spring assist on receipt of a shutdown signal. A long-term guaranteed shutdown state (GSS) can be achieved by the reactor regulating system either by addition of poison to the moderator, or by insertion of dedicated GSS rods into the core. The second shutdown system, Shutdown System 2 (SDS2), uses injection of a strong neutron-absorbing solution into the moderator. The automatic shutdown systems respond to both neutronic and process signals.

A dedicated system of GSS rods is provided to allow the reactor to be maintained in a GSS without the use of moderator poisons during planned and unplanned maintenance outages. During normal reactor operation, the GSS rods are withdrawn from the core. When the reactor is in the GSS state, the GSS rods are kept inserted in the core, along with the SDS1 absorber elements.

4.2.5 Heat Transport System

The HTS (see Figure 1) circulates pressurized light water coolant through the reactor fuel channels to remove heat produced by nuclear fission in the core. The fission heat is carried by the HTS coolant to the steam generators, to produce steam on the secondary side that subsequently drives the turbine generator.

The HTS is complemented by auxiliary systems, which support its operation and maintain parameters within the HTS operating ranges. HTS auxiliary systems are the pressure and inventory control (P&IC) system, HTS purification system, and HTS pump seal system.

The HTS and its auxiliary systems are similar to those in the CANDU 6 design. However, the overall design of these systems has been improved and optimized, based on operational feedback from existing CANDU plants.

The major components of the HTS are the 520 reactor fuel channels and associated feeders, four steam generators, four HTS pumps, four reactor inlet headers, and four reactor outlet headers configured in two figure-of-eight loops with interconnecting piping. Light water coolant is fed to the fuel channels from the inlet headers at each end of the reactor and is returned to the outlet headers at the opposite end of the reactor. Figure 1-1 provides a simplified illustration of the ACR-1000 nuclear systems.

The principal function of the HTS is to provide reliable cooling of the reactor fuel under all operating conditions for the life of the plant with minimal maintenance.

The HTS also provides a barrier to radioactive fission products released during NO to ensure that radiation doses to plant staff remain within acceptable limits. It is designed to retain its integrity under normal and abnormal operating conditions.

The pressure and inventory of the coolant in the HTS are controlled by the P&IC system. The long-term cooling (LTC) system is used to remove decay heat following a reactor shutdown and to cool the HTS to a temperature suitable for maintenance of the heat transport and auxiliary system components.

4.2.6 Steam Generator Design

Four identical steam generators with integral preheaters transfer heat from the HTS coolant on the steam generator primary side to raise the temperature of, and boil, feedwater on the secondary side of the steam generator. The steam generator consists of an inverted vertical U-tube bundle installed in a shell. Steam-separating equipment is housed in the upper portion of the shell. A venturi flow restrictor is installed at the outlet nozzle of each steam generator to reduce the pressure inside the RB containment in the event of a main steam line break (MSLB).

4.2.7 Heat Transport System Pump Design

The four HTS pumps are vertical, single-stage centrifugal pumps with single suction and double discharge.

When maintenance of the shaft seals or the pump internals is required, the coolant level in the HTS can be lowered to a level below the pumps. The LTC system cools and maintains the HTS after a reactor shutdown to a temperature suitable for maintenance.

A gland seal circuit supplies cooled and filtered water for lubricating and cooling the mechanical seals. A leakage recovery cavity takes the seal leakage to the light water leakage collection system.

Each pump is driven by a vertical, totally enclosed, air-to-water cooled squirrel cage induction motor. The motor has built-in inertia to prolong pump rundown on loss of power.

4.2.8 Moderator System

Neutrons produced by nuclear fission are moderated by the D_2O in the calandria. The D_2O moderator is circulated by the moderator pumps through the calandria at a relatively low temperature and low pressure, and cooled by the moderator heat exchangers. The moderator heat exchangers remove the nuclear heat generated in the moderator and the heat transferred to the moderator from the fuel channels. Helium is used as a cover gas over the D_2O in the calandria. Chemistry control of the moderator water is maintained by the moderator purification system.

The moderator system also acts as a back-up heat sink under certain postulated BDBA conditions. Moderator circulation and the calandria are shown in Figure 5.

4.2.9 Fuel Handling System

The fuel handling system is used to fuel the reactor on demand for the purpose of controlling the reactor power distribution. The fuel handling system stores and handles fuel, from the arrival of new fuel to the storage of spent fuel. The fuel handling system is divided into new fuel handling and storage, fuel changing, and spent fuel handling and storage.

Fuel changing is performed on-power and remotely, using two fuelling machines. One fuelling machine is connected to each end of the fuel channel being fuelled. A two-bundle shift scheme is used for the reference core and a four-bundle shift scheme is used for the transition from the initial core to the reference core. In the two-bundle shift operation, two new fuel bundles are inserted at the inlet end of the channel and two spent fuel bundles are removed from the outlet end of the fuel channel.

Fuel is cooled by the fuel handling system once it is removed from the fuel channel, and it remains in the fuel handling system until it is reinserted in the channel or discharged into the spent fuel bay. The normal refuelling sequence does not require reinsertion.

The spent fuel bundles remain fully submerged in water while being transferred from the fuelling machine in the RB, to the spent fuel reception bay in the RAB. The spent fuel bay has a storage capacity for more than 10 years of accumulated reactor operation plus a full core discharge. Equipment, including a series of tools and accessories, is provided for the physical handling of spent fuel in the spent fuel bay from the manbridge that operates above the bay. Special design provisions and administrative procedures protect against inadvertent criticality due to use of low enriched fuel.

The spent fuel bay cooling and purification system removes the decay heat generated by the fuel, removes the suspended activation products, and controls the water chemistry.

5 CONCLUSIONS

Based on decades of design development and R&D of different CANDU reactor designs in Canada, the ACR-1000 was developed by AECL is a 1200 MWe-class light-water-cooled, heavy-water-moderated pressure-tube reactor, which has evolved from the well-established CANDU® line of reactors. The ACR-1000 design retains the basic, proven, CANDU design features while incorporating innovations and state-of-the-art technologies to ensure fully competitive safety, operation, performance and economics. Improvements include greater operating and safety margins plus adherence and compliance with the latest safety thinking regarding external events and risk assessment.

The ACR-1000 design complies with all applicable Canadian Nuclear Safety Commission (CNSC) regulatory requirements. Although not mandatory in Canada, the ACR-1000 design takes into account all applicable international requirements as appropriate. Moreover, IAEA's safety standard "Safety of Nuclear Power Plants: Design Requirements", NS-R-1, has been used in the ACR-1000 design.

AECL has recently issued the ACR-1000 Generic Safety Case Report (GSCR) that provides a site-independent overview of the design, safety characteristics, and bounding safety analysis of the ACR-1000, which demonstrates design readiness and licensability in Canada and abroad.

6 ACKNOWLEDGEMENT

The authors of this paper acknowledge the major contributions from many AECL staff working on the ACR-1000 design and safety analysis, and on the production of the Generic Safety Case Report that was used as a key reference in this paper.

7 REFERENCES

1. "ACR-1000 Generic Safety Case Report", AECL document 10820-01320-SAR-001, June 2008.
2. "Safety of Nuclear Power Plants: Design Safety Requirements", IAEA document NS-R-1, September 2000.
3. "Design of New Nuclear Power Plants", CNSC Draft Regulatory Document RD-337, May 2008.

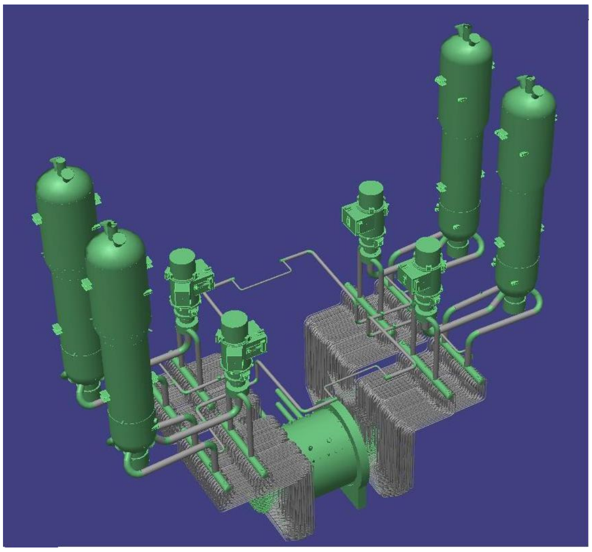

Figure 1: Schematic of the ACR-1000 Heat Transport System

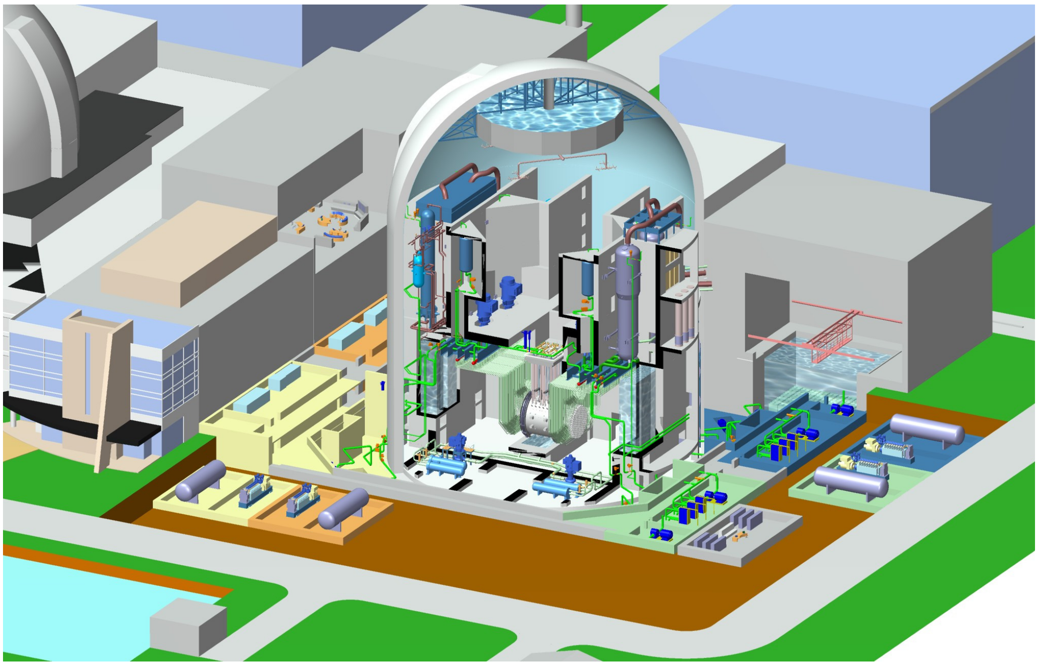

Figure 2: ACR-1000 Quadrant Layout

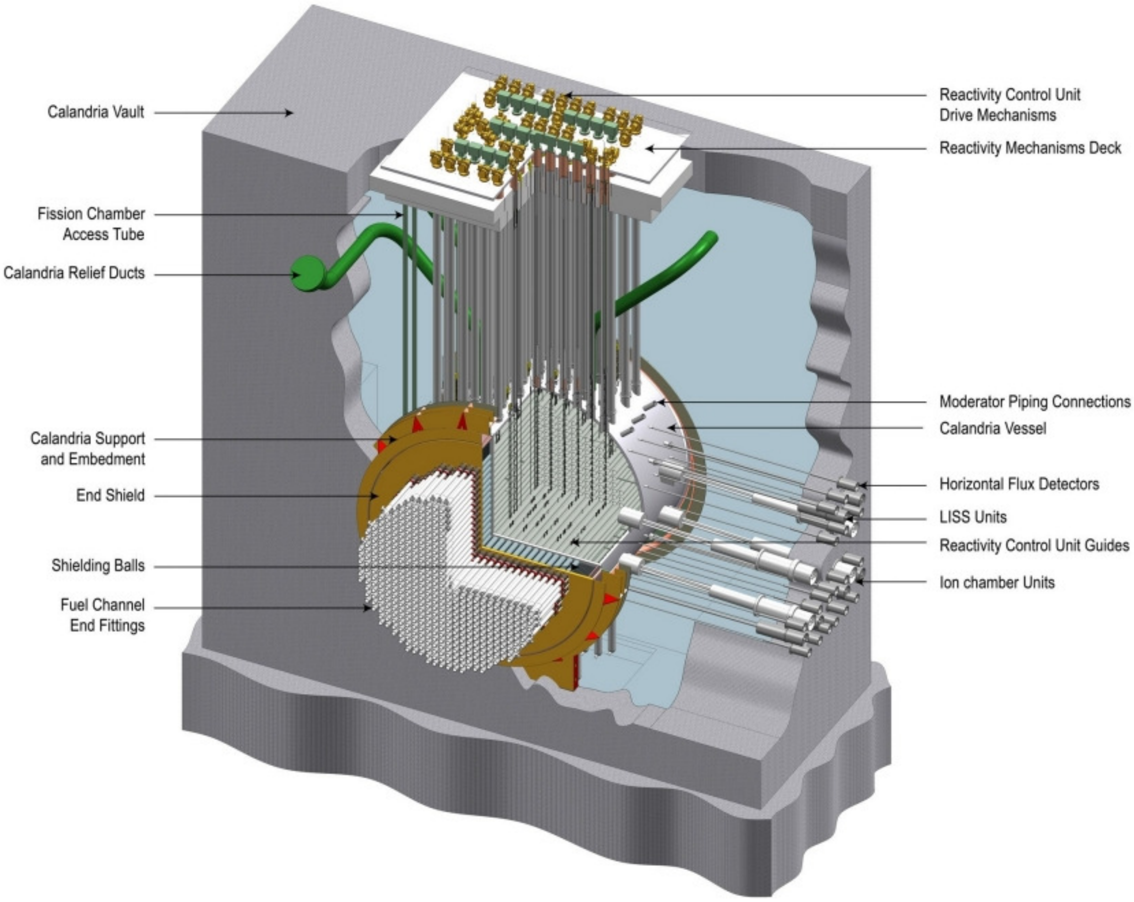

Figure 3: ACR-1000 Reactor Assembly

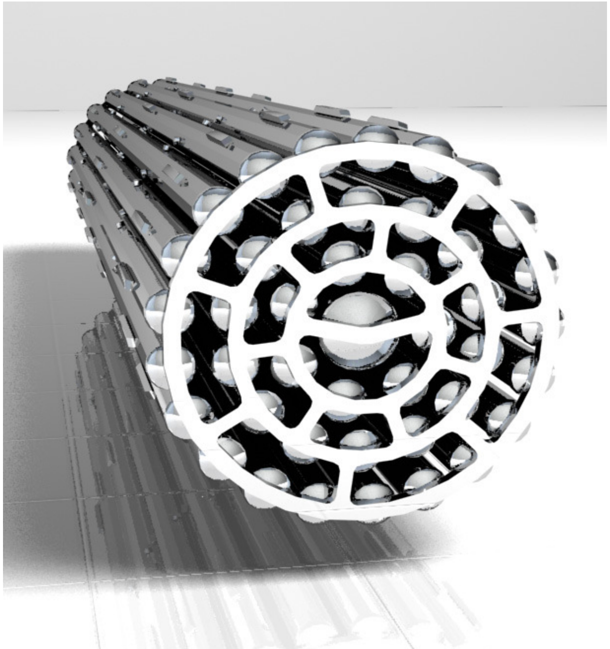

Figure 4: ACR-1000 Fuel Assembly

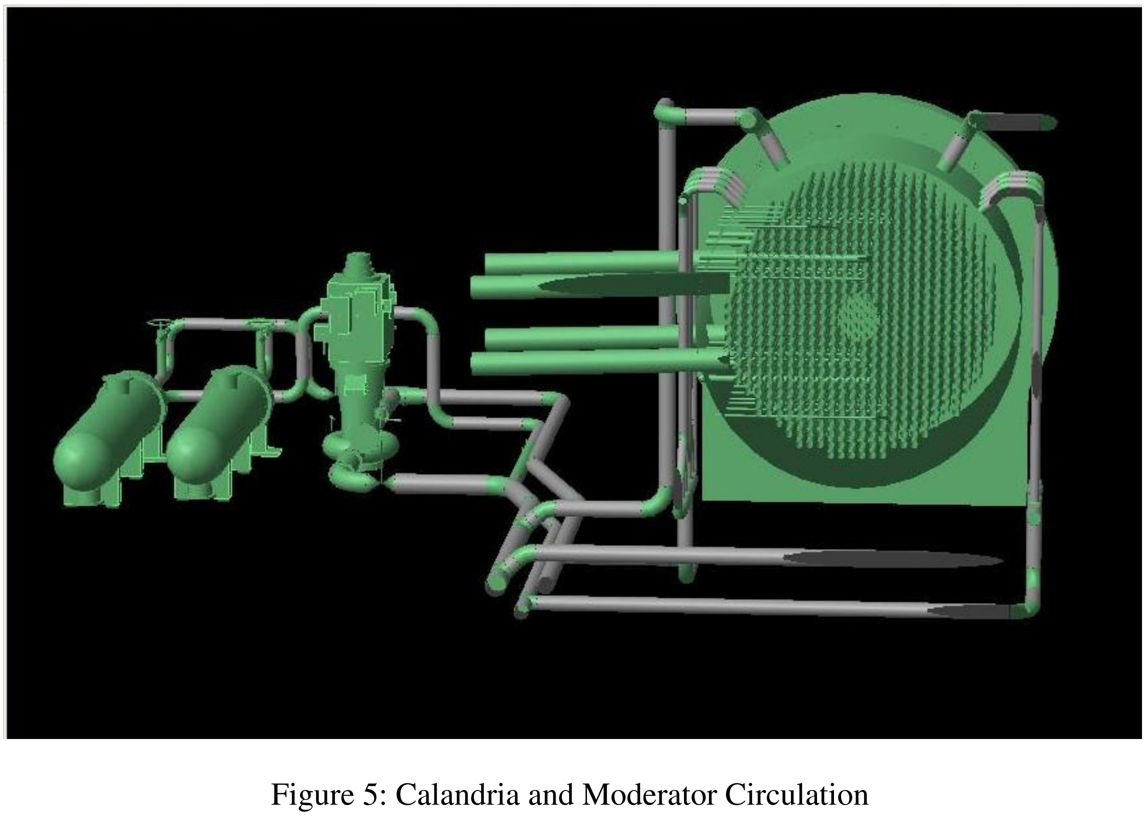

Figure 5: Calandria and Moderator Circulation

EPR SAFETY CONCEPT

Jürgen Wirkner, Jürgen Czech
AREVA NP
Freyeslebenstr. 1, 91058 Erlangen, Germany
Jürgen.Wirkner@areva.com / Jürgen.Czech@areva.com

ABSTRACT

The purpose of this presentation is to present the safety concept of the European Pressurized Water Reactor (EPR).
Special emphasis is given on safety strategy, separation of functions and core melt mitigation.

1 INTRODUCTION

For the development of future Nuclear Power Plants (NPPs) there are two fundamental objectives:

- High safety level
- Competitiveness with other power generating plants,

These two objectives may be mutually contradictory because improvement of safety features is normally linked to additional investment cost; however, higher investment can also result in higher availability, and this enhances competitiveness. It requires intensive engineering effort and intelligent technical solutions to harmonize these at first sight contrary requirements. For example, higher availability can be achieved by increasing system redundancy; this in turn allows preventive maintenance during power operation and therefore shorter refuelling outages; also increasing component quality results in fewer component failures and thus avoids shutdowns.

Adopting an evolutionary reactor design was chosen because this is the best way to take advantage of the operating experience gained from existing plants and the research and development studies conducted for them. Having chosen the “evolutionary approach”, operating experience was a significant driving force in actually improving the defence-in-depth of the EPR.

2 TECHNICAL FEATURES

The rated thermal power of the Nuclear Steam Supply System (NSSS) is 4300 MW. The high efficiency of the plant has resulted in an electrical output of about 1600 MW.

The development of the EPR is an evolutionary approach which is based on the experience gained in the construction and operation of the existing plants in France and Germany. The reactor coolant system design, loop configuration and the design of the main components are close to those of existing designs and can therefore be considered as well proven.

The system organization fulfils the principle of enhanced efficiency and reliability as well as the principle of diversification since any safety-grade system function can be backed up by another system or group of systems.

3 SAFETY APPROACH

A twofold strategy was pursued in the EPR safety principles:

- To improve the prevention of accidents and core damage;
- To mitigate severe accident consequences, even if their probability has been further reduced. This is achieved by implementing features to ensure containment integrity. Thus, it can be demonstrated that the need for emergency response measures is restricted to the immediate vicinity of the plant.

The safety approach includes a rugged deterministic basis complemented by probabilistic analyses in order to improve the prevention of accidents, and also their mitigation. Representative scenarios are defined for both core melt prevention and the prevention of large releases in order to provide a design basis for risk reduction features.

Accident and core damage prevention measures are enforced by:

- Enhanced efficiency and reliability of the safety systems
- Elimination of common mode failures by physical separation and diverse backup for safety functions
- Increased grace periods for operator actions by designing components (e.g. pressurizer and steam generators) with larger water inventories to moderate transients
- Reduced sensitivity to human errors by an optimized man-machine interface, digital instrumentation and control systems and information supplied by modern operator information systems
- Low-probability events with multiple failures and coincident occurrences up to the total loss of safety-grade systems are considered beyond the deterministic design basis
- Design provisions for severe accident management are:
  - Reactor coolant system depressurization within the containment in case of total loss of secondary side cooling;
  - Features for corium spreading and cooling, for hydrogen recombination, and for containment heat removal in case of severe accidents.

4 PLOT PLAN

The configuration of the individual power plant buildings is shown in Figure 1.

The main Nuclear Island buildings of the plant unit are:

- Reactor building
- Safeguard buildings
- Fuel building
- Nuclear auxiliary building

- Access building
- Diesel buildings
- Essential service water pump buildings
- Vent stack
- Radioactive waste building

The main turbine island buildings of the plant unit are:

- Turbine building
- Switchgear building
- Office building
- Switchyard, transformer structures
- Circulating water buildings and structures
-

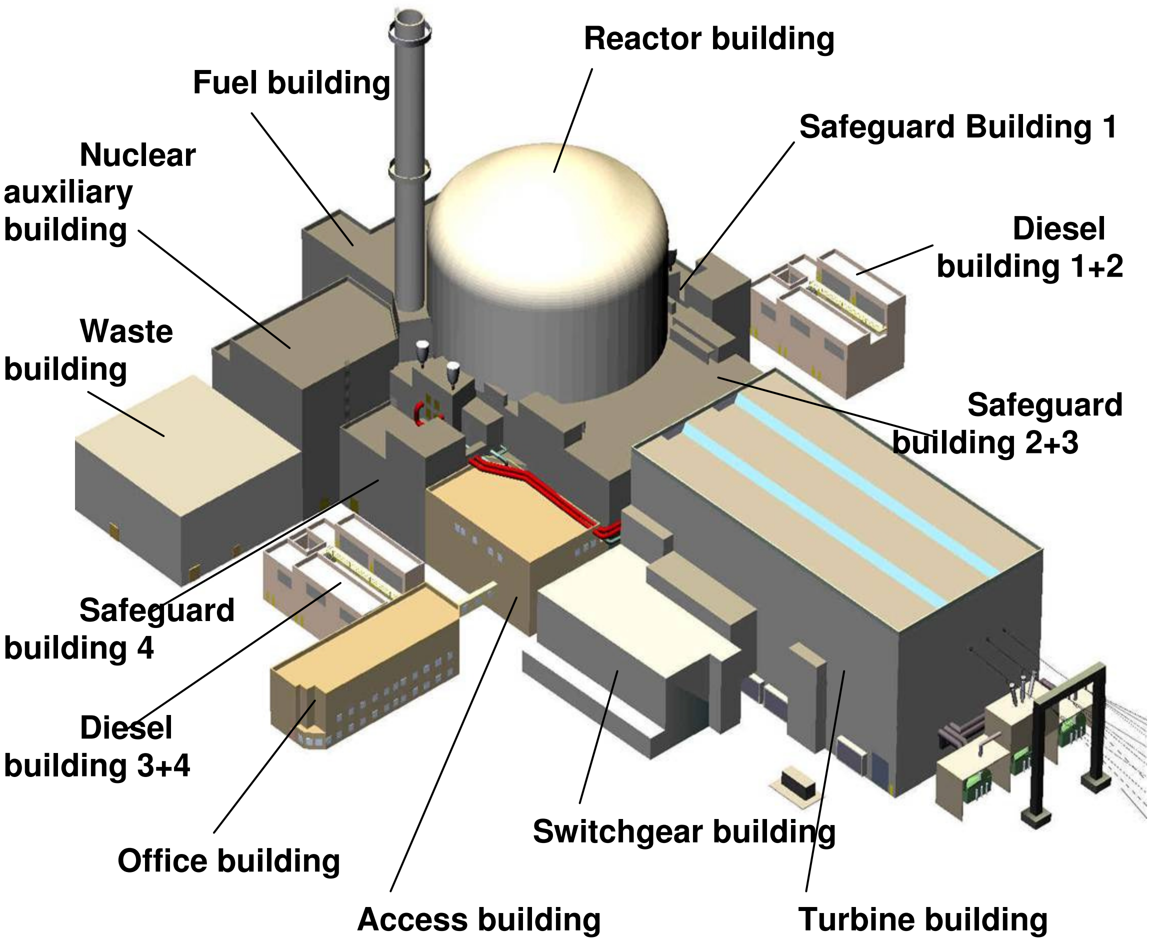

Figure 1: Plot plan

The four train redundant safety systems are located in the physically separated safeguard buildings. The following examples show the location of the emergency core cooling system.

Figure 2**:** Location of safety system in physically separated buildings

The safety-related buildings are designed to withstand natural external hazards (earthquake). Furthermore, buildings containing systems for core and spent fuel cooling are designed for manmade external hazards (explosion pressure wave, airplane crash). The features to mitigate airplane crash consist in either physical protection of the buildings (double walls for reactor building, safeguard buildings 2/3 and fuel building) or separation of redundant subsystems by distance (e.g. diesel generators).

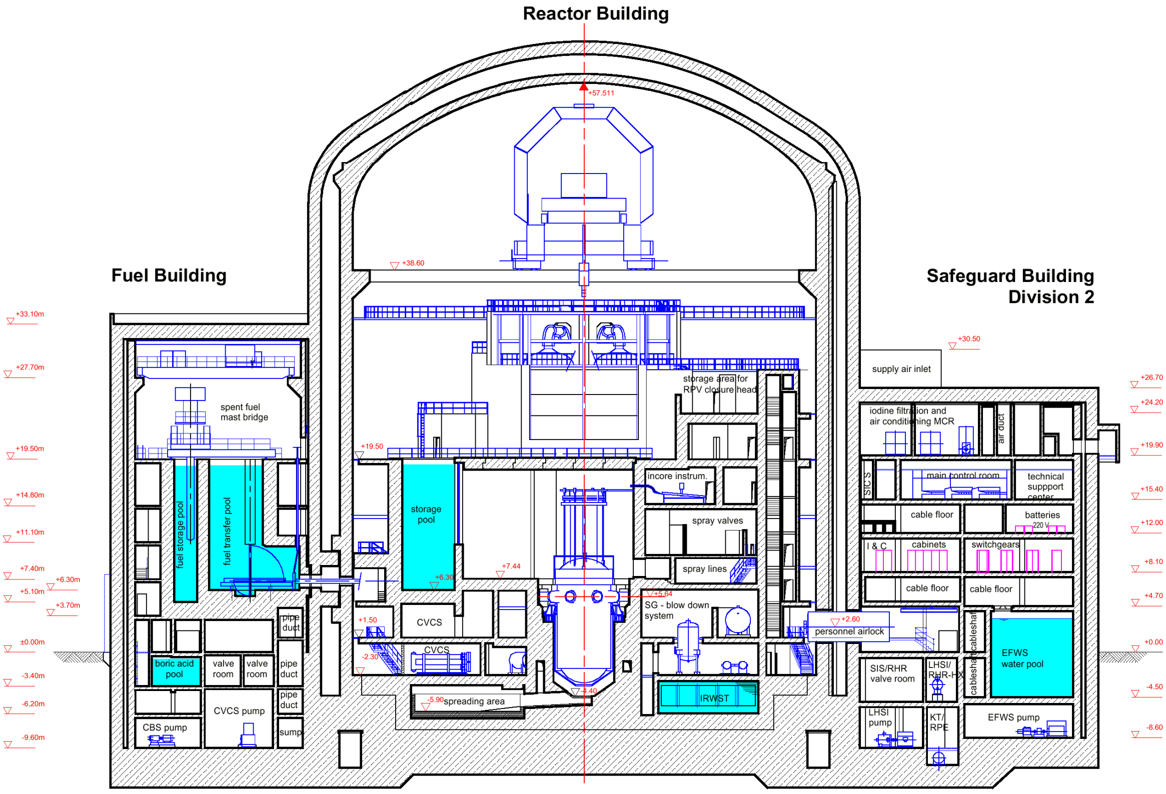

Figure 3**:** Section view reactor building, fuel building, safeguard building

5 EPR SYSTEM CONFIGURATION

5.1 Nuclear steam supply system

The Reactor Building houses the following main components of the Nuclear Steam Supply System (NSSS):

- A low-alloy steel Reactor Pressure Vessel (RPV) with stainless steel inner cladding containing the core;
- Four reactor coolant loops filled with water at a pressure of 15.5 MPa provide heat removal from the core. Each loop consists of a reactor coolant pump, a steam generator and connecting piping;
- The reactor coolant system (RCS) inclusive of its pressurizing system;
- Safety systems and auxiliary systems providing supporting functions.

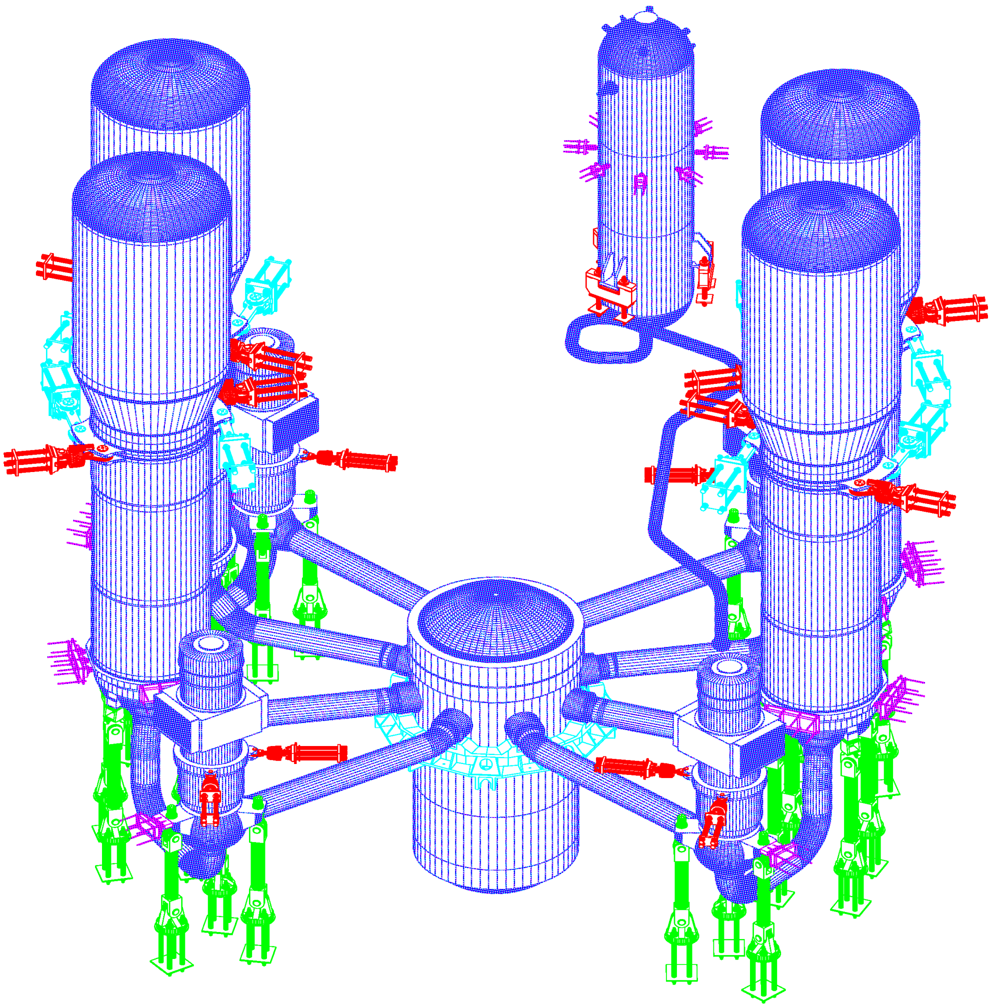

Figure 4: EPR primary circuit

Table 1: Data of pressurizer and steam generator

PRESSURIZER	
Total volume	**75 m^3**
Number of safety valves	**3**
Capacity of each safety valve	**300 t/h**
Diverse depressurization valve	**900 t/h**
STEAM GENERATORS (SG)	
Number	**4**
Heat transfer surface area per SG	**~ 7960 m^2**
Tube outer diameter	**19.05 mm**
Water mass per SG on secondary side at full load	**~ 80.9 t**
Saturation pressure in the tube bundle	**7.8 MPa**
Pressure at hot zero power	**9.0 MPa**

5.2 Main characteristics of the reactor

The fuel assembly structure supports the fuel rod bundle. It consists of bottom and top nozzles plus 24 guide thimbles and 10 spacer grids. The spacer grids are vertically distributed over the fuel assembly structure. Inside the assembly, the fuel rods are vertically arranged in a square lattice with a 17x17 array. 24 positions in the array are occupied by the guide thimbles, which are joined to the spacer grids and to the top and bottom nozzles. The bottom nozzle is equipped with a debris filter that almost completely eliminates debris-related fuel failures.

The fuel rods are composed of a stack of sintered enriched uranium dioxide pellets, with or without burnable absorber (gadolinia), contained in a hermetically sealed cladding tube made of M5™ alloy.

Table 2: Data of core and reactor coolant system

CORE	
Active height	420 cm
Number of fuel assemblies	241
Fuel rod lattice	17 x 17 -24
Type of fuel assembly	HTP X5
Average linear hear generation rate	!56.1 W/cm
Number of RCCAs	89
Core outlet temperature	≈ 330°C
REACTOR COOLANT SYSTEM	
Operating pressure	15.5 MPa
Design pressure	17.6 MPa
RPV inlet temperature	295.9 °C
RPV outlet temperature	327.2 °C
Coolant flow per loop	28330 m^3/h

5.3 EPR system configuration

The EPR system configuration provides generally a four-fold redundancy of safety systems.

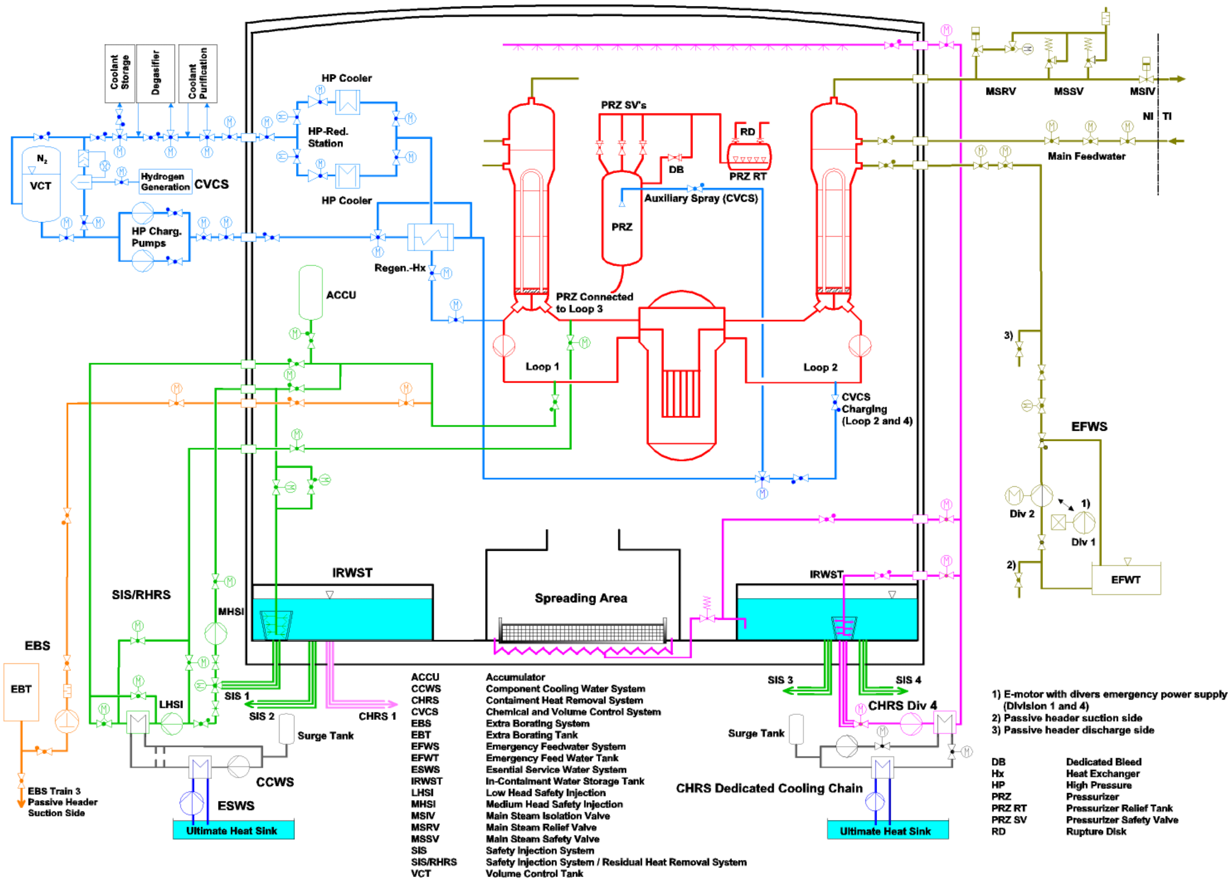

Figure 5: Survey of main fluid systems

5.4 Separation of functions

The principle of separation of functions applied in the EPR results in a system configuration in which the safety related tasks and the operational tasks are separated.

Beyond these deterministic considerations for the safety systems, backup functions for the loss of one complete redundant safety system are provided.

Table 3: Safety system backup by diverse functions

Safety-grade system	Diverse system functions				
MHSI Medium Head Safety Injection System	Fast Depressurization via Secondary Side + Pressurizer Relief Valve	+	Accumulator Injection System	+	Low Head Safety Injection System
LHSI Low Head Safety Injection System	Medium Head Safety Injection System	+	For small breaks: Secondary Side Heat Removal System		
RHR Residual Heat Removal System	RCS closed: Secondary Side Heat Removal System		RCS open: Medium Head Safety Injection System + Steaming into the Containment		
FPC Fuel Pool Cooling System	Fuel Pool Water Heat-up with subsequent Steaming	+	Coolant make-up		
EFWS Emergency Feedwater System + Steam Relief	Primary side Bleed via the pressurizer safety valves	+	Primary side Feed with MHSI		
Diesels	SBO Diesels				
TLOCC (Total Loss of Cooling Chain)	RPV closed: Secondary Side Heat Removal System		RPV open: LHSI + Steaming Note: LHSI pumps cooling by chilled water		

6 MITIGATION OF SEVERE ACCIDENTS

The EPR follows the deterministic approach with the objective of strengthening the design measures in such a way that “practical elimination” of large releases is achieved. This implies technical features to prevent early containment failure by transient events as well as features to ensure long-term integrity of the containment.

Despite their extremely low expected overall frequency, severe accidents and their consequences are taken into account in the design of EPR. In order to rule out severe consequences for the environment and for the population, the design objective is that containment integrity and leak tightness will be maintained during the entire course of potential severe accidents.

Potential processes which could jeopardize containment integrity during severe accidents and result in large fission product releases were identified in various risk studies.

They include:
- Failure of the reactor pressure vessel (RPV) at high pressure caused by the molten core with a risk of corium dispersal and direct containment heating (DCH);
- Energetic interaction of molten fuel and coolant: in-vessel and ex-vessel (MFCI);
- Interaction of molten corium with the basemat with the possibility of basemat melt-through;
- Fast hydrogen deflagration or detonation;
- Long-term pressure and temperature increase in the containment.

6.1 General design criteria

Certain general design criteria are used to design features to cope with the different phenomena to be considered. They are as follows:
- Use of passive components and means in the early phase of the accident appropriate to the plant state in case of severe accident conditions;
- Use of simple and robust designs;
- Use as far as possible of proven materials and technologies

The design criteria used for the specific design features for mitigation of individual phenomena are derived from the analysis of representative scenarios which are specified for the various phenomena and which are selected in a deterministic way.

6.2 Prevention of high pressure core melt sequences

High pressure failure of the RPV is eliminated by deliberate depressurization of the primary system to a pressure below 2 MPa using highly reliable dedicated valves in series which supplement the three pressurizer safety relief valves. Valve opening is manually actuated.

Depressurization also prevents early containment failure due to direct containment heating, and containment bypass due to creep failure of the steam generator tubes. Different scenarios have been analyzed in order to evaluate the pressure and temperature history within the reactor coolant system and the potential time frame and criterion for activation of the depressurization valves. The scenarios cover:
- Total loss of AC power with inoperability for 12 hours of all diesels;
- Total loss of feedwater with inoperability of RCS feed and bleed;
- Small-Break Loss-of-Coolant Accident (SBLOCA) with inoperability of the safety injection system.

Both power and shutdown states are considered. The analyses show that depressurization at a mass flow of 900 t/h before 650°C core outlet temperature is reached enables water injection by the accumulators to delay core melting and the reactor coolant system pressure to be reduced to well below 2 MPa by the time the RPV fails, in effect to around 0.5 MPa for many core melt scenarios. Sufficient time for activation is available and more than 1 h margin exists for latest activation to prevent vessel failure above 2 MPa and to prevent any risk of dispersal of corium debris which could induce direct containment heating.

Vessel support and cavity structures are designed to accommodate the loads resulting from a vessel failure at 2 MPa.

6.3 Hydrogen mitigation

The following design features are relevant for hydrogen control:

- The large dry containment; the average hydrogen concentration will be below 10 vol-% in all events;
- Around 50 recombiners distributed mainly over the equipment rooms; these remove most of the hydrogen before RPV failure occurs and significantly enhance mixing of the atmosphere;
- Direct discharge of the reactor coolant system inventory via a relief tank into the lower equipment compartments; this provides a large amount of steam at the time of hydrogen release and improves mixing;
- Convection and rupture foils and mixing dampers that provide passive and failsafe transformation of the two compartment containment into a single compartment following temperature and/or differential pressure increase.

The justification of the hydrogen mitigation concept is based on representative scenarios (mainly SBLOCA scenarios with breaks at different locations) and bounding scenarios with additional aggravation (e.g. with reflood of the hot core at the most penalizing moment) selected to explore the limits of the concept and proceeds as follows:
- Calculation of mass and energy input into the containment, resulting in 500 to 800 kg hydrogen for representative scenarios and up to 1000 kg with a peak rate of 6 kg/s for bounding scenarios;
- Calculation of the gas and temperature distribution for the relevant phase (until mixing) with CFD (Computational Fluid Dynamics) method;
- Assessment of the risk of fast deflagration or DDT (Deflagration to Detonation Transition with experimentally based criteria and the risk from slow deflagration via AICC (Adiabatic Isochoric Complete Combustion) pressure;
- Calculation of the combustion process with a CFD code in case flame acceleration and fast combustion cannot be ruled out under these criteria;
- Assessment of thermal loadings from recombination, short deflagration and long-lasting combustion (“standing flames”).

The most important results are as follows:
- AICC pressure is always below design pressure (0.53 MPa) for representative scenarios and below the ultimate containment pressure, where still leaks are avoided 0.96 MPa for bounding scenarios;
- Flame acceleration occurs locally, mainly in the steam generator compartments, but the flame decelerates as it progresses in the 3 dimensional space within the dome; hence, no significant dynamic loads occur on the shell, and the slow pressure increase is enveloped by the AICC pressure;
- The recombination rate is largely independent of the arrangement of the recombiners.

Temperature loadings due to recombination on the internal walls are benign.

6.4 Core melt mitigation

Features are provided to retain the melt within the containment to prevent penetration of the basemat by corium concrete interaction and to prevent a significant release of fission products, including groundwater contamination, as a consequence of loss of containment integrity at the basemat.

The EPR melt retention concept is based on ex-vessel melt retention. The risk of ex-vessel MFCI during failure is prevented by the provision of a dry reactor cavity and a dry spreading compartment.

The basic concept of EPR for melt stabilization is spreading of the melt into a large lateral compartment; this is followed by flooding, quenching and cooling from the top and bottom with water drained passively from the Incontainment Refueling Water Storage Tank (IRWST). It is a characteristic feature of this concept that the corium is not directly discharged into the spreading compartment as it is released from the RPV but first temporarily retained in the reactor cavity. This feature results in spatial separation of the functions:

1. To withstand the thermal-mechanical loads during RPV failure with only the rugged concrete structures of the reactor cavity being affected; and
2. To transfer the melt to a coolable configuration and stabilize it in an area in which only the structure of the core catcher is affected.

This separation results in a clear definition of loadings on the involved structures (Figure 6) and in defined conditions for spreading and stabilization of the corium.

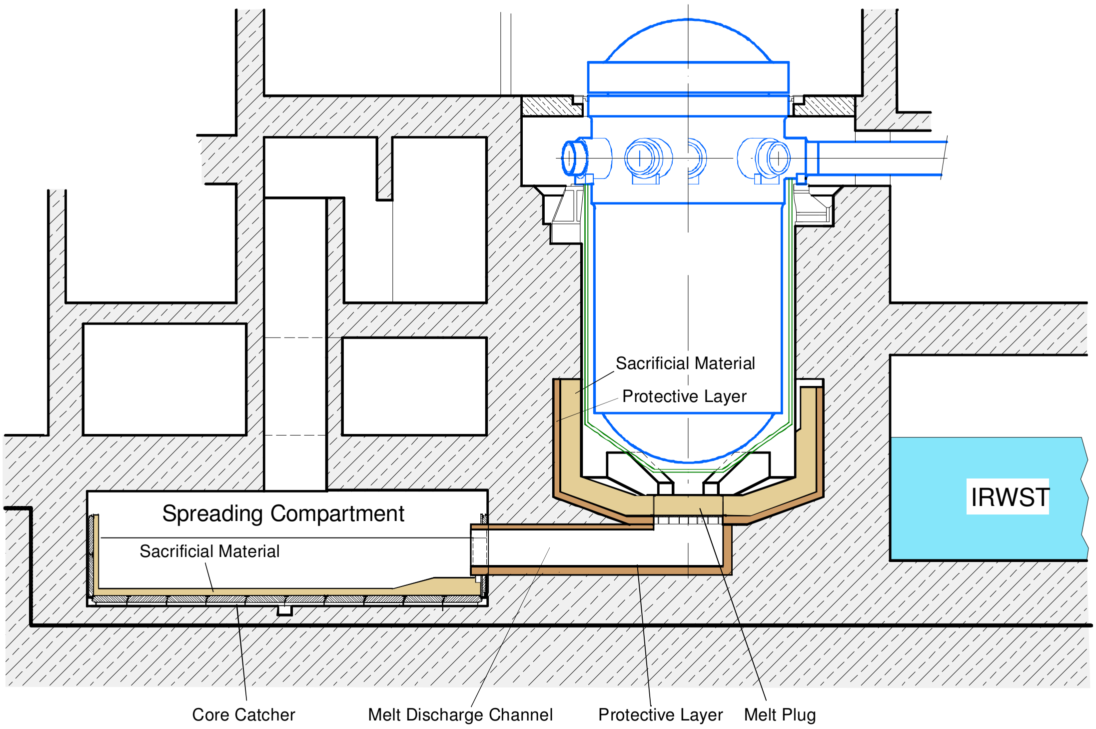

Figure 6**:** Reactor cavity section

The connection between reactor cavity and the spreading compartment is normally closed by a plug. This plug will be molten-through by the corium.

The wall of the reactor cavity is covered with a layer of sacrificial concrete which will be eroded by the corium. During the erosion time until the failure of the plug, the corium which has not been discharged during the first pour after RPV failure can accumulate within the reactor cavity. The accumulated melt will then relocate into the adjacent compartment. Spreading will occur under dry conditions.

The bottom and side structures of the core catcher are covered with sacrificial concrete to shield the steel structure from transient thermal loadings during spreading. Erosion of this layer also lowers the temperature of the melt and the density of the oxidic phase. This in turn creates a layer inversion with oxidic melt located above the metallic melt; this enables significant fragmentation of the oxides on contact with the cooling water from above. The lower, metallic melt will cool down below its solidification temperature, and the corresponding formation of crusts will dampen the initial thermal impact on the cooling structure.

Underneath the sacrificial layer a cooling structure is provided consisting of an array of massive steel blocks which at the bottom form parallel channels of rectangular cross section for cooling. Water from the IRWST for melt cooling will pass through the cooling channels and then flood the melt from above. Thus, the melt will be cooled from above and below. The generated steam will escape into the containment via a specific steam exhaust channel. Heat removal from the corium to the containment atmosphere and the structures can be performed completely and continuously in a passive mode.

However, this passive mode will apply only during the short term of the accident. In the long term, the containment heat removal system can perform active cooling by direct water injection into the cooling structure instead of spray injection into the containment atmosphere. Additional advantages are that the spreading compartment and the reactor cavity can be fully flooded, and the overflowing water which is still subcooled flows back into the IRWST.

6.5 Containment heat removal and leak tightness

Long-term pressurization of the containment is averted by a dedicated active two-train containment heat removal system which does not need to operate earlier than 12 h after accident initiation. The building structures provide sufficient heat capacity for the design pressure not to be exceeded within the uncooled period. The external recirculation cooling loops are located in special ventilated and shielded compartments with provisions for decontamination and repair of the involved components.

The system has two modes of operation:

1. Spray within the containment dome for fast condensation of the steam and pressure reduction;
2. Water recirculation through the cooling structure of the core catcher and the IRWST for long-term prevention of steaming into the containment volume, thus maintaining ambient pressure conditions in the containment and zero leakage from the containment.

Figure 7 shows a schematic diagram of the containment heat removal system indicating the different modes of operation.

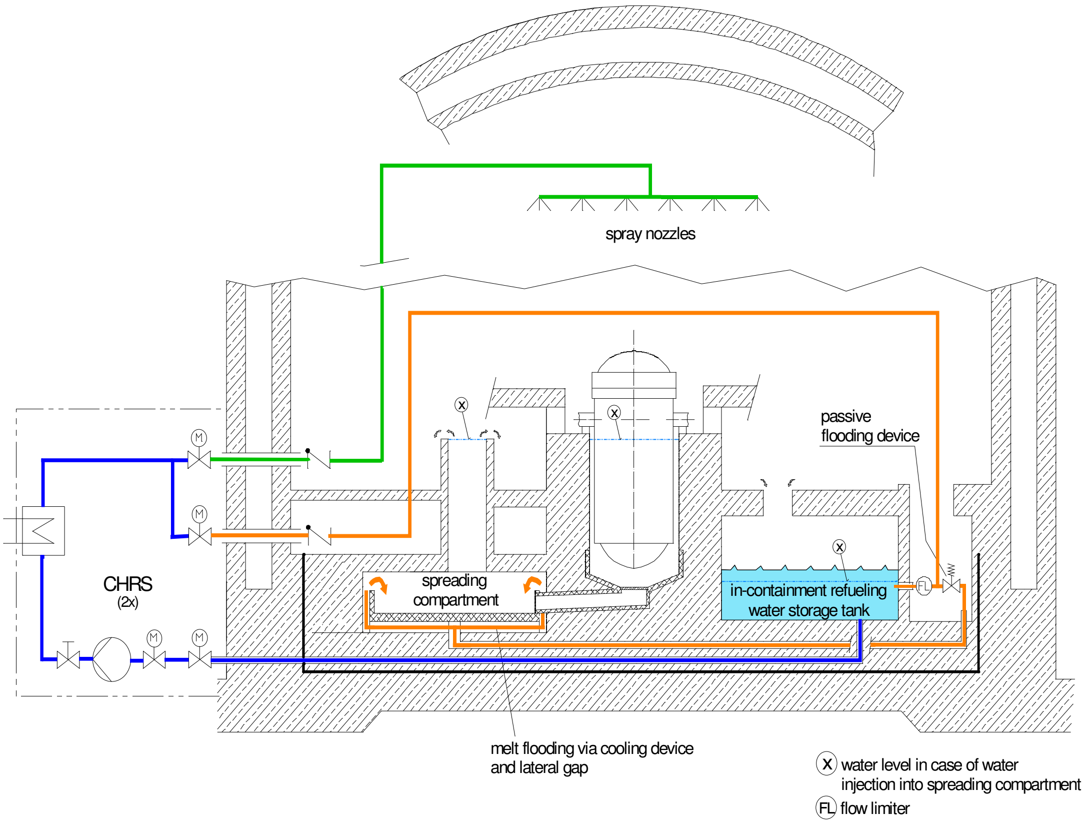

Figure 7: Scheme of containment heat removal system

Containment heat removal system and hydrogen recombination can reduce the containment pressure to near atmospheric pressure. As a consequence of Finnish regulations containment filtered venting is provided for the EPR in Finland to permit release of non-condensable gases in the long term, if necessary. Venting is performed to finally depressurize the containment and terminate any releases. With the containment and severe accident management systems functioning as designed there is no need for use of the containment filtered venting system. The design pressure of the venting system is 1.1 MPa and the design temperature 200°C.

The double-wall containment is designed to prevent a direct path forming from the inner containment to the environment. Design provisions ensure the limitation of radiological releases:

- Collection of leaks from the mitigation system in the peripheral buildings and filtration;
- Collection of leaks from the penetrations in the annulus before filtration;
- Safe and tight isolation of systems which could form a containment bypass.

7 CONCLUSION

The EPR fulfils the fundamental objective as defined in INSAG-3:

- "To protect individuals, society and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazards".

This objective lead to design the EPR according to deterministic criteria and verification by PSA.

In addition, probabilistic objectives and targets, associated with levels of radiological consequences, are as follows:

- the integral core melt frequency, considering all plant states and all types of events shall be less than 10-5/r.y.,
- the risk of large releases shall be "practically eliminated", large releases being defined as releases beyond the acceptable limits in terms of consequences: no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, and no long term restrictions in consumption of food.

The design of the EPR is such that the total core damage frequency 1.3 E-6/a (External hazards, internal hazards) and the total frequency to exceed 100 TBq CS is 7 E-8/a.

The external source terms are limited in a way that stringent countermeasures, such as relocation or evacuation of the population are restricted to the immediate vicinity of the plant and the restrictions of the use of food-stuff are limited to the first year harvest.

TOPSAFE
Dubrovnik, Croatia, 30.09 - 3.10.2008

European Utility Requirements (EUR) Volume 3 Assessment for AP1000 and EPP Phase 2E Program

Monica Frogheri, Gianfranco Saiu
Ansaldo Nucleare
Corso Perrone, 25 16152 Genova, Italy
monica.frogheri@ann.ansaldo.it, gianfranco.saiu@ann.ansaldo.it

ABSTRACT

The European Passive Pressurized Water Reactor (EPP) Program was initiated in 1994 by several European utilities, Westinghouse and its Industrial Partner ANSALDO Nucleare. The objective of the EPP Program is to develop a PWR Nuclear Island design based on the Westinghouse passive plant technology and ensuring compatibility of the plant design with the European Utilities Requirements (EUR), as well as key European licensing requirements. Since 2001, the EPP reference plant design is the AP1000.

The AP1000 is two-loop 1100 MWe pressurizer water reactor (PWR). It uses passive safety systems to provide significant and measurable improvements in plant simplification, safety, reliability, investment protection and plant costs. The Westinghouse AP1000 development program is aimed at making available a nuclear power plant that is economical in the world-wide deregulated electrical power industry in the near term.

In 2004, the EUR organization launched a program for the preparation of an EUR Volume 3 Subset for the AP1000 NPP. The assessment of the AP1000 plant versus the EUR requirements was a key activity in the frame of EPP Phase 2D program and it is an important step for the evaluation of the AP1000 design for application in Europe. EUR results also provided input to the EPP design group that has selected the most significant deviation for performing detailed studies to quantify the degree of compliance and, if needed.

Ansaldo Nucleare is currently involved in the EPP Phase 2E program, started at beginning of 2007. The EPP Phase 2E program is intended to bring the AP1000 design into optimum compliance with the EUR. As a result, it is expected that the AP1000 plant design will either be shown to be adequate to meet the EUR or design changes will be identified to fully meet the EUR Requirements. The proposal also addresses initial licensing steps in the EPP member countries.

The purpose of this paper is twofold:

1. to provide an overview of the EUR program, with particular reference to the EUR Volume 3 Subset for the AP1000 NPP.
2. to provide an overview of the results obtained during the EPP Phase 2E

1 INTRODUCTION

The purpose and main objective of the EUR is to produce a common set of utility requirements, endorsed by major European utilities for the next generation of LWR and BWR nuclear power plants. The aim of the requirements is to promote the harmonization of Safety, approaches, Targets, Criteria and assessment methods, Standardization of design conditions, Design objectives and criteria for the main systems and equipment, Equipment specifications and standards, Information required for assessment of safety, reliability and cost, thus allowing the development of standard designs that can be built and licensed in several European countries with only minor variations. The benefits of a common set of requirements are:

1. Improvement in the licensing of new nuclear power plants and in their public acceptance:
   - by setting common safety Targets which are consistent with the best European and international objectives,
   - by promoting within Europe common technical responses to safety problems,
   - by setting "good neighborhood" requirements like low targets for accidents and routine radioactive releases into the environment, and consideration of decommissioning aspects at the design stage.
2. Strengthening of nuclear electricity competitiveness:
   - by controlling construction costs and operating costs through standardisation, simplification and optimisation of maintenance at the design stage,
   - by establishing stable conditions for competition between the suppliers on the European Market,
   - by allowing low operation and fuel cycle costs, through flexible and efficient design features that allow easy adaptation to future plant operating and fuel management schemes,
   - by laying down ambitious (but achievable) availability and lifetime Targets.

The EUR document [1] is divided into four volumes. Each volume is divided into chapters that deal with a specific topic. Volumes 1, 2 and 4 provide the Main Policies and Top Tier Requirements, the Generic Nuclear Island Requirements and the Power Generation Plant Requirements for the generic European LWR and BWR nuclear power plants (NPP).

Volume 3 is intended to report the Plant Description, the Compliance Assessment to EUR Volumes 1 and 2, and finally, the Specific Requirements for each specific Nuclear Power Plant Design considered by the EUR.

As the result of the continuous updating program and making use of the experience and feedbacks from the development of Volume 3 detailed assessment programs conducted in the previous years, in April 2001, the EUR organization issued Revision C of the European Utility Requirements.

In 2004-2006 the EUR organization with the help of the European Passive Plant Program (EPP) Team performed a detailed compliance review of the AP1000 PWR design against the EUR Revision C Volume 1 and 2 requirements.

EUR results also provided input to the EPP design group that has selected the most significant deviation for performing detailed studies to quantify the degree of compliance and, if needed, to define possible design changes to fully meet the EUR Requirements. The EUR Volume 3 Subset for the AP1000 NPP has been issued in December 2007 [2].

2 EPP PROGRAM OVERVIEW

The European Passive Pressurized Water Reactor (EPP) Program was initiated in 1994 by several European utilities, Westinghouse and its Industrial Partner ANSALDO Nucleare. The objective of the EPP Program is to develop a PWR Nuclear Island design based on the Westinghouse passive plant (AP600, AP1000, EP1000 and SPWR) technology by ensuring compatibility of the plant design with the EUR as well as key European licensing requirements.

Through the middle of 1999, the EPP Program studied a 1000 MWe three-loop passive plant design, called EP1000. In 1999, in parallel with the EPP Phase 2B activities, Westinghouse, in response to U.S. market conditions, which indicated that new nuclear units must be competitive with natural gas fired generation alternatives, initiated a study to evaluate the feasibility of uprating the AP600 (2-loop passive plant design) to achieve better plant economics. Although the AP600 was the most cost-effective plant ready for deployment, it was more expensive than that needed to compete in the U.S. market. In order to develop a cost competitive nuclear power plant, Westinghouse undertook a program to develop and license a larger version of the AP600 with an increased power output of greater than 1000 MWe (3400 MWt), while maintaining the AP600 design configuration, use of proven components and licensing basis. The plant is called AP1000.

Before the end of EPP Phase 2B, the EPP utilities decided, to merge the EPP program with the U.S. AP1000 program. The new EPP reference plant design therefore became the two-loop AP1000 with minor design modifications for European application. The plant follows very closely the AP1000 U.S. design, that has implemented some of the design features developed during the EPP program, including Low Boron Capabilities, Auxiliary Systems Design, capability to operate with MOX fuel.

Merging the EPP and AP1000 programs provides a more cost effective way to achieve the final objective of the EPP Program which is to develop a 1000 MWe PWR design based on passive technology that meets the EUR and is licensable in Europe.

The AP1000 was granted Final Design Approval in September 2004 and Design Certification from the US NRC on December 30, 2005. Comparison studies of the AP1000 versus the EP1000 have shown that the AP1000 offers distinct advantages, particularly in generating cost and in level of design maturity, while retaining good compliance with EUR and European licensing requirements. The EPP Phase 2C and 2D activities supported refinement of the AP1000 design, with specific studies performed to identify and establish the design modifications needed for Europe.

Westinghouse and Ansaldo are currently involved in the Phase 2E program, started at beginning of 2007. The EPP Phase 2E program is intended to bring the AP1000 design into optimum compliance with the EUR. As a result, it is expected that the AP1000 plant design will either be shown to be adequate to meet the EUR or design changes will be identified to bring the AP1000 into compliance. The proposal also addresses initial licensing steps in the EPP member countries.

The objectives of Phase 2E of the European Passive Plant Program are:

- Continue to support progress in refining the AP1000 base design details for Europe.
- Address EUR non-conformances.
- Address initial licensing steps in the EPP member countries.
- Support the EUR organization to finalize EUR Volume 3 for AP1000.

The EPP program provides the participants with the opportunity to gain passive plant technology knowledge, participate in the final configuration of the European AP1000 design and define and resolve any European licensing issues.

2.1 AP1000 PLANT OVERVIEW

The AP1000 is a two-loops PWR (Figure 1), with a gross electrical power of 1117 MWe [3]. The AP1000 design includes advanced passive safety features and extensive plant simplifications to enhance the safety, construction, operation, and maintenance of the plant. The plant design utilizes proven technology, which builds on over 35 years of operating PWR experience.

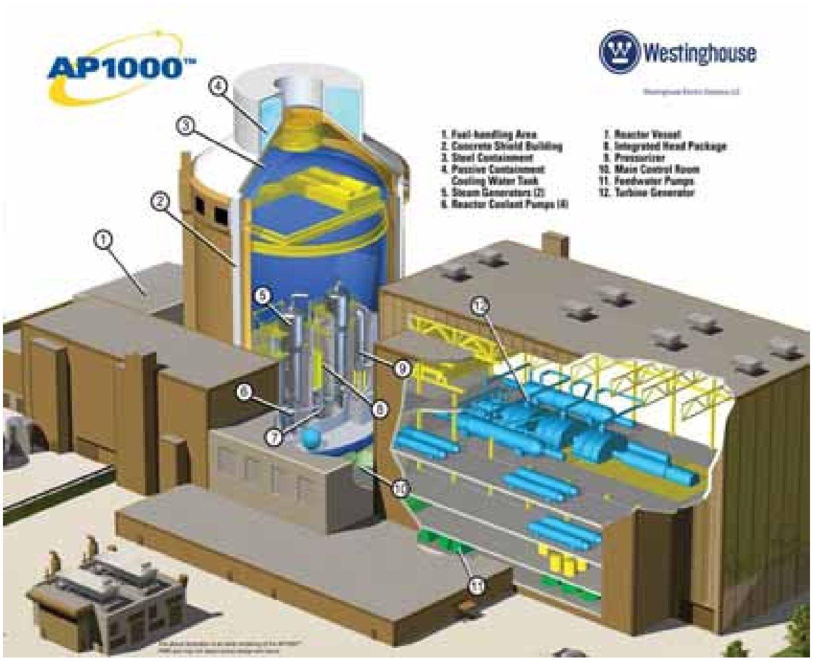

Figure 1: AP1000 Plant

On December 30, 2005 the United States Nuclear Regulatory Commission (NRC) approved Design Certification for the AP1000 standard nuclear plant design, making the AP1000 the first Generation III+ plant to receive such certification.

On 24th of July 2007 Westinghouse Electric Co. signed landmark contracts with China's State Nuclear Power Technology Corporation, to provide four AP1000 nuclear power plants in China. The four plants are to be constructed in pairs at the Sanmen (Zhejiang) and Haiyang (Shandong) sites. First concrete is expected to begin March 2009, with the first plant becoming operational in late 2013. The remaining three plants are expected to come on line in 2014 and 2015. In this framework, Ansaldo Nucleare in Joint Venture with Mangiarotti Nuclear, has signed a contract with Westinghouse for the design and the supply of innovative components to be installed in the first AP1000 unit at the Sanmen site.

The AP1000 uses passive safety systems to improve the safety of the plant and to satisfy safety criteria of regulatory authorities. The use of passive safety systems provides superiority over conventional plant designs through significant and measurable improvements in plant simplification, safety, reliability, and investment protection.

The passive safety systems require no operator actions to mitigate design basis accidents. These systems use only natural forces such as gravity, natural circulation, and compressed gas. Safety systems do not use active components (such as pumps, fans or diesel generators) and are designed to function without safety-grade support systems (such as electric power, component cooling water, service water, HVAC). The AP1000 design includes features such as simplified system design to improve operability while reducing the number of components and associated maintenance requirements. The AP1000 has 50%

fewer valves, 83% less piping, 87% less control cable, 35% fewer pumps and 50% less seismic building volume than a conventional plant of similar installed capacity. These reductions in equipment and bulk quantities lead to major savings in plant costs and construction schedules.

The AP1000 passive safety-related systems include, among the others, the Passive Core Cooling System (PXS) and the Passive Containment Cooling System (PCS).

The PXS (Figure 2) protects the plant against reactor coolant system (RCS) leaks and ruptures of various sizes and locations. The PXS provides the safety functions of core residual heat removal, safety injection, and depressurization.

The PXS uses three passive sources of water to maintain core cooling through safety injection. These injection sources include the core makeup tanks (CMTs), the accumulators, and the IRWST. These injection sources are directly connected to two nozzles on the reactor vessel so that no injection flow can be spilled for the main reactor coolant pipe break cases.

Long-term injection water is provided by gravity from the IRWST, which is located in the containment just above the RCS loops. Normally, the IRWST is isolated from the RCS by squib valves. The tank is designed for atmospheric pressure, and therefore, the RCS must be depressurized before injection can occur.

The depressurization of the RCS is automatically controlled to reduce pressure to about 12 psig (0.18 MPa) which allows IRWST injection. The PXS provides for depressurization using the four stages of the ADS to permit a relatively slow, controlled RCS pressure reduction.

The PXS includes a 100% capacity passive residual heat removal heat exchanger (PRHR HX). The PRHR HX is connected through inlet and outlet lines to RCS loop 1. The PRHR HX protects the plant against transients that upset the normal steam generator feedwater and steam systems. The PRHR HX satisfies the safety criteria for loss of feedwater, feedwater line breaks, and steam line breaks.

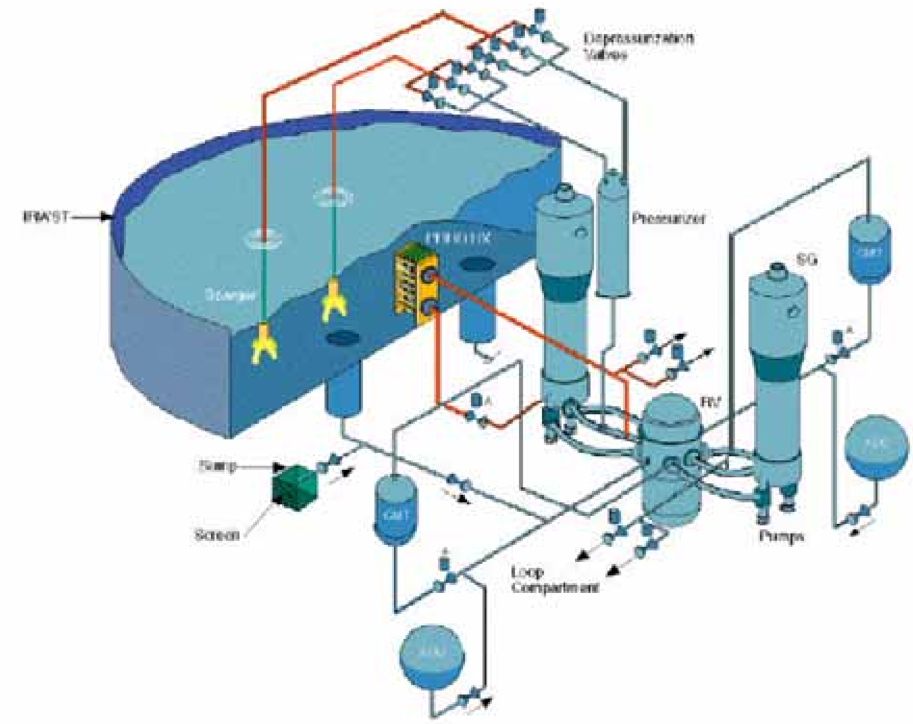

Figure 2 - AP1000 Passive core cooling system

The IRWST provides the heat sink for the PRHR HX. The IRWST water volume is sufficient to absorb decay heat for more than 1 hour before the water begins to boil. Once boiling starts, steam passes to the containment. This steam condenses on the steel containment vessel and, after collection, drains by gravity back into the IRWST. The PRHR HX and the passive containment cooling system provide indefinite decay heat removal capability with no operator action required.

The PCS (Figure 3) provides the safety-related ultimate heat sink for the plant. As demonstrated by computer analyses and extensive test programs, the PCS effectively cools the containment following an accident such that the pressure is rapidly reduced and the design pressure is not exceeded.

The steel containment vessel provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Heat is removed from the containment vessel by continuous natural circulation flow of air. During an accident, the air cooling is supplemented by evaporation of water. The water drains by gravity from a tank located on top of the containment shield building. Calculations have shown the AP1000 to have a significantly reduced large release frequency following a severe accident core damage scenario.

With only the normal PCS air cooling, the containment stays well below the predicted failure pressure for at least 24 hours. Other factors include improved containment isolation and reduced potential for LOCAs outside of containment. This improved containment performance supports the technical basis for simplification of offsite emergency planning.

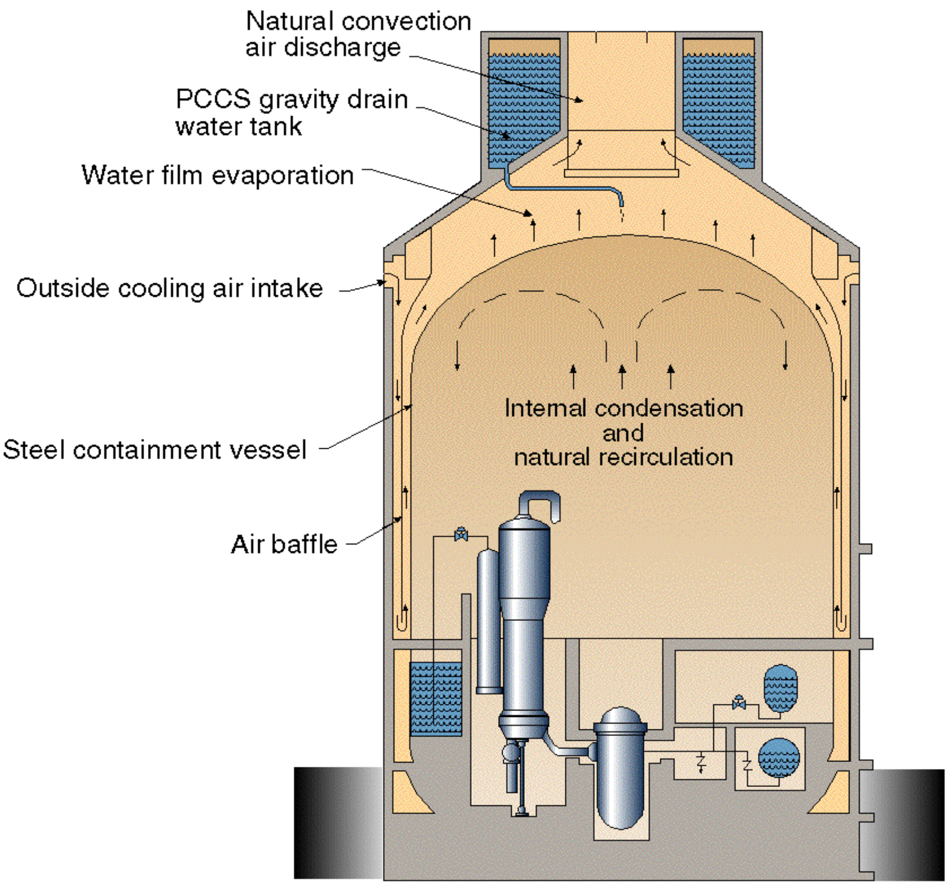

Figure 3 - AP1000 Passive containment cooling system

3 AP1000 COMPLIANCE WITH EUR

Compliance with the EUR has been a key design objective for the various plants studied and developed under the EPP Program. Assessments have been made, throughout the design process, to define the impact on the Westinghouse passive plant designs in meeting the EUR requirements.

As part of the EPP Phase 2D Program, the EPP Utilities proposed collaboration with the EUR group to prepare an EUR Volume 3 assessment for the AP1000 plant design. Following agreement by the EUR Steering Committee, the AP1000 EUR assessment commenced in January 2004 and was completed in March 2006 with the so colled EUR Administration Group Harmonization Meeting.

The EUR Volume 3 AP1000 Subset will include three main chapters:

- Chapter 1 – AP1000 Design Description
- Chapter 2 – Highlights of Results and Conclusions from the Analysis of Compliance
- Chapter 3 – Specific Requirements on the AP1000 Design by EUR

Overall scores for the AP1000 plant are to be considered the most positive in consideration of the reduced number of requirements that were not addressed, as a result of the complete documentation submitted that also included the AP1000 DCD approved by US NRC, and due to the decision taken at the beginning of the program to address the US plant with minor adaptation from EPP program.

In fact, while some of the NPPs addressed by the EUR have been designed taking the EUR as design requirements (e.g., EP1000 plant developed at the end of the '90 by the EPP organisation), the AP1000 plant has been designed following the EPRI URD requirements, integrating some major design feature to make the plant licensable in the largest number of countries around the world.

Nevertheless, EUR requirements have been considered in the development of the AP1000 plant design. In some areas, EUR requirements have even driven specific AP1000 design features (e.g., low boron core). In other areas specialized studies have been performed or are to be performed to specifically evaluate AP1000 compliance with European requirements in those areas that are felt the most important by the European Utilities. Examples include:

- the capability of the plant to accommodate at least 50% MOX fuel,
- a liquid radwaste system that incorporates boron recycle,
- heat removal systems designs (e.g., RHR, CCW, SWS) that can accommodate the EUR rapid cooldown requirements,
- assessment of the design to comply with the EUR DBA and DEC dose targets,
- the capability of the plant to mitigate European specific design basis accidents (e.g., fast boron dilution, multiple steam generator tube ruptures).
- Aircraft Crash Protection

4 EPP PHASE 2E OVERVIEW

Westinghouse and Ansaldo, in cooperation with two sponsor utilities (EdF and Swissnuclear) and with the thorough review of the AP1000 EUR Coordination Group (composed of the two aforementioned utilities and Tractebel, Iberdrola, TVO) completed the compliance analysis of the AP1000 Plant versus the EUR requirements. The dedicated EUR Volume 3 subset for AP1000 is scheduled to be issued at beginning of 2007. This represents the conclusion of Phase 2D of the EPP Program.

Some non-conformances found during this process have already been dealt with and other ones will have to be analyzed and solved in order to reach an AP1000 configuration suitable for construction in Europe while maintaining a close contact with the development and eventual construction of the AP1000 in the USA.

The new Phase of the EPP program, called Phase 2E started at the beginning of 2007. The program is supported by five European Utilities, namely EdF, Eon, RWE, SwissNuclear and Tractebel.

Based on direction provided by the Utilities members and by the EPP Steering Committee, Phase 2E activities are devoted to address the EUR non-conformances and the initial licensing steps in the EPP member countries.

In the following sections some major EPP Phase 2 activities are described.

4.1 Decommissioning Study

The purpose of this activity is to provide a preliminary Decommissioning Study for the AP1000 plant; to identify features which shall be considered at the design stage, as they may become relevant to the future decommissioning work.

The overall objective of the activity is to give evidence of the fact that the dismantling and decommissioning of the AP1000 reactor is feasible and, with respect to the operating reactor, the decommissioning costs and overall impact on the environment and personnel are minimized.

The activities to be performed include the following:

- Task 1: Outline of Decommissioning Objectives and Strategies
- Task 2: Evaluation of AP1000 Wastes and Overview of Waste Treatment
- Task 3: Review of Plant Layout Configuration for Decommissioning

In the first task an overview of the European strategies and policies for the decommissioning of nuclear installations has been provided. In particular, the regulating and licensing requirements in the various European countries have been analysed, s well as the decommissioning objectives for specific sites (green field, research centres, other plant locations), the approach relating to radiological protection to minimise doses to the workers and the public and other specific factors that may influence the selection of a decommissioning strategy for the AP1000 plant.

On the basis of the previous study, the identification and analysis of the factors influencing the selection of strategies for the decommissioning of nuclear installations in the European Union (EU) Member States has been carried out.

In addition to safety and the availability of practical decommissioning techniques, the following issues were identified to be particularly relevant to the selection of a decommissioning strategy:

- The basic decommissioning options; the scope of the decommissioning activities.
- The reactor type; the reactor size; the number of units on a site; the operational history.
- Project planning; analysis of material flow.
- Regulatory and policy requirements (timing; release criteria).
- Socio-economic issues.
- Waste management provisions.

A summary report has been prepared that defines the main decisions and plant specific parameters that may influence the decommissioning decisions and strategies and the overall decommissioning costs.

Task 2 objective is to provide an evaluation of AP1000 decommissioning materials/volumes, an overview of the wastes categorization and waste treatment.

The AP1000 passive plant provides a distinctive and measurable advantage with respect to active plants. In fact, the use of passive safety systems limits the use of support systems and permits an overall simplification of the plant auxiliary systems, see Figure 4. The use of passive systems and the resulting RCS compact design, also result in a major reduction of NI volumes and masses.

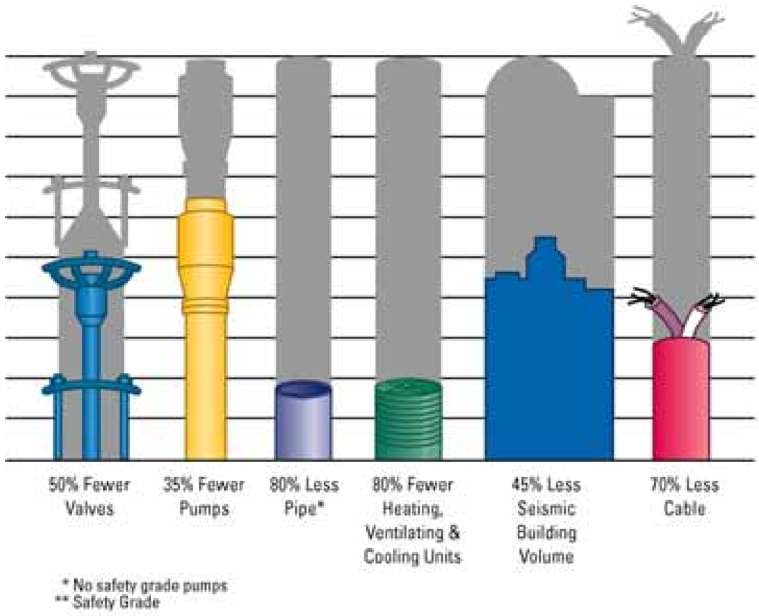

Figure 4: Reduction Bulk Quantities and Equipment Number

The evaluation of the AP1000 plant volumes is performed making a large use of data from the 3D PDS plant model. This tool has also been used to provide examples of dismantling of critical areas using the so called 4D Model that links the 3D geometrical model to a project tool (e.g., Primavera)

The evaluation (decommissioning material and radiological inventory) will be the basis for future decommissioning cost estimate.

Task 3 activities have focused the plant layout features that may impact the decommissioning activities.

In lessons learnt from actual decommissioning operations, areas have been identified and options formulated that should be considered when designing new facilities, the objective being to reduce worker exposure, to minimise waste generation and to simplify dismantling procedures, which will also result in cost saving.

Meeting these objectives requires that structures and equipment be designed in such a way that:

- Activation of materials is limited as much as possible;
- Contamination of plant and equipment can be avoided as much as possible;
- Contaminated or active areas can be easily separated from non-contaminated areas;
- Adequate space and access points are provided to allow the use of special tools and equipment for remote operation and handling, and also to allow the installation of appropriate shielding;
- Plant and equipment items can be easily dismantled, handled and transported, and that adequate openings are provided to allow for easy removal of components and materials from active areas;
- Equipment and buildings can be easily decontaminated;

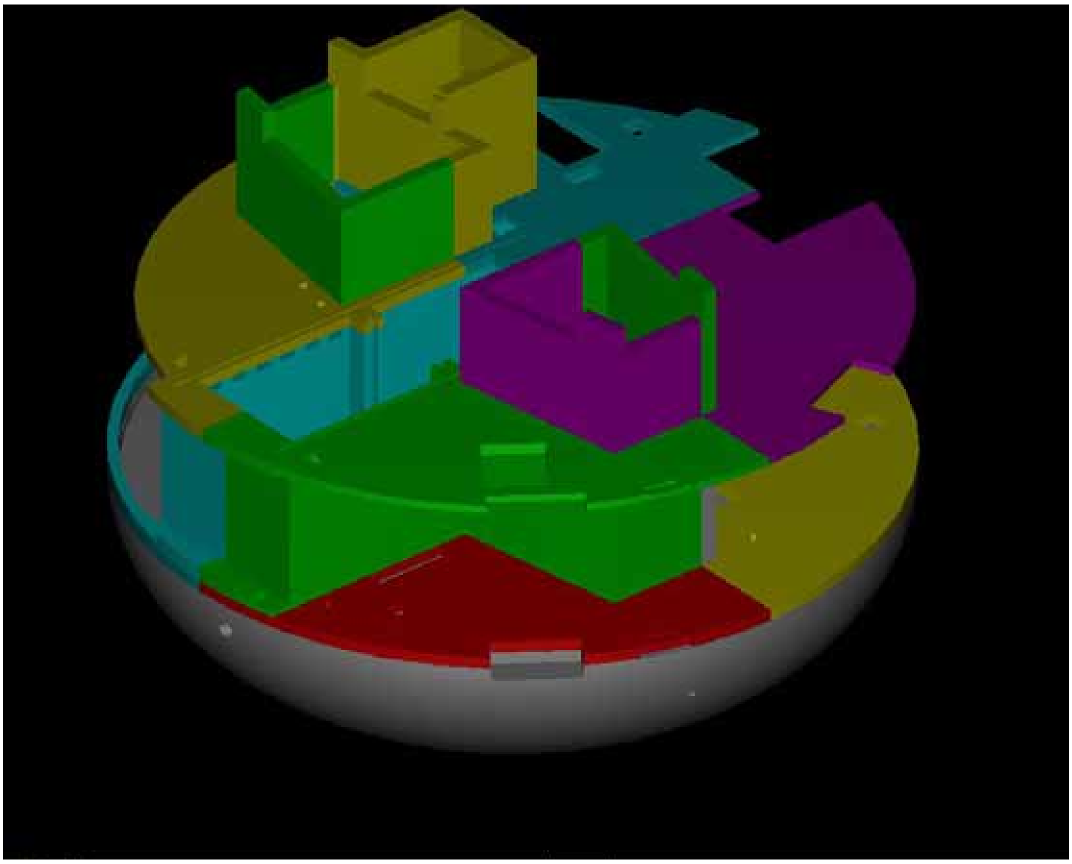

Figure 4: Use of the 3D Plant Model to Define Masses: Concrete Mass Inside Containment

AP1000 design has been analyzed according to the above lesson learned and major applicable requirements (e.g., EUR).

The overall results of the activities can be summarized as follow:

- The AP1000 plant can be safety dismantled and decommissioned;
- The AP1000 simplified plant design minimize decommissioning waste;
- Activation of materials is limited both by using stringent materials specification (e.g., limit of cobalt in primary components) and proper design of neutron shielding; this minimizes both operation and decommissioning doses;
- Separation between radioactive and non-radioactive equipment and areas is enforced
- Special features are implemented to facilitate decontamination of equipment and buildings (e.g., Structural modules);
- Adequate space, staging area and access are provided for easy dismantling and transportation of equipment

4.2 Aircraft Crash Study

The EUR states that “protection against aircraft crash shall be based on probabilistic approach unless the authorities require a deterministic approach”, with the following comment: “In many European countries only a probabilistic demonstration is requested for aircraft crash. In other countries the demonstration must be based on a deterministic approach i.e. against a regulatory loading function and associated criteria.”(EUR 2.1.5.3.4)

The purpose of this study was to define and evaluate enhancements to the AP1000 plant design to protect against an aircraft crash (ACC) with the goal of minimizing impacts on the existing plant layout and preventing perforation of the steel containment vessel (SCV). As a result of the studies, the AP1000 standard plant design includes provisions to address ACC scenarios. The external event of a terrorist initiated aircraft impact is considered for the AP1000 according to recently adopted U.S. NRC requirements. The plant resilience is demonstrated as a beyond Design Basis issue. The AP1000 ACC methodology was reviewed and found to be acceptable by an industry expert group which included several European Utility members. A technical report [4] describing the ACC evaluations has been submitted for review by the US NRC as part of their ongoing review of AP1000 for US utility combined license (COL) applications.

The AP1000 ACC assessment performed under the EPP Program considered both military and commercial aircraft impact. The ACC evaluations demonstrate that the AP1000 plant is able to provide adequate protection of the public health and safety with respect to aircraft impact. The aircraft impact would not inhibit AP1000's core cooling capability, containment integrity, spent fuel pool integrity, or adequate spent fuel cooling.

In this frame, Ansaldo has developed a new concept for the shield building roof and the air inlet opening (Fig. 5).

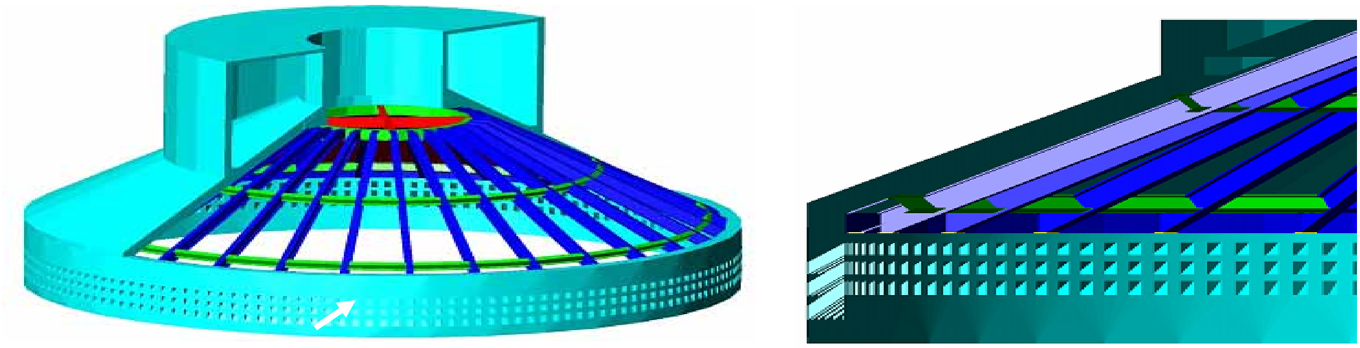

Figure 5: Redesign of Shield Building Roof

4.3 Secondary Containment Study

Per the EUR (Vol. 1, Appendix B), secondary containment is the final fission product confinement envelope which surrounds the following:

- Primary Containment
- Primary Containment penetrations and isolation valves
- Part of systems and components, connected with the reactor pressure boundary or the Primary Containment atmosphere, which may transport highly contaminated fluids outside Primary Containment.

The AP1000 primary steel containment vessel is characterized by a design leak rate of 0,1% vol/day at design pressure of 59 psig.

According to EUR "The optimal combination of the performance of the Primary Containment and the extent and the performance of the Secondary Containment, may be different for each type of confinement. In particular the requested extent of the Secondary Containment will appear as a consequence of the performance specified for the Primary Containment “.

Based on the above, AP1000 Policy, for the definition of the secondary containment boundary and performance is an optimization process that takes into account several parameters (for instance the ranges of leakrate of the Primary Containment, the range of efficiency of the Secondary Containment and its bypass rate). In this process Primary and Secondary containments are not considered individually but as a whole.

The first task of the activity consisted in the evaluation of the leak rate and its partition between the various leak paths.

AP1000 containment isolation is significantly improved by means of a large reduction in the number of penetrations (40 vs. 93 of conventional PWRs) and the number of normally open penetrations (11 vs. 38 of conventional PWRs)

The results demonstrate that the current design of the primary containment meets with margins both the leak rate design value and the 10 CFR50 Appendix J requirements.

On the basis of the above evaluation and of the requirements set in EUR a Secondary Containment boundary has been established.

In addition, following a realistic approach, leak rates have been neglected from closed systems, filled with water in post accident conditions (e.g., Component Cooling System, Fuel Transfer Tube) or equipped with filtering system. The above process allowed to reduce and simplify the rooms to be included in the SC boundary.

The secondary containment has been defined as the annulus between the shield building and containment vessel below the operating deck, the containment isolation valve penetration area, the Spent Fuel System penetration area, the RNS rooms located in the A/B and the staging areas of the equipment hatches.

The calculated Secondary Containment bypass (i.e. the sum of the leakages from the Primary Containment boundary released outside the Secondary Containment) is acceptable (largely below the EUR value of 10 %).

The SC efficiency, in terms of % of SC volume exchanged with the environment, can be assured without active support systems, but:

- Doors equipped with interlock/alarms system and appropriate gasket;
- Isolation dampers on ventilation ducts and penetrations in order to guarantee proper isolation in case of accident requiring Secondary Containment isolation.

Finally, Table 1 reports the results of the release evaluation following DBA according to the EUR methodology described in EUR Chapter 2.1.

It can be noted that the release requirements set by EUR for DBA, except for Iodines, can be meet without taking credit for Secondary Containment.

Table 1 – Releases evaluation with and without Secondary Containment following a DBA.

Design Target	**EUR limit (TBq)**	**With secondary containment**		**Without secondary containment**	
		External release (TBq)	**Compliance margin (%)**	**External release (TBq)**	**Compliance margin (%)**
No action beyond 800 m (Cat. 4)	5 E-03	5.02 E-04	90	1.26 E-03	74.9
Limited economic impact for I-131	10	5.55	44.5	13.9	- 38.6
Limited economic impact for Cs-137	1.5	0.3384	77.5	0.833	44.5

4.4 Core Design Study with 50% MOx Loading

The purpose of this study was to demonstrate the ability of the AP1000 reactor to meet the EUR MOx requirements without significant changes to the AP1000 plant design.

An important reactor physics consideration when designing a core with a combination of MOx and UO2 fuel is the impact of the differing neutron spectrum of the adjacent assemblies on the calculation of basic nuclear data for core modeling. Traditional modeling of an infinite lattice of a single assembly type in a higher order code is not sufficient for generating cross-sections and pin power reconstruction data for mixed MOx / UO2 cores. Instead, nodal code cross-section and pin power reconstruction data are calculated by modeling representative "mini-cores" of MOx and UO2 in the higher order lattice code.

This interaction between different assembly types also impacts the intra-assembly power distribution. Power peaking within the MOx assemblies can be controlled by selective enrichment zoning of the rods within the lattice, with the lowest enrichments on the assembly periphery to counteract the thermal flux current from the adjacent UO2 assemblies.

A key concern with MOx from a fuel performance perspective is the high fuel temperatures relative to UO2 fuel resulting in an increase in fission gas release. This issue with MOx fuel can result in a lower fuel burnup limit with respect to the no clad lift-off rod internal pressure limit, potentially limiting the number of cycles that MOx fuel can operate. The approach taken to mitigate this effect is to design the MOx fuel rod with annular pellets. The annular fuel decreases the peak pellet temperature which reduces fission gas release from the pellet, and also provides an increase in the fuel rod internal volume. Such a design can permit MOx fuel assemblies to operate for four annual cycles up to maximum rod burnups consistent with high burnup UO2 fuel.

A typical equilibrium annual cycle loading pattern with a 50% MOx fuel loading is shown in Figure 6. The feed region of this core design consists of 24 UO2 fuel assemblies containing various numbers and loadings of gadolinium rods for peaking factor control, and 24 MOx assemblies without any burnable absorbers. This core loading yields an annual cycle energy output of 338 EFPD at 3400 MWt in an AP1000 reactor.

The MOx fuel assemblies in this design consist of full length annular fuel rods with an annulus diameter of approximately 4 mm. This fuel rod design permits the MOx assemblies to operate for four annual cycles while maintaining the fuel rod internal pressure below the reactor coolant system operating pressure (≤15.5 MPa). This design, however, reduces the heavy metal loading of the MOx assemblies by approximately 23% relative to a solid rod design. As a result, a total of 48 fresh fuel assemblies are loaded each cycle to maintain the MOx lead rod burnups within the range of high burnup UO2 fuel experience. This is in comparison to a 100% UO2 core, which would feed 40 fresh assemblies for the same cycle energy output.

The peaking factor behavior of the core loading shown in Figure 1 is very well behaved over the duration of the cycle, with the cycle maximum integrated rod power being very similar to that of a 100% UO2 core as shown in Table 1.

The impact of the harder neutron spectrum on the 50% MOx core is illustrated by the critical boron concentration and moderator temperature coefficient (MTC) differences listed in Table 2. From Table 2, it can be seen that the MOx core design results in beginning of cycle critical boron concentrations 300 to 400 ppm greater than those in the 100% UO2 core design. However, even though the 50% MOx design has much higher BOC critical boron concentrations, the corresponding MTC is approximately 1.7 pcm/°C more negative than the

MTC for the 100% UO2 core. In a 100% UO2 core, this amount of boron concentration increase would typically increase the MTC by approximately 2 pcm/°C. Therefore, the neutron spectrum effect is quite significant on the BOC MTC.

Another impact of the harder neutron spectrum is a reduction in the control rod worth and available shutdown margin in the 50% MOx core as shown in Table 2. While the 50% MOx core has less available shutdown margin than the 100% UO2 core, the available shutdown margin still exceeds the 1.6% Δρ assumed in the AP1000 design bases.

The study illustrates that the AP1000 reactor is capable of operating with a core design consisting of both UO2 and MOx feed fuel, meeting the EUR requirements for up to 50% MOx feed regions. The MOx assembly designs are capable of operating for four annual cycles while still meeting the no clad lift-off fuel rod design criteria. Peaking factor margins of the 50% MOx core are consistent with those from a 100% UO2 design. While some excess shutdown margin is lost in the 50% MOx design, the available shutdown margin still exceeds the design requirements with comfortable margins without having to change the design of the AP1000 control rods.

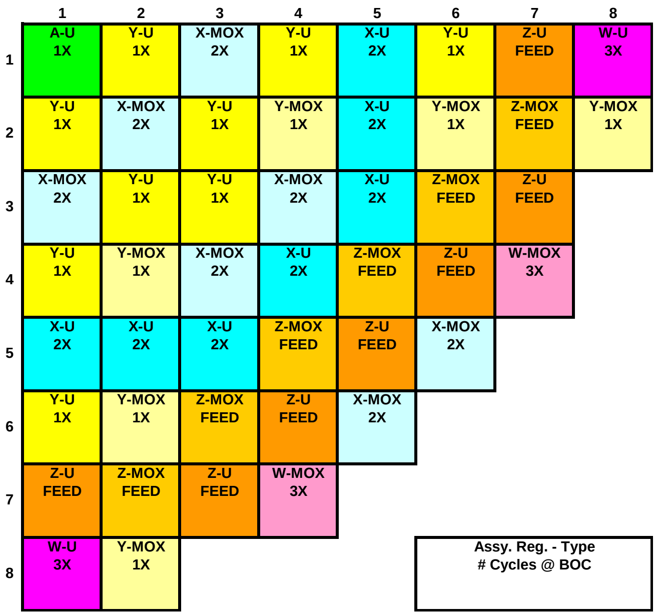

Figure. 6 Typical Loading Pattern for Annual Equilibrium Cycle with 50% MOx Loading

Parameter	100% UO2	50% MOx
Cycle Max. Integrated Rod Power, FΔH	1.494	1.48
BOC, HFP, ARO, EQXE CB (ppm)	1133	1412
BOC, HZP, ARO, No Xe CB (ppm)	1789	2187
BOC, HZP, ARO, No Xe MTC (pcm/°C)	-7.0	-8.7
EOC Shutdown Margin (%Δρ)	3.18	2.52

Table 2 Summary of Key Physics Parameters for 100% UO2 and 50% MOx Annual Equilibrium Cycles

5 CONCLUSIONS

The AP1000 program activities performed under the EPP Program further confirm the potential capability of passive PWR technology in meeting the safety standards established by the EUR while keeping a cost competitiveness The EUR requirements are being considered in the development of the AP1000 plant design. In some areas, EUR requirements have driven specific AP1000 design features.

The EPP Program and AP1000 EUR compliance assessment are valuable elements for sustaining a long-term positive view of nuclear power contributions to Europe's energy supply mix. It has become increasingly clear that nuclear power generation additions are most competitive with other energy choices when a standard plant design can be applied in multiple locations.

EPP Phase 2E activities further established that the AP1000-Europe plant design complies with the latest European Utility Requirements while retaining the standard plant economic advantage of being largely the same AP1000 plant design as for the U.S.

ACKNOWLEDGMENTS

The authors thank the European utilities currently participating in the EPP Phase 2E program, SwissNuclear (Switzerland), Electricité de France (France), Suez-Tractebel S.A. (Belgium), E.ON Kernkraft GmbH (Germany) and RWE Power AG (Germany), along with Westinghouse and Ansaldo Nucleare. Special thanks to Kathy Demetri (Westinghouse) for her precious contribution.

REFERENCES

[1] European Utility Requirements (EUR), Revision C, April 2001

[2] EUR Volume 3 AP1000 Subset, Revision A, June 2006

[3] APP-GW-GL-700, Revision 16, AP1000 Design Control Document (DCD)

[4] APP-GW-GLR-126, March 2008, AP1000 Nuclear Island Response to Aircraft Impact

DEVELOPMENT OF NATIONAL INFRASTRUCTURE FOR SAFE AND RELIABLE NUCLEAR POWER

R.I. Facer, N. Pieroni, A. Starz
International Atomic Energy Agency
Wagramer Strasse 5, 1400 Vienna, Austria
R.I.Facer@iaea.org; N.Pieroni@iaea.org; A.Starz@iaea.org

ABSTRACT

The development of an appropriate national infrastructure is an essential task for countries interested in introducing nuclear power or expanding the nuclear power programme. The infrastructure includes both software (legal, regulatory, training, etc) and hardware (grid, facilities, etc) that fulfil the conditions necessary for the safe, economic and reliable nuclear power plant (NPP) operation. Countries need to consider the whole spectrum of (safety, security, environment, finance, industrial etc) issues and the commitments and resources involved over the lifecycle of the programme.

The activities to prepare the infrastructure can be split into three progressive phases of development. The duration of these phases will depend upon the degree of commitment and resources applied in the country. The IAEA published a guide outlining 19 infrastructure issues that are expected to be addressed in each phase. The completion of the infrastructure conditions of each of these phases is marked by a specific milestone at which the progress and success of the development effort can be evaluated and a national decision made to progress to the next phase:

Milestone 1: Ready to make a knowledgeable commitment to a nuclear programme.

Milestone 2: Ready to invite bids for the first NPP.

Milestone 3: Ready to commission and operate the first NPP.

The initial development of the nuclear infrastructure can be facilitated by the establishment of an interdisciplinary team assigned to develop policies and make recommendations. This study group is identified as a Nuclear Energy Programme Implementing Organization (NEPIO). The role of the NEPIO particularly in the achievement of Milestone 1, addressing Government commitment and authorities, is explained. Also the responsibilities and functions of NEPIO, its composition including Government agencies, industry, and other stakeholders, and the competencies and resources necessary to complete the tasks of the NEPIO, are briefly described.

Many countries have some, if limited, infrastructure to support a nuclear power programme. To understand where a country is in its development, a methodology for the evaluation of the national nuclear infrastructure development status was elaborated based on the milestone approach. The envisaged guidance can be used either for the country self-evaluation or for an external evaluation. This will allow countries to evaluate or ask others to help them evaluate the level of their present readiness to introduce nuclear power and to determine those issues in which they need to make additional commitments.

1 INTRODUCTION

Many parts of the world have a pressing need for sustainable development from reducing poverty and raising living standards to improving health care, and industrial and agricultural productivity. Nearly every aspect of development requires reliable access to energy sources. Nuclear power can play a role in providing improved access to affordable energy. Careful planning in the early stages of a programme across a wide range of national infrastructure issues can help instil confidence in the country's ability to legislate, regulate, construct and safely and securely operate a nuclear power plant (NPP).

Many countries have approached the IAEA regarding support for the introduction of nuclear power. Under its Statute, the IAEA is authorized to assist any Member State that is considering or has decided to "go nuclear" to meet its energy needs. The Agency has considerable experience in doing this through its assistance programmes. The IAEA has recently prepared a number of guidance publications on infrastructure development for countries planning to launch a nuclear power programme consisting of the construction of one or more NPPs and all or any of the necessary supporting facilities and stands ready to provide expert assistance in this area if requested.

1.1 Development of national infrastructure

The IAEA publication "Milestones in the development of a National Infrastructure for Nuclear Power" [1] describes 19 infrastructure issues to support the infrastructure development. Early attention to all of these issues will facilitate the efficient development of a successful national nuclear power programme. Equally, lack of appropriate attention to any of the issues is likely to result in future difficulties that may significantly delay or otherwise affect the successful introduction of nuclear power.

The national infrastructure cover a wide range of issues, from the physical facilities and equipment associated with the distribution of electricity, the transport of the material and supplies to the site, the site itself, and the facilities for handling the radioactive waste material, to the legal and regulatory framework within which all of the necessary activities are carried out, and the human and financial resources necessary to implement the required activities. In short, infrastructure issues, as discussed in this paper, include all activities and arrangements needed to set up and operate a nuclear power programme.

The decision by a State to embark on a nuclear power programme should be based upon a commitment to use nuclear power for peaceful purposes, in a safe and secure manner within a stable political, economic and social environment. A viable NPP project and nuclear power programme will require the establishment of a sustainable national infrastructure that provides governmental, legal, regulatory, managerial, technological, human and industrial support for the nuclear power programme throughout its life cycle. This infrastructure is relevant whether the nuclear power programme is planned for the production of electricity, for seawater desalination or for any other peaceful purpose.

The development of a nuclear power programme entails sustained attention to many interrelated activities over a long duration and involves a commitment of at least 100 years throughout NPP operation, decommissioning and waste disposal. As with any major project, the commitment of resources for the NPP project needs to be phased and decisions to move to subsequent phases, where the commitment of resources will increase significantly, need to be made with a full understanding of the requirements, risks and benefits. Many of these issues are at all times the responsibility of the State authorities, others may also be State controlled. However, in some countries the responsible organisation for the construction and operation of a NPP may be in private or partially private ownership. Irrespective of the ownership, the utility/ operator will need to establish the necessary infrastructure to manage and control the NPP, including the establishment of human resources, and the long term arrangements. The infrastructure issues are presented as a coherent whole, recognising that eventually many different organisations may have the direct responsibility for developing and implementing the infrastructure.

Experience has shown that the time frame from the initial policy decision by the State to the operation of the first NPP may well be 10–15 years. For a country with a little-developed technical base the implementation of the first NPP would be expected to take closer to 15 years. A country with a strong technical base could take 10 years if it makes a significant and concerted effort to implement a programme. Even countries with existing nuclear power programmes may take about 10 years to approve and construct a new NPP.

1.2 Fundamental Importance of Safety, Security and Safeguards

The fundamental nuclear safety objective is to protect people and the environment from the harmful effects of ionizing radiation. A comprehensive safety culture needs to be developed that permeates all infrastructure development activities. The IAEA publication "Fundamental Safety Principles" [2] contains ten safety principles that represent the international consensus on the high level of safety required for the sustainable use of nuclear power. The first principle establishes that the ultimate responsibility for safety must rest with the operator. It is incumbent on the leadership and management of the State and the operator to develop awareness, encouragement and enforcement of a safety culture throughout the entire programme. It cannot be overemphasized that everyone involved in a NPP project carries a responsibility for safety.

In addition to nuclear safety, and no less significant, are the issues associated with the control of nuclear material, either to ensure the security of the material, or to ensure that all of the activities in a State can be demonstrated to ensure that there is no risk of proliferation of nuclear weapons and that all the materials are adequately accounted for and protected. This also requires the development of a culture, system and practices that ensure that all staff are aware of their responsibilities and the importance of their actions.

2 INFRASTRUCTURE PHASES AND MILESTONES

2.1 Overview

The IAEA publication "Milestones in the Development of a National Infrastructure for Nuclear Power" [1] makes the distinction between a nuclear power programme and a NPP project. A nuclear power programme encompasses all of the elements of a national infrastructure that would support the first NPP as well as any planned expansion. It would include consideration of strategic decisions such as human resource development, industrial support and fuel cycle planning. The nuclear power programme is thus the responsibility of the Government itself. The NPP project is defined as being related to the plant itself, and is likely to be managed by the owner-operator. Before a NPP project can proceed, a nuclear power programme must establish the infrastructure to support that NPP project during its planning, construction, operation and eventual decommissioning.

The milestones approach for implementation of the national infrastructure is depicted in Figure 1 taken from [1] that shows the activities split into three progressive phases.

The duration of these phases will depend upon the degree of commitment and resources applied in the State. The duration, especially the bidding process and construction phases, may also be influenced by national requirements, such as public consultation during the licensing process. The term "infrastructure milestone" refers to the point at which it can be demonstrated that the preceding phase has been successfully completed and that the State is fully prepared to embark on the subsequent phase. The "infrastructure milestone" is thus a set of conditions and does not necessarily have specific time based implications.

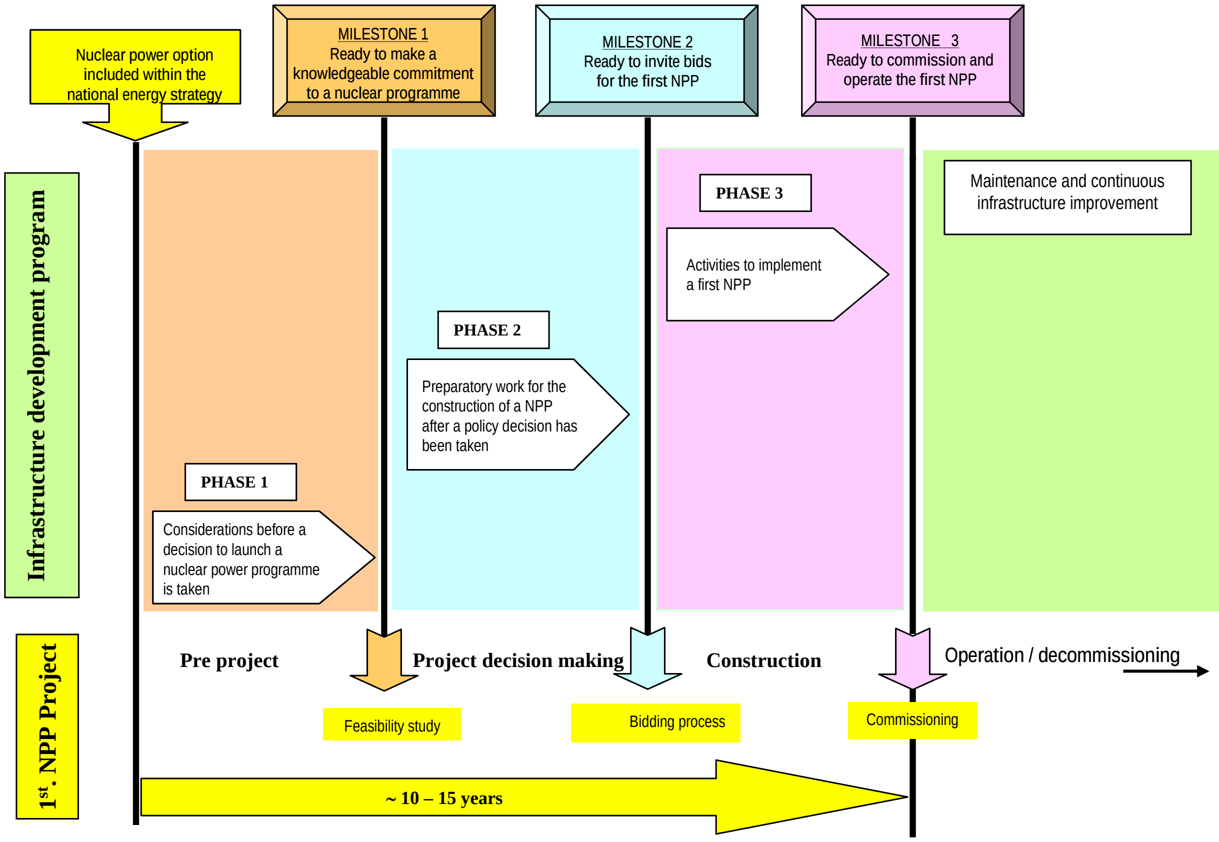

Figure 1: Nuclear infrastructure development programme

For each milestone, 19 issues that need to be considered are shown schematically in Table 1. The order of the issues does not indicate an importance or hierarchy. Each issue is important and requires careful consideration.

The development of the infrastructure necessary to support a nuclear power programme would be expected to proceed through Phases 1–3, leading to the achievement of the corresponding milestones, while at the same time many other specific activities are progressing in order to ensure implementation of the first NPP project. The three programme phases of development are:

—Phase 1: Considerations before a decision to launch a nuclear power programme is taken;

—Phase 2: Preparatory work for the construction of a NPP after a policy decision has been taken;

—Phase 3: Activities to implement a first NPP.

The completion of the infrastructure conditions of each of these phases is marked by a specific milestone at which the progress and success of the development effort can be assessed and a decision made to move on to the next phase. These milestones are:

—Milestone 1: Ready to make a knowledgeable commitment to a nuclear power programme;

—Milestone 2: Ready to invite bids for the first NPP;

—Milestone 3: Ready to commission and operate the first NPP.

Table 1: Nuclear infrastructure issues and milestones

ISSUES	MILESTONE 1	MILESTONE 2	MILESTONE 3
National position			
Nuclear safety			
Management			
Funding and financing	CONDITIONS	CONDITIONS	CONDITIONS
Legislative framework			
Safeguards			
Regulatory framework			
Radiation protection			
Electrical grid			
Human resources development			
Stakeholder involvement			
Site and supporting facilities			
Environmental protection			
Emergency planning			
Security and physical protection			
Nuclear fuel cycle			
Radioactive waste			
Industrial involvement			
Procurement			

2.2 Conditions to reach Milestone 1

This is the point at which the State would be in a position to make an informed decision on whether it is appropriate to introduce a nuclear power programme. In order to achieve this milestone the State will not only have assessed that it needs additional energy and included nuclear power as a possible option to meet some of these needs, but will also have carried out the first phase of the programme, which would culminate in the attainment of Milestone 1.

The initial phase in the development of a nuclear power programme involves the considerations and planning before a firm decision to develop a nuclear power programme is taken. During this phase the responsible organizations are the Government and the government formed multi-disciplinary group charged with the initial investigation and promotion of nuclear power programme, here referred to as the “Nuclear Energy Programme Implementing Organization” (NEPIO). The NEPIO should be appropriately staffed and resourced and be given the authority to carry out its work. It should develop for the State a complete understanding of the commitments associated with the use of nuclear power. Their report at the end of Phase 1 should clearly show an understanding of the infrastructure that needs to be developed and demonstrate viable plans for its introduction, identifying resource requirements and timescales. It should include plans for the development of organizations to undertake the role of regulator, owner, operator and technical support. It is also essential even at the earliest phases that recognition of an appropriate safety culture is developed in each organization and their responsibilities for ongoing safe operation.

2.3 Conditions to reach Milestone 2

At this point the infrastructure in the State would be sufficiently developed so that the responsible organizations, State owned or private, would be in a position where inviting formal bids for the first NPP would be possible with confidence that all national and international infrastructure issues have been resolved.

Following the policy decision to proceed with the development of a nuclear power programme, substantive work for achieving the necessary level of technical and institutional competence will have been undertaken. This phase requires a significant and continuing commitment from the Government, which continues to have a role as an advocate and guiding organization for the State's programme. The necessary legal framework will be in place and separate regulator and operator organizations will have been established.

The State will have carried out the work required to prepare for the necessary infrastructure for construction of a NPP. The regulatory body will need to be developed to a level at which it can fulfill all of its oversight duties. The necessary infrastructure should be developed to the point of complete readiness to request a bid or enter into a commercial contract. The owner/operator (or utility), which may or may not be State owned, has a key role at this time, ensuring that it has developed the competence to manage a NPP project, to achieve the level of organizational and operational culture necessary to meet regulatory requirements, and the ability to demonstrate that it is an adequately informed and effective customer.

2.4 Conditions to reach Milestone 3

At this point the State will be in a position to commission and operate the first NPP. The completed work programme will have brought the Government to the point of having established a nuclear power programme that will bring the benefits of energy security and economic development envisioned in the initial policy decision. The owner/operator will have developed from an organization capable of ordering a NPP to an organization capable of accepting the responsibility for commissioning and operating one. This will require significant development and training for all levels of staff, and the demonstration that the owner/operator can manage the NPP project throughout its life.

While achieving the third milestone is a major accomplishment, it should be remembered that it is only the beginning of a lasting commitment to the safe, secure and effective application of nuclear power. Confidence to move beyond Phase 3 will be enhanced by the use of existing international assessment and review methodologies.

3 RESPONSIBILITIES AND COMPETENCIES OF NEPIO

3.1 Overview

A State generally needs information to make a knowledgeable decision about a major policy initiative such as the launch of a nuclear power programme. For convenience, the milestones publication [1] labels the organization charged with developing such information NEPIO. The important aspects of the NEPIO are that it be given the appropriate authority and resources to prepare the material necessary for a State decision and that it be able to work closely with all of the relevant stakeholders internal and external to the Government. As a pre-decisional body, its leadership by a trusted and respected person is vital to the credibility of the effort.

The initial purpose of the NEPIO is to compile the information necessary for a knowledgeable policy decision to proceed with the development of a nuclear power programme so that this decision can be made with full realization of all that it entails. During Phase 1, the NEPIO researches, studies and makes policy and strategy recommendations for the Government with respect to the 19 infrastructure issues called out in [1]. While the resolutions of all 19 issues do not have to be finalized by Milestone 1, their implications and the approach to their resolution must be considered. It is the responsibility of the NEPIO to see that this is done. If the decision is made to proceed with a nuclear power programme, specific areas of responsibility may eventually migrate from the NEPIO to other organizations such as the regulatory body and the owner/operator.

During Phase 2, the NEPIO may become the champion and advocate for seeing that the policies and strategies are turned into firm action plans for each of the 19 issues. As it changes roles from policy-formation to coordination, it may also see that the corresponding responsibilities are assigned to those institutions which will become a permanent part of the overall programme infrastructure. As these organizations assume their responsibilities, the NEPIO should assume an oversight roll to assure that the overall programme is proceeding as envisioned. The NEPIO's functions are most important to the long-term success of the nuclear programme in Phase 1 and 2. In an advocacy role, it may continue in Phase 3, although by then other institutions may take over this role.

3.2 Description of the NEPIO[1]

The purpose of the NEPIO is to lead the national effort to come to a firm decision with respect to the introduction of nuclear power. It should be a specifically assigned body with clear leadership. Appropriate authority needs to be provided to develop the staff and to include other government organizations as may be necessary. The establishment of the NEPIO should be at a senior government level so that this authority is understood by all. It should have available to it the appropriate expertise to address all of the infrastructure issues identified in [1]. Some of this expertise may be temporarily seconded from other Government organizations, or filled by consultants.

During Phase 2 of the development effort, the NEPIO should oversee the transition of infrastructure development to specific, permanent organizations which will carry on the long term responsibilities for regulating, constructing and operating a fully functional nuclear energy programme. The staffing of the NEPIO may have to be adjusted as the State proceeds through Phase 2 to the achievement of Milestone 2.

With the achievement of Milestone 2, the responsibilities for the various aspects of a nuclear power programme should have been assumed by permanent entities.

3.3 Responsibilities

During Phase 1, the NEPIO will be responsible for compiling all the information necessary for the Government to make an informed decision on whether or not to proceed with the development of a nuclear power programme. If a positive decision to do so is taken, the NEPIO will be responsible during Phase 2 for coordinating and overseeing the development of the necessary infrastructure to bring the country to a point of issuing a bid for

[1] The IAEA is developing a Technical Report providing practical guidance on the responsibilities, competencies and interfaces needed by NEPIO, scheduled to be issued by end 2008.

the first NPP project. The NEPIO will study the 19 issues discussed in [1] and produce a comprehensive report clearly laying out the commitments and processes necessary to undertake a nuclear power programme. It may also produce the implementation plan for Phase 2, with resource requirements to which the Government would have to commit to launch this Phase 2.

The outcome of the NEPIO's work will be a National Position on the introduction of nuclear power which demonstrates an understanding of the nation's energy needs and the alternative options available for meeting those needs. Along with the energy needs, the existing size and design of the electrical grid should be thoroughly understood. In addition, the international commitments, bilateral relationships, and implications for technology strategy should be understood. Available, existing NPP designs can then be studied to assess their compatibility with those needs and the present grid structure.

During Phase 2, the NEPIO will assume a coordination and oversight role as the owner-operator, the regulator and other organizations take on permanently the functions necessary to carry out the national strategy.

3.4 Structure, Competencies and Life Span of NEPIO

A possible structure for the NEPIO organisation is shown in Figure 2.

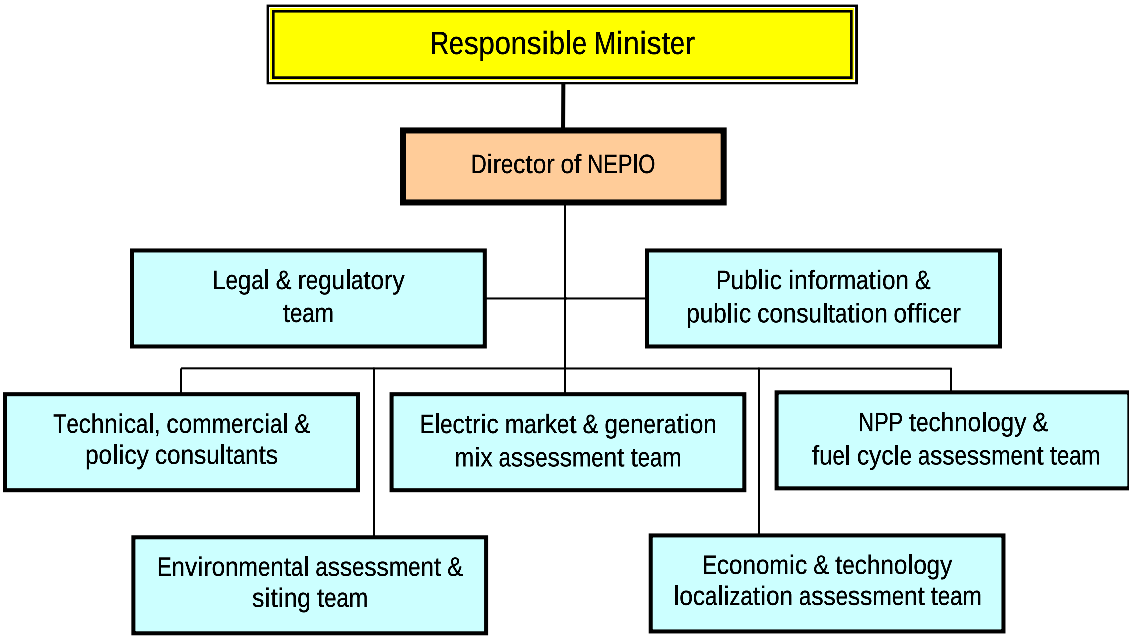

Figure 2: Example of a Nuclear Energy Programme Implementing Organization (NEPIO)

To successfully accomplish its responsibilities, the NEPIO must be staffed with individuals capable of exploring and understanding each of the 19 infrastructure issues. Broad use of consultants or interactions with international organizations is strongly encouraged, especially where domestic expertise may not be available. However, the leadership and decision making should remain with national authorities.

During Phase 1 of a nuclear programme development, the NEPIO should be the lead organization guiding the State to Milestone 1. It has to completely define the commitments and requirements necessary to employ a safe, secure and peaceful nuclear power programme. Equally important, during Phase 2, it will have to see that those commitments and requirements are assumed and carried out by the permanent organizations designated to do so. According to experience, it is likely that the members of the NEPIO will become leaders of the respective institutions and organizations designated to implement the first and subsequent NPP projects.

4 EVALUATION OF INFRASTRUCTURE DEVELOPMENT STATUS

4.1 Overview

The publication ”Milestones in the Development of a National Infrastructure for Nuclear Power” [1] provides guidance on the timely preparation for a nuclear power programme through a sequential development of the necessary infrastructure issues. The development is described through a framework of milestones that mark the completion of infrastructure conditions during the progressive phases. The guidance can be applied to evaluate the progress and to aid in planning the further steps necessary for a consistent development of the national infrastructure. The practical implementation of the evaluation requires a suitable means to determine the status of the infrastructure conditions[2].

The evaluation methodology is intended to complement the information presented in [1] by providing a criteria for appraising the status of a country against each of the infrastructure issues. It is focused on the early two phases of the NPP project (up to ready to begin construction) for two key reasons:

- it is important in any major national project to invest wisely and effectively in the early preparatory stages,
- existing IAEA assessment tools and methodologies already provide a sound basis for assessing the status of infrastructure during Phase 3 and beyond, i.e. construction and operation.

The intent of the evaluation is to allow a holistic review of a State readiness to move forward to the next phase of introducing a nuclear power programme. It is also a means for identifying gaps and focusing resource allocation. It is vital to evaluate readiness across all 19 infrastructure issues because each and every one is essential and because there are significant interactions. The management of each infrastructure issue and the human and financial resources required to support them need to be fully integrated.

This holistic evaluation tool based on the IAEA milestones approach can be used either as a self evaluation tool by any State wishing to review its readiness to proceed to the next phase of a nuclear power programme or as a peer review tool where the State can invite others to carry out an independent evaluation of their readiness.

The aim of the evaluation tool is:

- To ensure that all relevant infrastructure issues are reviewed
- To ensure consistency between infrastructure issues
- To ensure a consistent approach between countries thereby providing other countries and potential vendors with an internationally recognized evaluation of a countries readiness
- To bring the results together in order to identify actions required to move into a subsequent phase of the project to establish a nuclear power programme.

[2] The IAEA is developing a Technical Report providing a means of evaluation based on the milestones approach, scheduled to be issued by the end 2008.

4.2 Evaluation Approach

The basis of the evaluation is a review against the criteria developed for each condition of each infrastructure issue. The scope of this evaluation includes both the 'hard' (grid, facilities, etc.) and 'soft' (legal, regulatory, training, etc.) infrastructure needed for a NPP. Criteria are defined for each issue of Milestones 1 and 2 in the development of a nuclear power programme. The purpose of the criteria is twofold:

- Firstly to check that all the work required in the phase leading up to the milestone has been adequately completed
- Secondly to help ensuring that the plans for the following phase are comprehensive and realistic

Operation, decommissioning, spent fuel and waste management are addressed to the degree necessary prior to NPP commissioning. All the issues, including those for operation and decommissioning, as well as for spent fuel and waste management, should be considered by the time the bid request is issued. Having reached the point of readiness to commission a NPP, the Member State should have developed an understanding of the commitments required for a successful nuclear power programme and be able to uphold those commitments throughout the NPP life.

In general the evaluation at Milestone 1 is looking at the proposed work programme for Phase 2 and beyond in order to establish if the requirements have been fully understood, scoped and resourced. It is important to look at what infrastructure and operations already exist. For example, countries contemplating a nuclear power programme will already have in place an operational, legal and regulatory system for the safe use and transport of radioactive material and may have a research reactor in operation. One of the key inputs to the overall evaluation will be the results of national and international evaluations of existing activities.

At Milestone 1, there is no nuclear safety risk related to nuclear material; the evaluation is mostly about programme risk management. A State can do less planning in Phase 1 but then may carry a much greater risk of delays or unexpected outcomes because the necessary issues have not been properly scoped. The wide international experience of how best to control these project risks have be taken into account in the development of the evaluation methodology. The activities that need to be undertaken in Phase 1 in order to manage project risks include some activities that need to be started in Phase 1 because of the very long lead time. The typical example of this is related to siting where unless some activities towards site selection and characterisation are carried out in Phase 1, their long lead time will introduce significant delays into the programme.

One of the elements that the evaluation methodology looks for is a clear national commitment. A State will want to make use of international experience and cooperation in the introduction of a nuclear power programme. The use of partnership agreements with vendors and/or countries with experience of NPP operations and the use of recognised experts as consultants are encouraged by the IAEA. However, any evaluation of readiness to proceed to a further phase will want to ensure that a full ownership and understanding of the key issues is with the Member State wishing to implement the nuclear power programme.

4.3 Criteria for Evaluation and their Use

The methodology provides detailed criteria to obtain evidence that a particular condition has been met. They are presented for each of the 19 infrastructure issues and specific criteria are proposed for each condition. An example is provided in Table 2.

Table 2: Example of criteria for evaluating the issue National Position/Milestone 1

Issue: National Position / Milestone 1	
Conditions	**Criteria**
Safety, security and non-proliferation needs recognized	1. A document clearly demonstrating the Governments commitment to the safe, secure and peaceful implementation of nuclear energy for the long term.
NEPIO established and staffed	1. The charter showing that the NEPIO has been established by and reports to a Senior Government Minister 2. The basis of the charter is known by other Government ministries and key members of NEPIO 3. The NEPIO charter clearly charges and authorizes the preparation of a comprehensive report to identify the commitments and conditions necessary to establish a national nuclear power programme. It defines an adequate scope of investigations and clear definition of objectives and timescales. It should identify how its mandate and activities fit with overall plan for implementing nuclear power option 4. A clear description of how NEPIO operates in terms of funding, office accommodation and equipment, reference material 5. Evidence showing adequate interactions between and support from appropriate ministers such as those responsible for Energy, Environment, etc 6. A documented budget planning and reporting process showing appropriate funding is provided to and expended by NEPIO to fulfil its charter in the scheduled time 7. Organisation chart; job descriptions and CVs of members demonstrating appropriate skills, qualifications and experience to address all the infrastructure issues based on requirements in IAEA-TECDOC–1513 [3]. This includes appropriate use of consultants and the demonstration of national staff as “intelligent customer” 8. Comprehensive report produced by NEPIO covering all areas identified in [1] and recognising the resources and timescales required for the activities required for Phase 2. A demonstration that the Member State can provide the overall resources required integrated across all areas. 9. Executive summary of comprehensive report is based on detailed report, contains estimates of total resources and timescales and has been properly reviewed by senior government officials

For each issue, there should be a clear work programme for the next phase of the project which states the objectives of the work programme, the detailed activities, the funding and resources required, how it will be provided and the timescales for each activity.

Several of the 19 issues apply to any major project and need to be evaluated in a similar way as to any other project. However, there are often additional “nuclear” project features and these are identified under the appropriate issue.

Strong evidence of a holistic approach to information gathering, resource development and decision making is needed. It is, for example, no use having a small team fully aware of the nuclear safety requirements if there is not a clear plan to develop a competent operating organization with a strong safety culture, or a plan to ensure the capability to produce components with an assurance of the required integrity or reliability. This view will be obtained by looking at each of the 19 issues and then drawing the detailed evaluation together. For example, to be assured that Milestone 1 has been reached, it is necessary to see that:

- the State has the knowledge that is required;
- sufficient resources and attention of senior officers has been given to the analysis;
- the overall strategy and objectives are sound;
- programme risk is adequately managed;
- the plans for work and resources required for Phase 2 are sound;
- the existing activities associated with radiation sources are adequately managed; controlled and regulated and that has been benchmarked;
- there is strong government commitment to the programme.

The criteria proposed often refer to obtaining “evidence” and “plans”. Evidence can include reports, meeting notes, correspondence, talks and presentations, conferences attended with meeting report, discussions, CV’s, organisation descriptions, job descriptions etc. Plans need to have clear actions with associated timescales, resources required and evidence that they are available. In all cases there should be evidence that the documents have been approved by a person/organisation with the appropriate authority.

ACKNOWLEDGMENTS

The authors are the Scientific Secretaries responsible for the development of the IAEA publications related with infrastructure and planning for nuclear power programmes. The material of this paper was taken from several IAEA publications developed with the contributions from many experts from inside as well outside IAEA.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Milestones in the Development of a National Infrastructure for Nuclear Power, IAEA Nuclear Energy Series No. NG-G-3.1, IAEA, Vienna, Austria (2007).

[2] EUROPEAN ATOMIC ENERGY COMMUNITY, FOOD AND AGRICULTURE ORGANIZATION OT THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, Vienna, Austria (2006).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Basic infrastructure for a nuclear power project, IAEA-TECDOC-1513, IAEA, Vienna, Austria (2006).

European Nuclear Society

Rue Belliard 65
1040 Brussels
Belgium
Telephone +32 2 505 30 54
Fax + 32 2 502 39 02

topsafe2008@euronuclear.org

www.euronuclear.org