PIUS: Status and perspectives

A review of the design and development of reactors based on the principle of "process inherent ultimate safety"

by Tor Pedersen

Nuclear energy represents an important option for future energy supply, but the nuclear option could be jeopardized if severe reactor accidents, such as the ones at Three Mile Island or Chernobyl, were to continue to occur anywhere in the world.

The PIUS (Process Inherent Ultimate Safety) reactor development effort is based on the conviction that future developments of nuclear energy for civilian applications should incorporate technologies where safety against severe accidents becomes an inherent feature of the reactor configuration that cannot be compromised by malfunctioning equipment or human intervention. This requirement is being recognized in several industrialized countries where an evaluation of the PIUS concept is underway or planned and should be particularly important in countries contemplating the introduction of nuclear power.

The PIUS principle, simply stated, aims for the preservation of core integrity under all reasonably conceivable upset conditions by utilizing reactor configurations in which only processes based entirely on irrefutable natural laws (such as gravity and thermo-hydraulics) are necessary for reactor shutdown and cooling. Maintaining core integrity guarantees the absence of significant releases of radioactivity to the environment and hence results in an insignificant safety risk.

Plant design

Several reactor designs have evolved in more than a decade of development work that has been performed by ABB Atom on the design and development of PIUS type reactors. All of the designs have been based on well-established light-water reactor (LWR) technology and infrastructure with a rearrangement of the configuration and the incorporation of some special components to implement the PIUS principle. For the generation of electric power, one of the early designs was a 1600-MWth, 500-MWe, plant with one reactor core and four steam generator/coolant pump loops all installed inside a large concrete vessel. Several other designs, involving the modular concept of coupling single reactor cores with an associated steam generator/coolant loop into plants of various outputs, followed. In 1987, design studies were initiated jointly with international contributions on a version of the PIUS reactor using steam generators and reactor coolant pumps mounted external to the concrete vessel containing the reactor core. This configuration has been selected as the standard design. The thermal power output of the reactor is 2000 MW and the nominal net electric power output of the plant is 640 MWe. *(See accompanying table and figure.)*

The reactor core is a pressurized-water reactor (PWR) type core made up of 213 fuel assemblies with standard PWR fuel rod diameter and a reduced height. The core is located at the bottom of the reactor pool which is a high-boron content water mass enclosed by a pre-stressed concrete reactor vessel (PCRV). The reactor core does not use control rods, neither for reactor shutdown nor for power shaping. The reactivity control is accomplished by means of reactor coolant boron concentration and temperature.

Mr Pedersen is an electrical engineer in the Reactor Division of ABB Atom, Västerås, Sweden.

PIUS-600 key data

Core thermal power	2000 MW
Net electric power	640 MWe
Circulation water temperature	18°C
No. of fuel assemblies	213
Core height (active)	2.50 m
Core equivalent diameter	3.76 m
Average fuel linear heat rate	11.9 kW/m
Average core power density	72.3 kW/1
Core inlet temperature	260°C
Core outlet temperature	290°C
Operating pressure	9 MPa
Core flow	13 000 kg/s
Average burnup	45 000 MWd/t
Concrete vessel cavity diameter	12 m
Concrete vessel cavity volume	3300 m^3
Concrete vessel total height	43 m
Concrete vessel thickness	7 m
Number of steam generators and coolant pumps	4

The core data are significantly relaxed in comparison with current PWR practice in terms of average linear heat load, temperatures, flow rates, and associated pressure drops. The reactivity compensation for burnup is accomplished by means of a burnable absorber (gadolinium) in some of the fuel rods, and the moderator temperature reactivity coefficient is strongly negative throughout the operating cycle.

From the core, the heated coolant at a temperature of 290°C passes up through the riser pipe and leaves the reactor vessel through nozzles on the sides of the upper plenum. The coolant continues in the hot leg coolant pipes to the four straight tube once-through steam generators mounted beside the PCRV. The main coolant pumps are located below, and structurally integrated with, the steam generators. The pumps are sized-up versions of the glandless, wet motor design pumps that have been utilized as recirculation pumps in the ABB Atom boiling-water reactor plants.

The cold leg piping enters the reactor vessel through nozzles in the upper plenum at the same level as the hot leg nozzles, and the 260°C return flow is directed downwards to the reactor core inlet via a downcomer. On its way down the flow is accelerated, and there are open connections between the downcomer and the pressurizer providing a siphon breaker arrangement. The siphon breaker prevents siphoning-off the reactor pool water inventory in the hypothetical event of a cold leg rupture. There are also some open connections between the downcomer and the riser. At the bottom of the downcomer the return flow enters the reactor core inlet plenum.

A one-meter diameter pipe which is open to the enclosing reactor pool is located below the core inlet plenum. A tube bundle arrangement inside this pipe minimizes water mixing and ensures stable layering of hot reactor loop water on top of the colder reactor pool water. This pipe, with the bundle arrangement and the stratified water, is called the lower "density lock" and is one of the special components required to implement the PIUS principle. The position of the interface between hot and cold water is determined by temperature measurements, and this information is used for controlling the speed and hence the flow rate of the main coolant pumps to maintain the interface level at a constant position during normal operation. There is another density lock arrangement at a high location in the pool, connected to the upper riser plenum. This upper density lock has a similar arrangement of tube bundles and a number of small openings between the riser and this density lock.

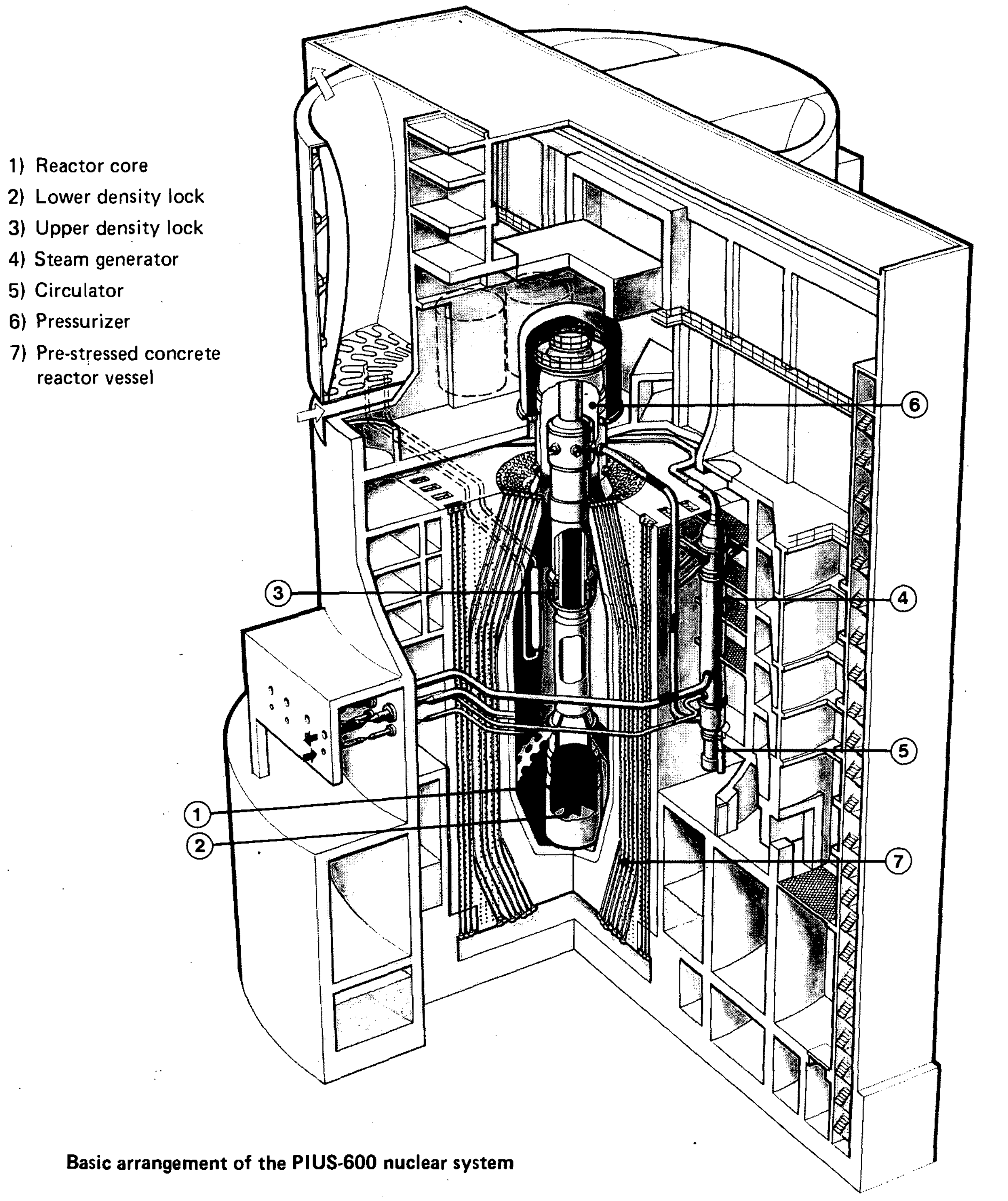
This reactor system configuration, with the two always open density locks connected to the high boron content pool, provides the basis for assuring the PIUS principle. There is always an open natural circulation path from the pool through the lower density lock, to the core via inlet pipes, through the core itself, the riser, the passage from the upper riser plenum to the upper density lock back to the pool. During normal plant operation this natural circulation circuit is kept inactive by means of the speed control of the main coolant pumps maintaining the hot/cold interface in the lower density lock and, in combination with primary loop water volume control, maintaining the hot/cold interface level in the upper density lock. The temperature measurements for the interface level in the upper lock are used for primary loop volume control purposes. The coolant flow rate is determined by the thermal conditions at the reactor core outlet relative to the reactor pool. The resulting pressure drop across the core and up through the riser must correspond to the static pressure difference between the interface levels in the upper and lower density locks. The main coolant pumps are operated to establish a pressure balance in the density locks during normal steady-state and load-following operations. A sudden collapse in this pressure balance, as would occur during a severe transient or accident, would result in the natural circulation of borated pool water through the core, providing both reactor shutdown and continued core cooling. The upper portion of both density locks, i.e. the volume above the hot/cold water interface, is normally filled with hot primary loop water, serving as a buffer volume to prevent ingress of pool water and spurious reactor shutdowns during minor operational disturbances.

The borated water in the reactor pool is cooled by two systems — one with forced circulation of pool water through heat exchangers and pumps outside the reactor vessel, and one passive system utilizing coolers in the reactor pool and natural circulation loops up to dry, natural draft cooling towers located on the top of the reactor building, one at each corner. The natural circulation system ensures the cooling of the reactor pool in accident situations and prevents boiling of the pool water even with one circuit out of operation. If all pool cooling systems were to fail, the pool water ensures core cooling for one week.

The PCRV has a cavity with a diameter of about 12 metres and a depth of about 38 metres, containing some 3300 m^3 of water. The concrete vessel monolith has a cross-section of about 27 metres and a height of about 43 metres. It is anchored to the foundation mat structure by means of pre-stressing tendons. The pressure retaining capability of the vessel is ensured by a large number of pre-stressing tendons, partly horizontal tendons around the cavity and partly vertical tendons from the top to the bottom, and by reinforcement bars.

The inside of the cavity is provided with a stainless steel liner. In addition, there is a second barrier, an embedded steel membrane about one metre into the concrete, up a level above the upper density lock to ensure that the reactor pool water volume below this level cannot be lost by liner leakage. Concrete vessel penetrations are not permitted below this level.

On top of the pre-stressed concrete vessel there is a steel vessel extension which is fixed by means of separate tendons anchored to the bottom of the concrete vessel. This extension contains the pipe nozzles for the hot and cold leg pipes, for the forced circulation loops



Basic arrangement of the PIUS-600 nuclear system

of the reactor pool cooling system, and for some other system pipes and also encloses the upper riser plenum, and the pressurizer.

The PCRV and the reactor system are enclosed in a large containment structure of the pressure suppression type. Blowdown pipes lead from the drywell into a large condensation pool in the wetwell. All equipment containing reactor loop or reactor pool water at high pressure and high temperature is located inside the containment, which is designed to withstand a double-ended break of the largest pipe. The structure is made of reinforced concrete with a strength capable of resisting the impact of a crashing aircraft. The whole containment is provided with a steel liner in order to ensure leak tightness.

The reactor power is controlled by the boron content and temperature of the reactor coolant. During normal plant operation the reactor power can be controlled without adjustment of the boron content in the reactor coolant, by utilizing the strongly negative coolant temperature reactivity coefficient. A power change is accomplished by simply adjusting the rate of feedwater flow to the steam generators. An increase in feedwater flow rate results in a reduced temperature of the return primary coolant flow to the reactor and thus an increase in reactor power. This procedure can be applied over a ±40% power range with a 20% per minute rate of change in plant power. Daily load-following operation, for example between 100% and 50%, can be accomplished with only a minor adjustment of the boron concentration at the start of the first day's cycle. For subsequent days no additional adjustment is needed. Beyond this range, adjustment of the boron content is needed in order to keep the reactor core coolant outlet temperature within acceptable limits. The boron content is controlled by injecting distilled water for power increase, or high boron content water for power decrease, and withdrawing a similar amount of water, corresponding to the procedures in normal PWR plants.

The plant is also provided with instrumentation systems, protection, logic, and actuation systems for reactor shutdown, residual heat removal, containment isolation, etc. in a similar way as current LWR plants. However, their importance for ensuring safety is significantly reduced. The equipment of these instrumentation, monitoring, protection, and actuation systems is separated from that of other systems and located in separate, physically well-protected compartments at the bottom of the reactor building. The reactor protection system, with a two-out-of-four coincidence logic, has the task of initiating power level reduction, reactor shutdown or reactor scram when reactor process parameters exceed set limits.

In most cases a runback to a lower power level by means of the feedwater flow control, or a further reduction to hot standby or hot shutdown conditions by injecting high boron content water into the primary system coolant, will be an adequate countermeasure. A reactor scram is initiated only in a few situations by tripping one of the main coolant pumps. Pool water will then enter the primary system and shut down the reactor to hot, subcritical conditions. The primary loop structures will be subjected to a rapid cooling down by some 50–60 degrees Celsius, but this transient has no critical effect as regards thermal fatigue of the components.

Compared with current commercial LWR designs a number of safety-grade systems have been eliminated: the control rods and the safety injection boron system are replaced by the density locks; the automatic depressurization system is not required; the auxiliary feedwater supply system for residual heat removal is replaced by the reactor pool; the containment heat removal and containment spray systems are replaced by the passive cooling of the reactor pool. The safety-grade closed cooling water system, the heating, ventilation and control systems, and the AC power supply systems have been replaced by non safety-grade systems, allowing major simplification. The remaining safety-grade functions are performed by: the reactor protection system tripping one coolant pump to achieve a reactor scram; the containment isolation system isolating the containment by closing isolation valves; the reactor vessel safety valves activated by pressure differentials; and the passive reactor pool cooling function. Only the last function is needed for the protection of the core, however, and only beyond the minimum one week grace period.

As a result, the plant should be simpler to operate and maintain than current LWR plants, and the elimination of severe reactor accidents as a practical concern should also facilitate simplified operation.

Response to severe transients

The behaviour of the various plant designs incorporating the PIUS principle in severe transients and accident situations has been analysed extensively over the years, partly by experiments but mostly by computer simulations. For the latter purpose, a very efficient computer code has been developed especially for the simulation of the dynamic behaviour of this type of plant. The capability of the code to simulate the reactor and plant behaviour with sufficient accuracy has been checked, by performing experiments in a test rig and predicting the outcome by calculations.

A large number of transients and accident situations have been analysed and the outcome has always been a reactor shutdown or continued operation at a safe, limited power level. No accident sequence leading to uncovering the core and/or departure from nucleate boiling (DNB) has been identified.

In terms of requirements on the containment, there are some very significant differences between a plant configured according to the PIUS principle and current LWR plants. In the former, the integrity of the nuclear fuel is protected by self-activated passive functions, the reactor core will not be uncovered nor the fuel overheated following any credible incident. The radioactive matter released to the containment following pipe breaks will arise from the possible leaking fuel rods in the core prior to the accident; the accidents will not cause additional fuel damage. Only part of the hot primary loop water inventory will be released to the containment during the depressurization of the reactor.

Following the initial depressurization of the reactor system, the core will be cooled by the natural circulation of reactor pool loop, transferring the reactor residual heat to the reactor pool. The natural circulation pool cooling system always ensures residual heat removal to the air via dry natural draft cooling towers. There will be no boiling of the water inside the reactor vessel.

The short-term release of steam and hot water during a blowdown of the reactor is absorbed in the condensation pool. The containment will be pressurized to 2 bar. There is no long-term release of steam, and the pressure will decrease to slightly above atmospheric pressure within 2–3 hours due to condensation of steam on structures and components. No safety-grade containment cooling systems are needed.

The releases to the containment of radioactive matter will be small, and as a result of the moderate, temporary pressurization of the containment relative to the atmosphere, the maximum credible release rates to the environment will be very small.

The calculated doses at the site boundary are well below the lower level Protective Action Guidelines (PAGs) of the US Environmental Protection Agency (EPA), which stipulate maximums of 1 rem whole-body and 5 rem thyroid doses. This should provide the basis for easing the offsite emergency planning requirements.

Plant construction

Construction of a plant involves a few major items that are important for the critical path; the PCRV together with the containment part of the reactor building, and the reactor pressure vessel steel extension. The construction activities have been analysed by the team of civil engineering, installation, and commissioning supervisory personnel that built and commissioned the Oskarshamn-3 nuclear power plant in Sweden in 57 months from the first pouring of concrete to the start of commercial operation. The containment for this plant is similar to that in Oskarshamm 3, and the PCRV can utilize the same construction techniques. Hence the planning team was drawing on their own experience when establishing the schedule for the plant construction.

The resulting schedule indicates a total construction time of 45 months for the plant, from pouring of the first concrete to the start of commercial operation, or 39 months to fuel loading. This time is probably conservative since possible reductions as a result of onsite or offsite prefabrication (or modularization) have not been taken into account.

Detailed turnkey cost estimates have been made for an nth-of-a-kind plant and for a conventional ABB Atom advanced BWR plant (700 MWe), manufactured and constructed under Scandinavian conditions. These cost estimates show a significant advantage for the PIUS-600 over the BWR 700 MWe in overnight cost per net kWe output.

The lower steam pressure and temperature of the plant do result in a lower thermal efficiency and thus a somewhat higher fuel cycle cost than for the BWR. On the other hand, the construction time is shorter and the personnel costs are anticipated to be lower due to the plant's simplicity. As a result the generating costs are estimated to be lower than for the 700-MWe BWR plant.

Future outlook

ABB Atom has been working on reactors incorporating the PIUS principle for more than a decade with considerable detailed design and analyses performed for a number of design versions. Based on these activities and inputs from utilities and others, a very promising design concept has evolved in which demonstrated component technology is utilized to the maximum extent. The novel features have been sufficiently studied to eliminate concerns regarding the technical feasibility and practicability of the concept, especially from the point-of-view of safety and operability. Some additional testing is planned, however, to provide information and data to support the detailed design and arrangement for a commercial plant.

PIUS has been discussed with utilities from different countries and, at this time, an evaluation and assessment of the practical feasibility and a joint design study for adaptation to Italian conditions is being contemplated. A feasibility study is being performed in the People's Republic of China, and evaluation studies are under way in other countries. Efforts are also being made to establish a joint venture in the USA for marketing the design and having it reviewed and licensed by the US Nuclear Regulatory Commission. In the USA, an evaluation study of the reactor concept was performed in 1985. An indirect result of that study may be a teaming arrangement to investigate adapting the design to US conditions and for further activities related to marketing in the USA.

Reactors incorporating the PIUS principle represent a major evolution in nuclear safety, while maintaining the application of established and proven LWR technology with a limited need for testing for final design verification. The cost estimates and the construction schedule indicate that competitiveness with other energy sources can be achieved.