# THE SECURE REACTORS

Kåre Hannerz, AB ASEA-ATOM
Box 53
S 721 04 VÄSTERÅS, Sweden
Telex 40629 atomva s

ABSTRACT

The principle of Process Inherent Ultimate Safety (PIUS) is a new approach to Light Water Reactor (LWR) safety that could represent a solution to the present problems of public distrust, regulatory maze and plant design complexity plaguing the nuclear industry in many countries.

A unique thermohydraulic design of the primary system ensures core integrity, and thereby guarantees freedom from significant releases of radioactive matter, in all credible emergencies. This is accomplished entirely without reliance on potentially failure prone engineered safety systems and with immunity to operator mistakes. The potential for human fallibility to cause accidents is thereby drastically reduced in an easily understood way. Plant design can be greatly simplified because redundant, diverse safety systems are no longer needed.

The paper briefly describes the PIUS design principle and the two SECURE reactor designs based on it, i.e. SECURE-H for district heating and process steam and SECURE-P (usually known simply as PIUS) for electric power generation.

Demonstration of simulated system over-all thermohydraulic function and transient response in a large electrically heated test loop is described and results from some component development work is given.

## INTRODUCTION

As recent experience has demonstrated nuclear power, improperly handled, can be an hazardous technology. Release of a significant fraction of the radionuclide inventory in a large power reactor following a core degradation accident can have uniquely harmful consequences. In view of the deep-rooted public fear of radiation there must be correspondingly uniquely strong guarantees that accidents leading to such release will not occur.

The problem with the present Light Water Reactor (LWR) technology is that it is widely perceived as not providing sufficiently convincing such guarantees. In many countries the reactor safety issue therefore now represents the principal impediment to the future rational use of the nuclear option.

It appears unlikely that mere administrative reforms and the passage of time will remedy the situation. A new, more convincing approach to reactor safety is probably needed to break the impasse.

This paper briefly describes how existing LWR technology can be adapted and modified to provide the requisite safety assurance in a simple, easily comprehended way and to eliminate the need for complex plant design originating in safety considerations.

## DESIGN ASSUMPTIONS

All technological risks can ultimately be traced back, directly or indirectly, to human fallibility or malevolence. The risks associated with a technology may be considered proportional to the extent to which its safe use demands human excellence and precision, or inversely proportional to its "forgivingness", i.e. tolerance to mistakes.

Chernobyl, TMI and some other incidents have shown that there is almost no limit to the extent that human negligence and confusion can sometimes interfere with reactor operation even now when nuclear plants are still run by an elite corps. Their occasional presence in the fabrication, construction and maintenance phases of plant life must be assumed to be equally unavoidable.

Quality assurance programs, operator training etc can reduce their impact, hopefully to a low level, but claims that it can be eliminated always and everywhere have no credibility.

In all probability it is the universal awareness of ubiquitous human fallibility and the widely held perception that a reactor of todays design may eventually become a victim of it that is at the root of public distrust. The latter, in turn, has become the basis for safety based regulation that is the reason for the complexity and high costs that have befallen the technology.

Accordingly, a probable requisite for turning around the attitude to nuclear power and to make possible a much-needed simplification of the technology is a reactor design where, for easily understood reasons, the susceptibility of safety to human fallibility is greatly reduced. A clearly perceptible quantum jump is needed, not just incremental improvements.

In other words, there shall be no credible paths to core degradation accidents in spite of undisputably pessimistic assumptions regarding the extent to which human negligence, carelessness or confusion have entered the manufacturing, construction, operation and maintenance phases of a reactor.

In line with the above, the assumptions underlying the design of the SECURE reactors are as follows:

1. In emergencies plant operators are assumed to be completely confused and to make any mistakes possible with the controls at their disposal.

2. No credit is given for function of active equipment (such as pumps, valves etc) in emergencies.

3. It is assumed that any load carrying structural member can fail at any time. However, unrelated, simultaneous structural failures can be disregarded as having a vanishingly small probability.

4. It is assumed that the plant will be subject to deliberate destructive intervention, by outside terrorists, military attack with off-the-shelf nonnuclear weapons or by insiders.

5. It may be postulated that, after a post accident "grace period", outside intervention to ensure continued core cooling can be relied upon. A "grace period" of one week has been assumed in all design work to date.

It is obvious that assumption 4 must be subject to some qualification since, given sufficient time and resources, it will always be possible to achieve an extensive radionuclide dispersion from a large power reactor core. However, further discussion of this is beyond the scope of this paper.

## DESIGN PRINCIPLES

The above-mentioned assumptions strongly restrict the choice of design solutions.

The reactor system design and heat extraction process must clearly be such that, following any credible incident or failure, whether it involves equipment break-downs, operator mistakes or intentional destructive acts, the system reverts by itself, without reliance on acts of humans or on activation of equipment, to a state with assured passive long term core cooling. Active control measures are needed to keep the systems in operation but they must always by overruled by ever-present natural forces under abnormal conditions before core damage can occur.

This principle for designing for safety has been given the acronym PIUS (for Process Inherent Ultimate Safety). The power reactor SECURE-P designed according to this principle has become internationally known simply as PIUS.

The implementation of the PIUS design principle has been previously described in the literature[1,2] and will therefore be only briefly reviewed here.

The fullfillment of two simple conditions is sufficient to prevent core degradation accidents in an LWR:

a. The core is to be kept submerged in water at all times.

b. The heat generation rate of the core must not exceed the cooling capability of the submerging water, i.e. there shall be no dry-out (other than of very short duration).

According to assumption 2 above no pumps, valves etc may be relied upon for supply of water to the core, and failure of any single load bearing structural member must be assumed.

The only practical way of complying with condition a. is therefore to place the core in a pool of water, which is kept in place by a multibarrier prestressed concrete vessel, the integrity of which is ensured by a large number of independent tendons.

The water volume must be sufficient to remove, by its evaporation, decay heat for the whole of the "grace period".

The water must contain a neutron poison (boric acid) to keep the reactor subcritical since, again according to assumption 2, control rod insertion cannot be relied upor for that.

It should be pointed out that arrangements like cooling towers for an indefinite period of core cooling are not a solution since because of their size they could not in practice be made resistant to outside attack. The same is true for a conventional containment vessel.

A core sitting at the bottom of a pool with boric acid containing water is an arrangement of no interest as such. A circuit for extracting useful heat from it, with requirement b. above fulfilled, must be introduced. However, this must be done without in any way blocking it off from the pool by valves etc.

How this is achieved is most easily understood from figures 1A-1D.

In figure 1A a heat source (reactor core) is placed near the lower end of a pipe (riser) in a pool of water. The heat evolution will cause a natural circulation flow through the pipe as shown.

In figure 1B the flow is returned by means of a pump to the inlet of the pipe instead of emerging into the pool. The heat generated by the core now remains in the circulating water.

In figure 1C a heat exchanger has been added to the circulating system to keep its temperature constant. The heat generated by the core is now extracted for a useful purpose. The upper part of the pipe is bent downwards to permit stable layering of hot water above cold at both ends in so-called density locks. The pump and heat exchanger can be located either in the pool or outside the concrete vessel.

Finally, in figure 1D a pressurizer has been added so that heat extraction can occur at elevated temperature e.g. for generating steam.

By keeping the pump flow within a relatively narrow control range, determined by permissible level fluctuations in the density locks, the hot circulating coolant can be kept separated from the cold water with high boron concentration in the pool.

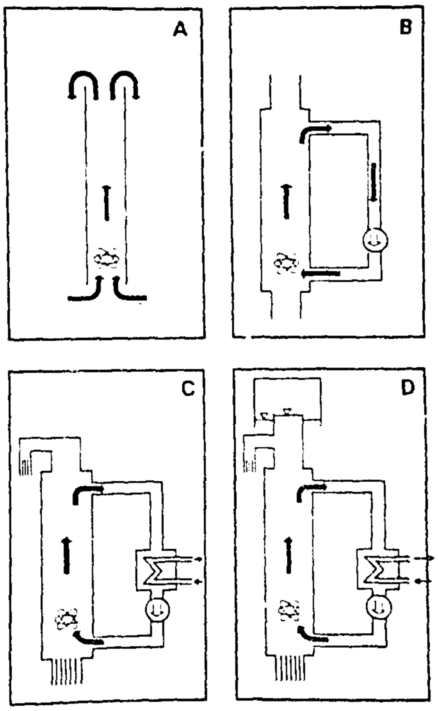


Figure 1

The operating principle of the PIUS primary system

Reactivity and power output can be controlled by means of changes in boron content and coolant temperature while there is always an open natural circulation path through the core and the pool.

It turns out that the arrangement just described fullfills condition b. mentioned above. Keeping the system in operation requires a rather sensitive flow control (which however offers no problem during normal operating conditions). In major transients of potential safety importance the control system can no longer cope with the forces imposed on the coolant by the laws of thermohydraulics and gravity and borated pool water enters the coolant system through the density locks and causes reactor shut down or stabilization of core heat output at a safe level.

As an example, upon loss of the secondary system as heat sink the primary system heats up, and, depending upon the choice of design parameters, void may appear in the riser. This causes increased "buoyancy" of the coolant in the riser and an increase in core flow beyond the upper limit of the delivery capacity of the pump. The deficiency is taken through the lower density lock whereby boron is added to the recirculating coolant and reactor shut down occurs.

## SAFETY PERFORMANCE

Verification that the reactor designs produced under the SECURE-PIUS program are indeed safe under the design assumptions mentioned above has been a centerpiece of the work.

The analysis is not yet complete and final verification has to await the completion of a Final Safety Analysis Report. However, there is every indication so far of satisfactory behavior.

Before describing the designs their general safety performance, which is common to them is briefly reviewed here.

A special program, for system dynamic simulation, RIGEL, was developed to make possible the study of a large number of design configurations in many transients. Its principal use has been analysis of various severe transients and failure situations, under the design assumption that no protective measures, such as scram, initiated by operators or automatic safety equipment are credited. Some of this work has been published[2]. Examples of severe transients studied with these assumptions are the following:

- Main recirculation pump trip (this causes reactor shut down in a few seconds)
- Loss of secondary system heat sink
- Large secondary steam line break
- Large primary system leaks (depressurization)
- Continuous inadvertent reactivity insertion by boron dilution (corresponds to a control rod withdrawal accident in a conventional reactor)
- Multiple steam generator tube breaks

In all cases the final outcome is the same: The core remains unharmed, either shut down or at a safe power level and the water inventory for post incident cooling for the "grace period" always remains at its disposal. The safety is a consequence of the laws of gravity and thermohydraulics acting on the coolant rather than of the activation of engineered safety systems.

## THE HEAT PRODUCING REACTOR SECURE-H

This reactor is designed primarily to supply heat to district heating systems and to process industries using low temperature heat. A 400 MWth version is now offered by ASEA-ATOM on normal commercial terms. A few principal data are given in Table I.

TABLE I

Principal data for the SECURE-H reactor

| | |
|---|---|
| Thermal power | 400 MW |
| Core outlet temperature | 190°C |
| Core power density in fuel | 15 W/gU |
| Active core height | 1.845 m |
| No of fuel assemblies | 308 |
| No of fuel rods per fuel assembly | 60 |
| Primary system operating pressure | 2.0 MPa |
| Fuel enrichment (equilibrium) | 2.5% |
| Fuel burn up | 29000 MWd/t |

According to the design principles earlier explained the reactor and the riser and density locks are placed in a prestressed concrete pressure vessel filled with boric acid containing water.

The heat exchangers and pumps are located outside the vessel and are directly accessible for service. A siphon breaker arrangement is used to prevent loss of pool water after a postulated break in the outer part of the primary circuit.

An orifice in the main outlet pipe limits break flow and assists in terminating transients involving overtemperature in the coolant by the hydraulic losses due to two-phase flow in the high velocity section. This will decrease coolant flow through the core and help cause boron ingress through the lower density lock and reactor shut down or power decrease.

Reactor power is controlled by means of boric acid concentration in the coolant; there are no control rods. The resulting limitation in rate of power change (1-2% per minute) is of no consequence for a heating reactor.

There is an intermediate circuit between the reactor water and the water in the district heating system. This contains boric acid containing water like the reactor pool and is operated at a higher pressure than the latter. Leaks in the primary heat exchanger will therefore not lead to loss of primary coolant.

This reactor was offered to the city of Helsinki where it was to produce water of 150°C for the municipal district heating grid. In an economic comparison with other alternatives, including coal fired and nuclear fired cogeneration of electricy and heat, it came out on top. However, for political reasons the nuclear option was ruled out for the time being.

The licensability of the SECURE-H reactor has been informally investigated by the Finnish Nuclear Safety Inspectorate and has been reviewed by the Gesellschaft für Reactor Sicherheit (GRS) in Germany. In both cases the outcome was favorable with only minor adjustments to national requirements asked for.

SECURE-H is under active consideration for use in among others People's Republic of China.

Application in the country of origin, Sweden, is presently prevented by laws against construction of new nuclear plants.

THE POWER REACTOR SECURE-P (PIUS)

Several versions of the PIUS reactor, as it has become known internationally, have been studied and some of them have been described in the literature[1,3].

Most of the early work was on designs where the whole primary system is located inside the prestressed concrete vessel so that steam is sent directly from the latter to the turbine.

A detailed design study of a 600 MWe unit incorporating this arrangement that was extensively documented and costed is described in reference 3.

An arrangement with everything inside the vessel has many attractions, above all from the point of view of protection from external events. However, it necessitates the introduction of a fairly large number of mechanical components of a new and unique design (the most important of which is the steam generator).

A program for implementation of the design would therefore be burdened with relatively extensive and time consuming component testing work.

Following recent events there are indications that the need for nuclear power plants with a new level of inherent and transparent safety is more imminent than previously foreseen. For this reason current work on the PIUS reactor aims at a design where the need for component development is strongly reduced. This necessitates the placement of the steam generators and recirculation pumps outside the prestressed concrete vessel, as is the case with the SECURE-H reactor. It should be pointed out that this is in no way incompatible with the basic PIUS safety characteristics.

This new design uses a single 2000 MWth core centrally located near the hemispherical bottom of a large bottle-shaped prestressed concrete vessel.

The primary coolant leaving the core flows vertically upwards through the riser pipe and leaves the vessel from a centrally located smaller diameter steel extension at the top of the concrete vessel. Once-through steam generators, similar to those used in some US PWRs, are located outside the concrete vessel with wet motor recirculation pumps at their lower end.

The always open natural circulation path through the core and pool branches out from the top of the riser to the upper density lock.

The riser-downcomer consists of an upper and a lower part which are removed one at a time from the concrete vessel for refueling whereupon the core becomes directly accessible from above. Spent fuel for 20-30 years of operation can be stored inside the vessel.

Space allocated to this paper does not permit a detailed description. As an example the reactor proper is shown in figure 2.

A few pertinent data for a 2000 MWth SECURE-P (PIUS) reactor are given in table II

TABLE II

Main data for the SECURE-P (PIUS) reactor

| | |
|---|---|
| Core thermal power output | 2000 MW |
| Net electrical output (Scandinavian Conditions) | 625 MW |
| Core inlet temperature | 261°C |
| Core outlet temperature | 293°C |
| Active core height | 1.97 m |
| Core diameter (equivalent) | 4.03 m |
| Fuel pin diameter | 12.25 mm |
| No of 16x16 rod fuel assemblies | 213 |
| No of control rods | 0 |
| Primary system operating pressure | 9.0 MPa |
| Secondary steam pressure | 4.0 MPa |
| Concrete vessel internal diameter | 13.4 m |

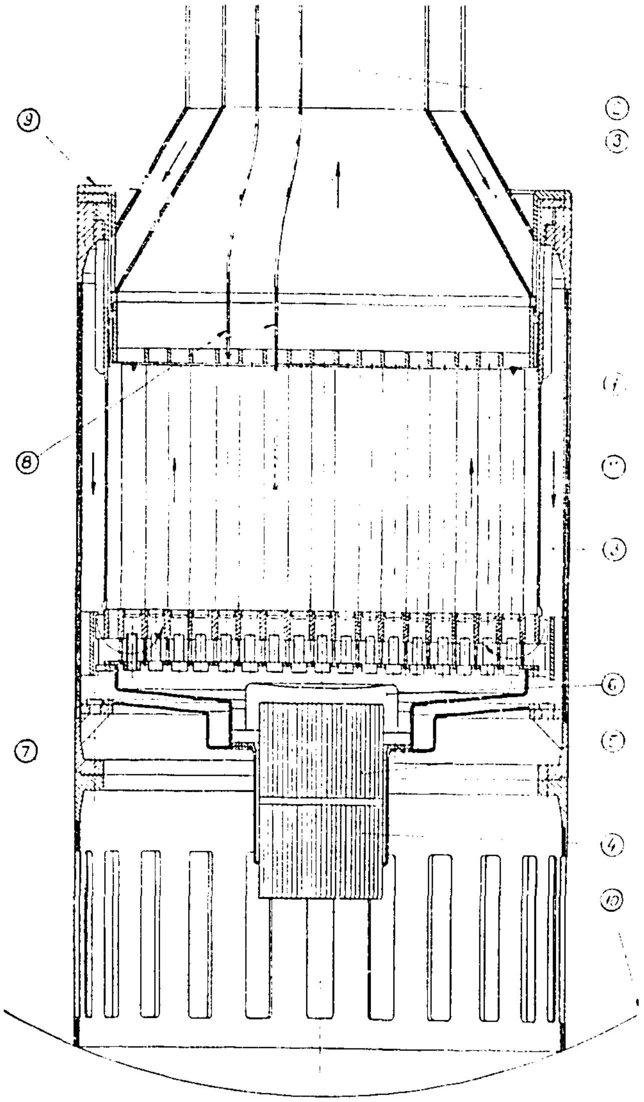


Figure 2

2000 MWth SECURE-P (PIUS) reactor

1. Core
2. Riser
3. Downcomer
4. Lower density lock
5. Buffer region
6. Hood for gas lock for start up
7. Pool water inlets to fuel assemblies
8. Core instrument thimbles
9. Main flange
10. Concrete vessel bottom liner
11. Pool

As with a conventional PWR burn up compensation and slow power changes are accomplished by means of chemical shim. However, rapid load following is made by means of temperature changes in the coolant, utilizing a strongly negative moderator temperature reactivity coefficient. The latter is obtained by suitable deployment of gadolinia in the core. As an example, just by increasing feedwater flow and opening the turbine throttle a power increase from 50% to 80% of full load can be made in less than a minute.

Critics of the PIUS concept often point to inadvertent boron ingress through the density locks as a potential problem with plant availability. There is no indication that this will be the case;on the contrary analysis shows that severe disturbances such as grid short circuits can be tolerated and extensive test loop operation has never experienced problems of this origin.

With all primary system components except the reactor vessel itself either out of the concrete vessel or removable from it and readily accessible in connection with refueling there should be few problems with maintenance.

The generating costs with the design initially mentioned with the primary circuit wholly inside the concrete vessel was estimated in fairly great detail. The results was lower costs than from a conventional LWR in the 600-700 MWe range, with the margin estimated to increase with decreasing plant capacity. The design with external steam generators has not yet been costed.

A valid question is how competitive costs can be achieved with a primary system that is clearly more expensive than that of a conventional PWR, and with some penalty in thermal efficiency due to the lower operating pressure in the concrete vessel.

The answer is found in the simplicity in the plant design outside the concrete vessel, which is basically due to the fact that it can be obliterated without risk of core meltdown. Eliminated need for redundant, diverse and spatially separated safety systems permits enormous savings. A good illustration to this is that the number of separate rooms in the plant has been reduced by a factor of four in comparison to a "conventional" plant.

VERIFICATION PROGRAM

A major development effort has been made in order to verify the over-all thermo-hydraulic function of a SECURE-PIUS type primary system. In particular its dynamic behavior in safety related transients. In this way an internal check on the predictions of the RIGEL program has been possible.

A complete electrically heated mock-up of a SECURE-H system was built for this purpose. The electrical heater had geometry and rate of heat generation identical to one full scale 8 x 8 fuel rod assembly.

In addition to a heat source simulating the core, the test rig was provided with a riser/downcomer, a large diameter pipe simulating the pool, the two density locks, a recirculation pump with speed controlled by the hot/cold interface position in the lower density lock, a heat exchanger and a secondary system. Some data for the rig are given in table III.

TABLE III

Pertinent data for the SECURE test rig

| | |
|---|---|
| Maximum power of heat source ("core") | 2500 kW |
| Maximum electrical current (DC) through heat source | 40000 amp |
| Design pressure | 2.5 MPa |
| Normal outlet temperature from heat source | 190°C |
| Normal inlet temperature | 150°C |
| Total height of rig | 30 m |
| Number of measuring points (taking recordings every second) | 203 |

The coolant conditions up- and downstream of the "core" are monitored. Temperature and void content (if any, in transients) are directly measured. Boric acid is simulated by a strong electrolyte and conductivity measured. The variables are fed to an ASEA-MASTER process computer that controls the solid-state rectifiers supplying DC to the heat source. The output of heat to the coolant is thus made to vary as in a real reactor according to the equations of neutron kinetics and heat conductivity, making possible a total integral system simulation.

A large number of transients were run in the rig and a very satisfactory agreement with the predictions of the RIGEL code was obtained. A couple of typical cases are described below. They closely resemble the corresponding cases with a full scale plant.

Figure 3 shows the result of a total loss of secondary side heat sink without scram or any other safety system intervention. After some time, when the primary coolant has heated up and the recirculation pump speed has gone to the upper end of the control range in attempting to compensate for increased riser buoyancy, pool water ingress occurs and the "reactor" is shut down.

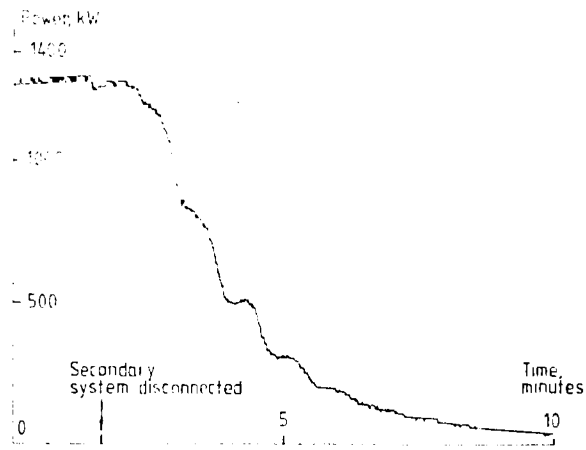


Figure 3

Loss of secondary system heat sink

Figure 4 illustrates the self-protective thermohydraulics of the primary system in the simulated case of continuous inadvertent reactivity insertion by boron dilution (injection of boron free water into the primary circuit). Initially the primary coolant heats up as in the previous case and at about 35% overpower (still no risk of dry-out) pool water ingress occurs, reducing the power. With continued supply of clean water the process is repeated in a cyclic manner while the reactor power remains at a safe level.

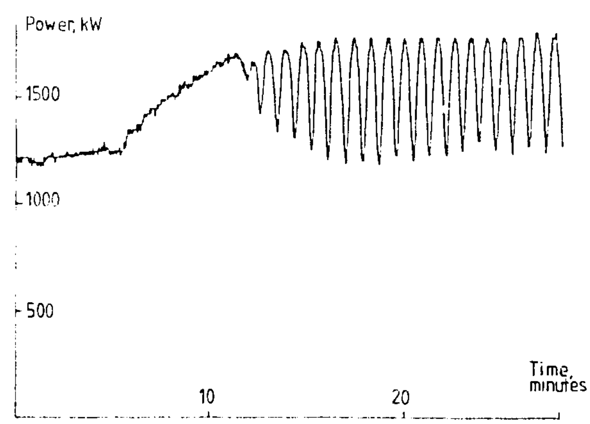


Figure 4

Continuous inadvertent reactivity insertion by boron dilution

As previously pointed out, location of the steam generating part of the primary system outside the concrete vessel strongly decreases the need for new component technology. The "new" components are essentially the density locks and submerged thermal insulation. The core data in terms of power density, flow rates, temperatures and pressures are well below those of recent commercial PWR practice.

An extensive testing program concerning the density locks has been carried out in order to establish the amount of transport, primarily of boric acid, from the pool into the primary system under the influence of turbulent disturbances in the temperature stratified region. This included high presure tests at fully representative conditions. It has by now been established that undesirable mixing across the density locks will not constitute a problem.

Submerged thermal insulation was extensively tested in connection with a Scandinavian development program for BWR reactor vessels of prestressed concrete in the 1970:s. Improved versions will also be tested.

The concrete vessel itself is not considered a developmental item. Extensive experience exists with the PIUS type double-barrier design since all eleven BWR containments for ASEA-ATOM BWRs are of this type. Application for high pressure was sufficiently studied in the development program just mentioned. Firm bids for construction of a PIUS concrete vessel can now be obtained and independent checks of the costs have been made.

The results of the development work constitute an adequate design basis for the SECURE-H reactor that is now commercially offered and the licensability of which has been checked in two countries.

For the more demanding application for power generation economic competitiveness requires that credit be given for the inherent safety endowed by the PIUS design so that the usual plethora of engineered safety systems no longer needed can be deleted.

To accept this break with their traditional approach licensing authorities may require further verification beyond that reported here, particularly of basic system response to safety related transients, with a real nuclear heat source.

This can be done in a small demonstration reactor - a "proof of principle device". Such a reactor has been designed by ASEA-ATOM but its construction in Sweden is presently not possible for reasons mentioned above.

## CONCLUDING REMARKS

A principal goal for mankind in the next century will be to provide an acceptable energy supply for a rapidly expanding world population without provoking a global environmental crisis. Availability of the nuclear option could prove to be of crucial importance for meeting this goal.

However, in view of the strong public reservations concerning nuclear energy use, this availability could be lost if there continues to occur, from time to time, serious accidents like those at TMI and Chernobyl.

The first commandment to the nuclear industry in all countries is therefore to achieve accident free operation of the present fleet of nuclear plants and those now under construction.

The second commandment is to provide, for the future, reactors that ensure continued accident free operation even with a vastly expanded number of nuclear plants spread all over the world, some of them inevitably operated under more adverse conditions than hitherto considered. To make this happen nothing less is likely to suffice than "built in" safety that cannot be corrupted by human fallibility or malevolence. With the SECURE reactors ASEA-ATOM has shown how one can go a long way towards meeting this goal by means of an adaption of established LWR technology.

1. T. PEDERSEN and C. SUNDQVIST, "PIUS, the Forgiving Reactor", Modern Power Systems, 5,68 (1985)

2. D. BABALA and K. HANNERZ, "Pressurized Water Reactor Inherent Core Protection by Primary System Thermohydraulics", Nucl. Sci. & Eng, 90,400 (1985)

3. K. HANNERZ, "The PIUS principle and the SECURE reactor concepts", Advances in Nucl. Sci. & Technology 19 (1987)