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ISIS, SAFETY AND ECONOMIC ASPECTS IN VIEW OF CO-GENERATION OF HEAT AND ELECTRICITY

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Abstract

ANSALDO has conceived a reactor called ISIS (Inherently Safe Immersed System), an innovative light water reactor with easily understandable safety characteristics.

The main targets are: passively safe behaviour, no pressurization of the Reactor Containment under any accident condition, control of plant capital cost and construction schedule by virtue of the modular concept and the compact layout.

The ISIS concept, described in general terms in the paper, builds up on the Density Lock concept originally proposed by ABB ATOM for the PIUS plant (ref. /1/), featuring innovative ideas derived from ANSALDO experience and based on proven technology from both LWR and LMR.

1. MAIN TARGETS OF THE ISIS CONCEPT

1.1 Safety targets

Significant progress has been made in the last years towards nuclear reactors that rely to the smallest possible extent on safety-related active systems, which, even using up-to-date technology, are felt by the public as prone-to-fail, no matter how low the frequency target for their loss is set.

The ISIS concept, under development in ANSALDO, largely embodies this progress. The main safety targets may be summarized as follows:

- No core melt-down and negligible release of radioactivity in any accident condition, by virtue of the reactor concept itself.
- Prompt reactor shut down occurring naturally after any abnormal condition.
- Reactor cooling in natural circulation for unlimited time.
- Self-depressurization of the reactor after a postulated failure of the pressure boundary.

1.2 Economic targets

The economic target aims at a viable industrial power plant based on specific overnight-capital cost and construction time competitive with those of the Light-Water Reactors under development. This is achievable by means of following features:

- Modular reactor.
- Integrated components (Compact layout).
- No pressurization to be taken into account in the design of the reactor containment.
- Primary system installation after reactor building completion.

2. BASIC CHARACTERISTICS OF THE ISIS CONCEPT

The **Inner Vessel** (fig. 1), which encloses the circulating, low boron concentration,pressurized hot water of the Primary System, is immersed in the highly borated pressurized cold water of the **Intermediate Plenum**. The Inner Vessel is provided with Wet Insulation to limit heat losses towards the Intermediate Plenum in normal operation.

The **Reactor Vessel**, which is the essential part of the pressure boundary, encloses the Intermediate Plenum and contains the Integrated Components of the Reactor Module.

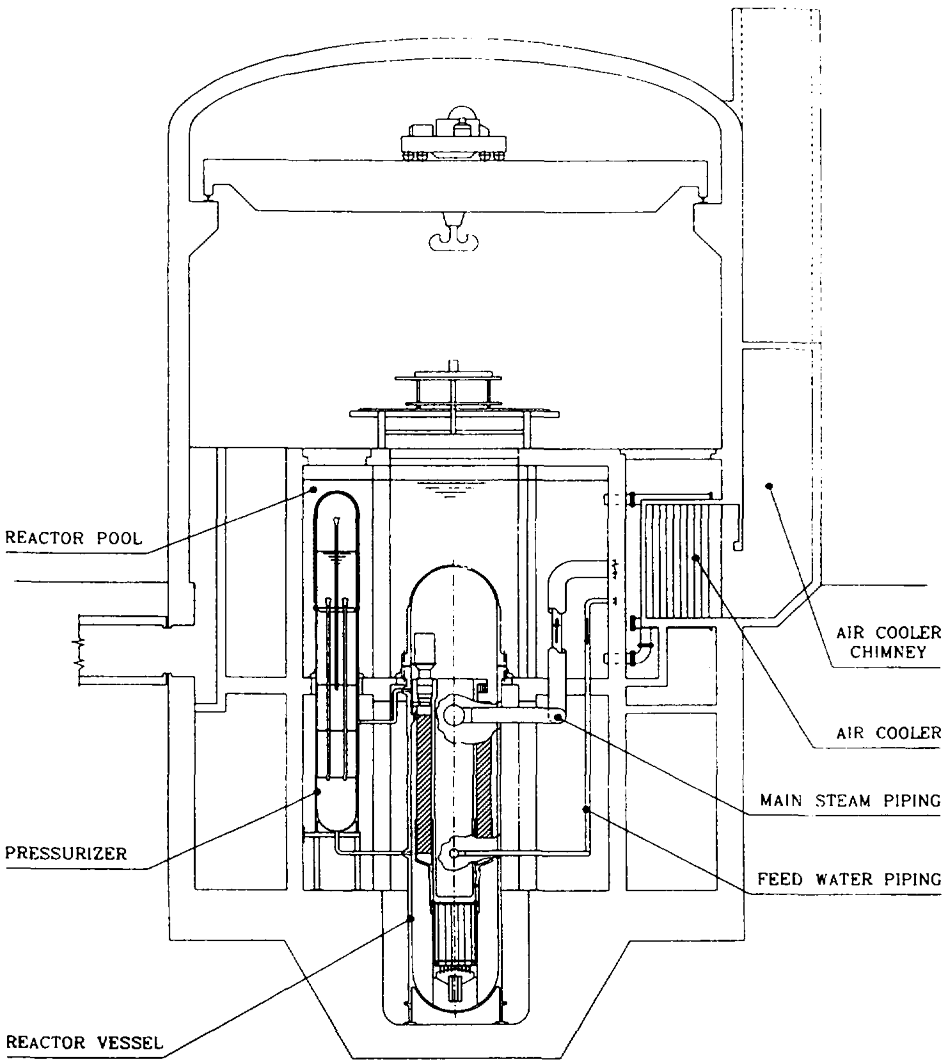

Fig. 1 - ISIS Reactor Building

- The Reactor Vessel is immersed in the cold borated water of a large **Reactor Pool** at atmospheric pressure. The Reactor Vessel is not insulated; this allows heat transfer to the surrouding water of the Reactor Pool under accident conditions.
- The **Pressurizer** upper portion performs the pressure control function; the lower portion contains cold water and provides additional heat transfer surface to the Reactor Pool under accident condition.

3. THE ISIS PRIMARY SYSTEM

The Primary System of the ISIS reactor is of the integrated type (fig. 2), with the Steam Generator Unit (SGU) housed in the Reactor Vessel, to which feedwater and steam piping are connected.

Within the Reactor Vessel, an Inner Vessel provided with wet metallic insulation separates the circulating low-boron primary water from the surrounding highly borated cold water.

Hot and cold plena are hydraulically connected at the bottom and at the top of the Inner Vessel by means of open-ended tube bundles, referred to in the following as Lower and Upper **Density Locks**. The Inner Vessel houses the Core, the Steam Generator Unit and the Primary Pumps.

Outstanding feature is the complete immersion of the Pressure Boundary, made up, for each module, of a Reactor Vessel and of a separated Pressurizer with interconnecting Pipe Ducts, in a large pool of cold water.

ISIS overall design Parameters

Thermal power	650	MWth
Net electric power	200	MWe
Core inlet temp.	271	°C
Core outlet temp.	310	°C
Operating pressure	14	MPa
Feedwater temp.	120	°C
Steam pressure	4,6	MPa
Steam outlet temp.	290	°C

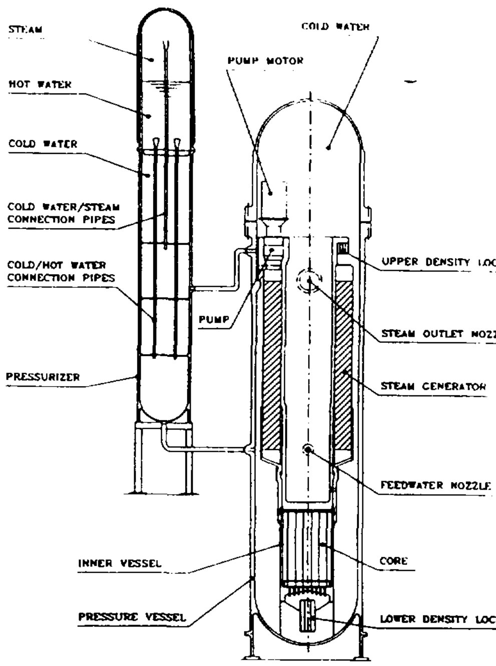

Fig. 2 - ISIS Reactor Module

During normal operation, the heat generated in the core is transferred to the SGU via the water circulated by the Primary Pumps, which are located at the top of the Inner Vessel. In case of unavailability of this heat transfer route, the cold and highly borated water of the Intermediate Plenum enters the Primary Circuit from the bottom, mixes up with the hot primary water, shuts down the reactor and cools the core in natural circulation. The same process, by heating the intermediate plenum water and the Pressure Boundary metal, activates the natural heat transfer route towards the Reactor Pool, which contains approximately 6.000 cubic meters of cold water.

The water inventory in the Reactor Pool is large enough to allow the water itself to remain below the boiling point after removal of the decay heat for about a week.

Cooling down of the plant pool is guaranteed, anyway, for an unlimited time, by virtue of two loops provided with water-air heat exchangers in natural circulation, sized to reject to the atmosphere, at steady state, approximately 2 MW and thereby capable to prevent the pool water from boiling.

Similarly to the PIUS reactor concept, the shut down and cooling functions of the core are carried out, in any condition, by the highly borated cold water of a plenum, which is hydraulically connected to the primary system by means of density locks.

However, unlike the PIUS, the intermediate plenum of ISIS contains a relatively small inventory of cold water (approximately 300 cubic meters per reactor module) at primary system pressure.

4. MAIN COMPONENTS

Reactor Vessel

The Reactor Vessel is of cylindrical shape with hemispherical heads.

The construction material is low-alloy carbon steel, internally lined with austenitic stainless steel.

The main openings of the Reactor Vessel are the water/steam nozzles and the two connections to the Pressurizer.

Core

The reactor core consists of 69 typical (17 X 17) PWR fuel assemblies with a reduced length to limit pressure losses.

Steam Generator Unit (SGU)

The SG features an annular tube bundle with helicoidal tubing.

The steam is generated tube-side. The feed water piping is connected to feed water headers, located symmetrically inside the reactor vessel within a calm zone, provided each with two tubeplates laid out vertically. The tubes depart circumferentially from the tubeplates.

A similar arrangement is provided at the top for the two steam headers connections.

The vertical arrangement of the tubeplates aims at preventing crud deposition at the tube-to-tubeplate connections, where the corrosion is likely to occur.

The higher outer rather than inner tube pressure, a reversed situation with respect to a conventional SGU, reduces the risk of flaw growth in the tubes.

Primary Circulation Pumps

The two Primary Pumps of the variable speed, glandless, wet winding type (like the pumps manufactured by Hayward Tyler Fluid Dynamics) are fully enclosed within the Reactor Vessel. The pump motor is cooled by the water of the Intermediate Plenum.

Above Core Structure (ACS)

The ACS, shaped like a flat-bottom cylindrical glass, provides the support for the core instrumentation and forms the inner wall of the annular riser of the primary water. The ACS is open at the top. The water within it is part of the intermediate plenum and this helps to limit the primary water inventory in the reactor module to a minimum. The ACS is flanged to and suspended from the top of the Inner Vessel for easy removal to allow standard fuel handling.

Pressurizer

The Pressurizer is of a slim cylindrical shape with hemispherical heads.

The pressure control function is carried out in the upper part, which is externally insulated to limit heat losses from the steam and hot water plena.

The remaining bottom part contains a cold water plenum, hydraulically connected to the upper hot water plenum by means of a number of pipes.

The function of the pipes is to enhance mixing of the hot water with the cold water, in case of water flow towards the reactor vessel during transients.

Interconnecting Pipe Ducts

The two Pipe Ducts between Pressurizer and Reactor Vessel connect hydraulically the top and the bottom of the respective cold water plena in order to create a common cold water plenum.

The choice of two connection levels makes natural circulation possible in case of temperature difference between cold plena. If the normal decay heat removal route (i.e. the active steam/water system) is lost, the uninsulated wall portion of the Pressurizer would thus help removing by conduction the decay heat towards the Plant Pool.

Conveyed water to and from each vessel, belonging to a common cold water plenum, does not significantly contribute to the thermal loadings on the pressure boundary during transients.

Air Coolers

Two finned-tube Air Coolers are arranged in loops in natural circulation.

Each Air Cooler is rated 1 MWth at 30 °C ambient air and 95 °C pool water inlet temperature.

The onset of natural circulation occurs every time the pool water temperature becomes higher than the ambient air temperature.

Operation of the air coolers would prevent, for unlimited time, the pool water from boiling, in case of long-term loss of the operational decay heat removal system.

The technology of the air coolers in natural circulation is derived from the design and operating experience of ANSALDO in the field of the LMFBRs.

5. FULL-POWER OPERATION OF THE ISIS REACTOR

During normal operation the hot/cold water interface level in the Lower Density Lock is maintained constant by varying the speed of the Primary Pumps.

Any rising of the interface level is counteracted by an increase of the pump motor speed. Any lowering of the interface level is counteracted by slowing down the pump speed.

Dynamic Analysis of ISIS Control System is in progress. Preliminary results, not yet published, confirm that the reactor power is controlled by the concentration of boron in the primary water and by the intrinsic negative feedback of the core.

6. NATURAL BEHAVIOUR OF THE ISIS REACTOR UNDER ACCIDENT CONDITIONS

In the design of ISIS emphasis has been put in the prevention of core damaging accidents.

The two main safety functions, reactor shutdown and decay heat removal, are performed without recourse to the usual sensor-logic-actuator chain, i.e. with no inputs of "intelligence", nor external power sources or moving mechanical parts, according to the definition of Category B Passive Components (ref. /2/).

An active Reactor Protection System, aimed at anticipating passive system interventions, is included in the design, but is not credited in the safety analysis.

As anticipated in the Reactor System Description, mixing of the Primary Water with the Intermediate Water and the consequent natural heat transfer toward the Reactor Pool is the basic feature to assure safety under Design Basis Accidents such as Loss Of the Station Service Power and Loss of Heat Sink (ref. /3/).

During these DB Accidents the pressure boundary integrity assures the availability of water to cool the core and to transfer the decay heat to the Reactor Pool.

In case of LOCA Accidents, the Core shutdown and cooling functions are possible only if a sufficient inventory of water remains available.

The design features of ISIS guarantee the availability of this water because of the prompt self-depressurization of the system which is the consequence of the same hot-cold water mixing process.

To illustrate the effectiveness of this self-depressurization capability, the two following DB Accidents are presented :

- double ended break of the lower pipe connection between RPV and Pressurizer;
- Steam Generator tube rupture.

Additionally, the extremely fast transient following an hypothetical break at the bottom of the RPV is reported as an exercise to better understand the thermalhydraulic phenomena linked to the self-depressurization.

All transient analyses have been carried out using the RELAP5 computer code, with a nodalization made up of 256 control volumes 262 flow junctions and 78 heat structures; neutronic point kinetics has been used to evaluate the power in the core.

Loss Of Coolant Accident

This accident consists in a double ended break of the lower, 150 mm nominal diameter line connecting the RPV and the Pressurizer. This accident scenario has been chosen because this is the largest line of the pressure boundary and also because the break location is far from both Density Locks, thus worsening the loss of cold water from the vessels (ref. /4/).

Considering that, the break location is 25 m below the Reactor Pool water level, the absolute pressure at the break outside the RPV is 3.5 bar. No action is credited of any active protection or control system.

When the accident starts, interconnected thermalhydraulic phenomena occur simultaneously within both RPV and Pressurizer. Cold water outflows from both RPV and Pressurizer; hot primary water replaces the losses in the Intermediate Plenum through both Density Locks. This phase lasts about 2-3 seconds. Then flashing hot water causes Primary Pumps cavitation which, in turn, allows the inlet of intermediate water into the primary system and the Core via the Lower Density Lock with a quick decrease of generated power.

The Reactor behaviour can now be explained considering that the Primary Pumps remain cavitating all over the transient and the primary system behaves like two channels hydraulically connected in parallel.

Both channels, the one made up by the Core and the Riser, and the second by the Downcomer and the SGU, are alternatively flooded by intermediate water entering the primary system through the Lower Density Lock.

Self-depressurization of the system takes place mainly because of the following two water mixing effects (fig. 3):

<u>In the RPV,</u> hot primary water flowing from the Upper Density Lock mixes up with the large volume of cold intermediate water of the RPV Head, purposely provided for this function.

<u>In the Pressurizer,</u> hot water flowing down through the vertical pipes mixes up with the large volume of cold intermediate water underneath.

The system pressure at the break equals the external pressure in about 450 seconds.

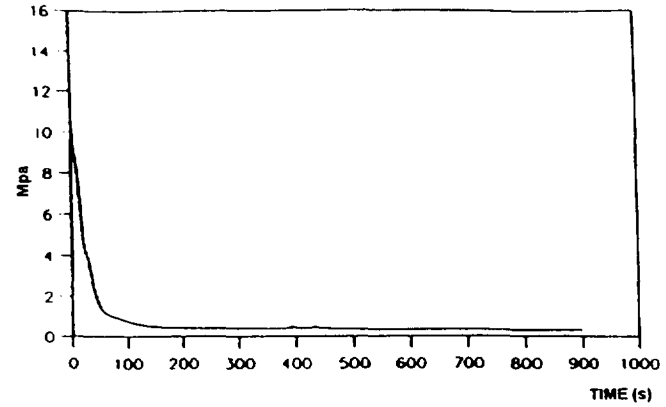

Fig. 3 - LOCA
Core pressure

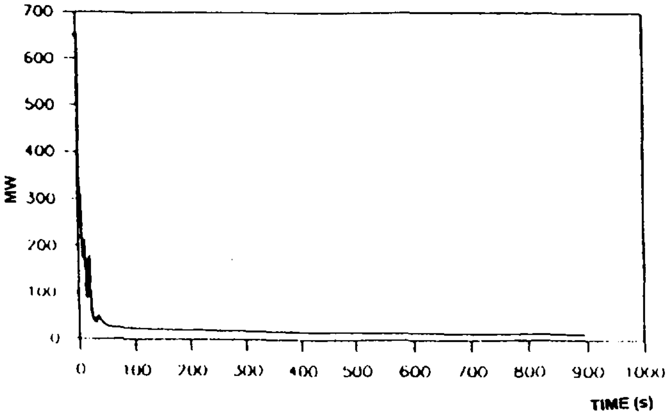

Fig. 4 - LOCA
Nuclear power

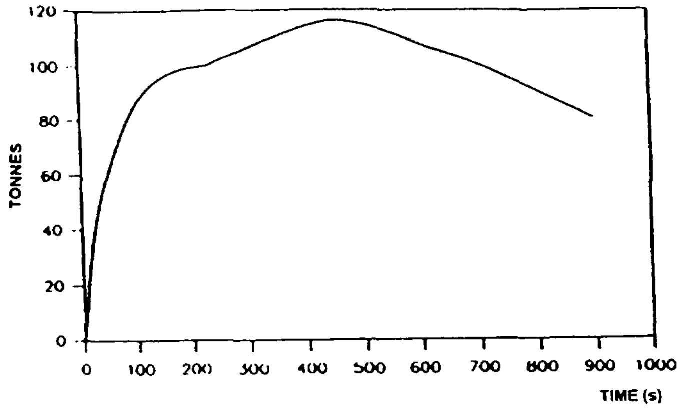

Fig. 5 - LOCA
Cumulated water loss

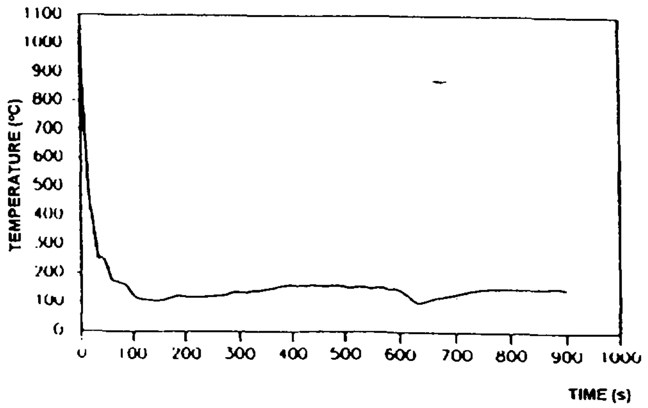

Fig. 6 - LOCA
Maximum temperature of average fuel rod

At this moment the RPV water stops flowing out and reversal flow of cold, high-boron water from the Reactor Pool sets on.

The core is shutdown (fig. 4) by intermediate water entering through the Lower Density Lock.

Figure 5 shows that the maximum cumulated amount of water loss is less than 120t (approximately 25% of the initial inventory) and only the following regions of the Reactor Module remain temporary uncovered:

- the Head of the RPV (with the water level always remaining above the Upper Density Lock);
- the Pumps, the upper part of the Riser and the SGU;
- the hot region of the Pressurizer.

Later on in the transient, reversal flow from the Reactor Pool starts recovering the water level in the RPV; at the end of the computer run (i.e. after 900 seconds) about 40 t of water have already entered the RPV from the Reactor Pool.

During the transient the Core never uncovers or heats up as shown in Figure 6. The maximum temperature of the "average" fuel rod has remained lower than at nominal conditions. A similar behaviour is shown for the clad surface temperature.

Steam Generator Tube Rupture

In this accident a break of 10 cm^2 cross section located at the connection between SGU tubes and steam headers is simulated; the break size is approximately equivalent to the cumulated cross sections of 8 SGU tubes.

No credit has been taken for action of active systems that can mitigate the consequence of the [1] accident, but for the Primary Pumps Speed Control System which delays the inlet of highly borated water through the Lower Density Lock. The steam pressure and the feedwater flow rate are assumed accordingly to remain constant during the transient[1]

When the accident occurs, water from the primary system enters the SGU ruptured tubes at a max mass flow rate of 96.5 Kg/s.

An equal amount of intermediate water enters the primary system through the Upper Density Lock as long as the Primary Pump Control System is capable to control the hot-cold interface level in the Lower Density Lock. Primary water with increasing boron concentration enters the core and reduces the generated power (fig. 7).

The amount of intermediate water entering the Upper Density Lock is replaced in the RPV by water leaving the Pressurizer. In the Pressurizer itself fast depressurization takes place because of the hot-cold water mixing process already explained above for the LOCA transient.

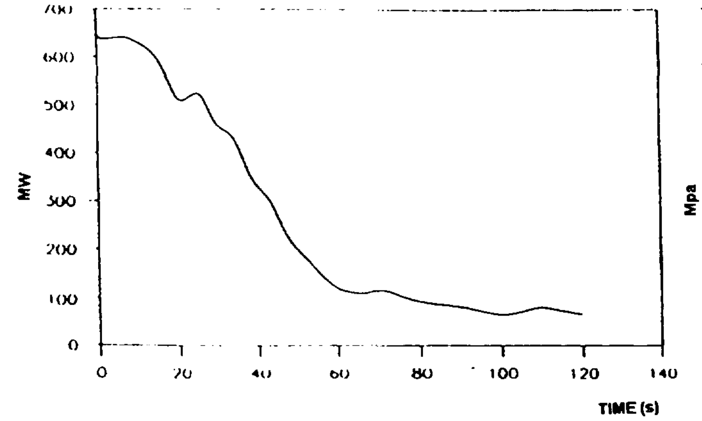

Fig. 7 - Steam Generator Tube Rupture
Nuclear power

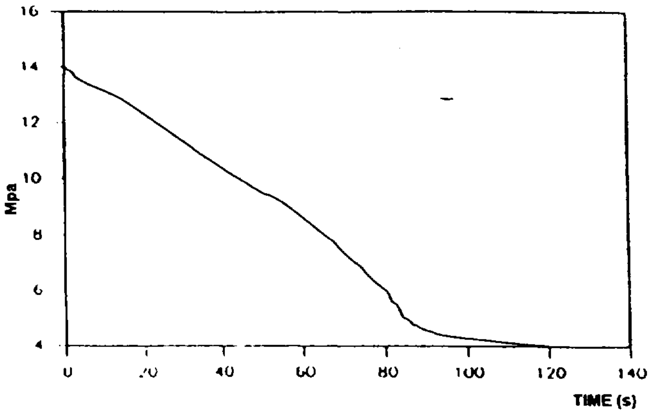

Fig. 8 - Steam Generator Tube Rupture
Core pressure

Both effects of reduced core power with associated lower primary water temperature and Pressurizer self-depressurization reduce the overall primary system pressure (fig. 8) down to the secondary system pressure (tube-side SGU pressure) which has been assumed to remain at its nominal value.

At this time the primary water stops flowing into the SGU tubes. Figure 9 shows that the cumulated amount of water loss is less than 8 tonnes which corresponds to the inventory of the hot water in the pressurizer.

The curve of the fuel temperature shows a steadely decreasing pattern, fig. 10.

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[1] Crediting the SGU isolation, the transient would behave very similar to the transient of loss of heat sink which has been shown to cause a fast reactor shutdown (ref. /4/)

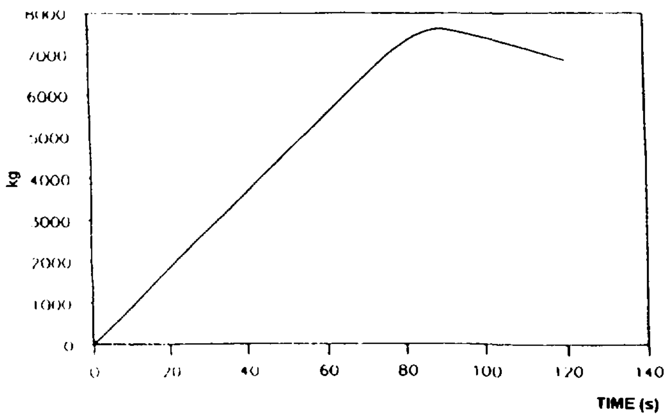

Fig. 9 - Steam Generator Tube Rupture
Cumulated water loss

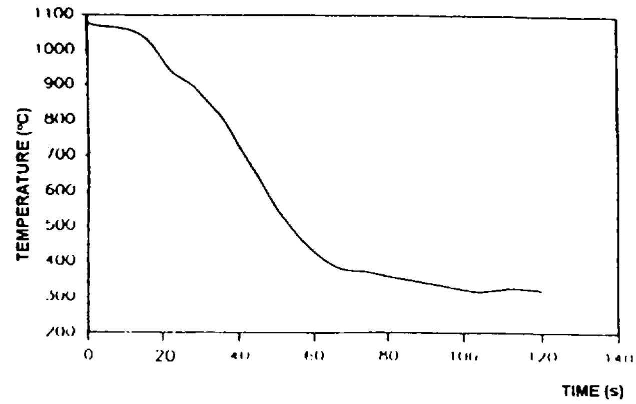

Fig. 10 - Steam Generator Tube Rupture
Maximum temperature of average fuel rod

Break at the bottom of the Pressure Vessel

In this exercise an hypothetical break of 500 cm^2 cross section has been assumed to occur at the bottom of reactor pressure vessel; this accident scenario is arbitrary and imagined to generate a very severe thermalhydraulic transient; in fact the break is positioned at the lowest location of pressure boundary and therefore has the potential of completely emptying the RPV. This exercise is intended to demonstrate that the self-depressurization process can avoid the uncovering of the core even in this case. No protection or control systems, no any other active system was credited during the accident analysis.

When the transient starts, there is a large blowdown of intermediate water from the RPV and Pressurizer into the Reactor Pool (the initial mass flow rate through the break is about 7000 kg/s). The escaping flow rate is fed by displaced primary water which is mostly contained in the SGU. Primary water leaves the SGU from the bottom via the Downcomer and, after few seconds, also from the top via the Primary Pumps and Upper Density Lock.

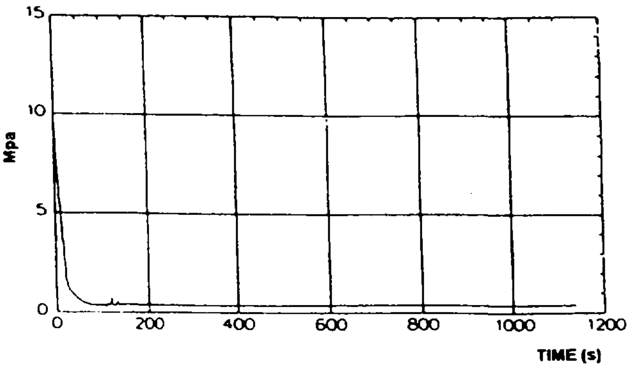

Fig. 11 - Break at the bottom of the RPV
Core pressure

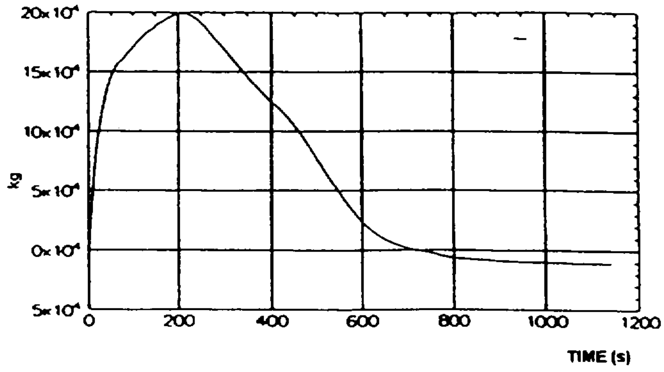

Fig. 12 - Break at the bottom of the RPV
Cumulated water loss

At the very begining of the transient the water flowing down through the Downcamer splits in two streams: the one leaves the Inner Vessel through the Lower Density Lock and the second flows up through the Core, the Riser and leaves the Inner Vessel through the Upper Density Lock. The reactor core is continously fed by primary water flowing upwards and its temperature is continuously decreasing because it is kept cooled since the beginning of the transient.

At the time of about 7 seconds, with Primary Pumps in cavitation, the primary water stops leaving the Inner Vessel through the Lower Density Lock and a reversal flow of intermediate water sets on flooding the core.

At this moment the usual way of natural circulation of ISIS reactor is recovered and the primary system fed with cold and borated water.

The mixing of cold and hot water initiates the self-depressurization of the system in the way described before for the case of LOCA (fig. 11).

The system continues its depressurization up to the time of about 200 seconds when its pressure drops below the Reactor Pool pressure at the break location.

At this moment, the total mass of displaced water (figure 12) is less than 200 tonnes (approximately 50% of the total inventory of one module) and the RPV has been emptied only down to about the center line of the SGU.

After about 1000 seconds, the initial water inventory is completely recovered and the reactor is in the state of stable cold shutdown.

The evolution of the generated power is shown in fig. 13; the power reduction during the first 7 seconds is caused by the void effect associated to the depressurization and the following shutdown is assured by the borated water.

The fuel temperature steadily decreases as shown in fig. 14 and 15.

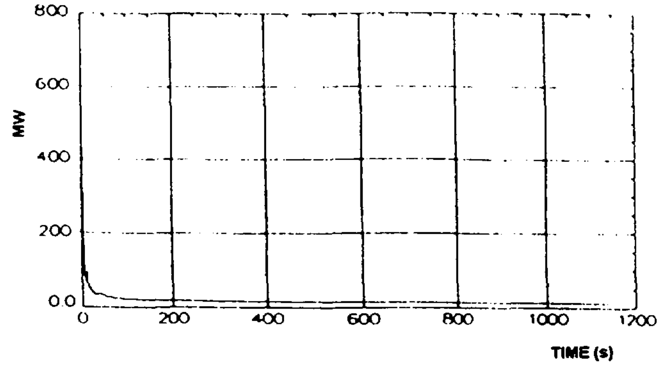

Fig. 13 - **Break at the bottom of the RPV**
Nuclear power

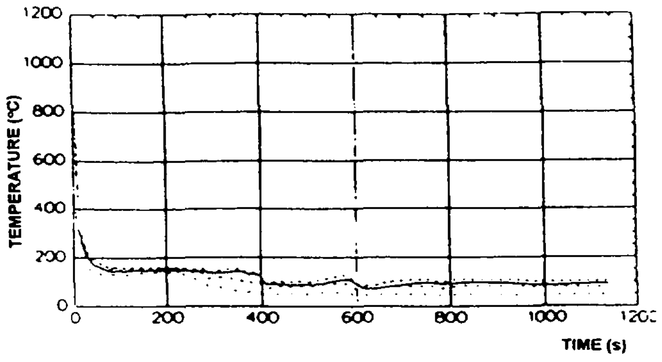

Fig. 14 - Break at the bottom of the RPV
Maximum temperature of average fuel rod

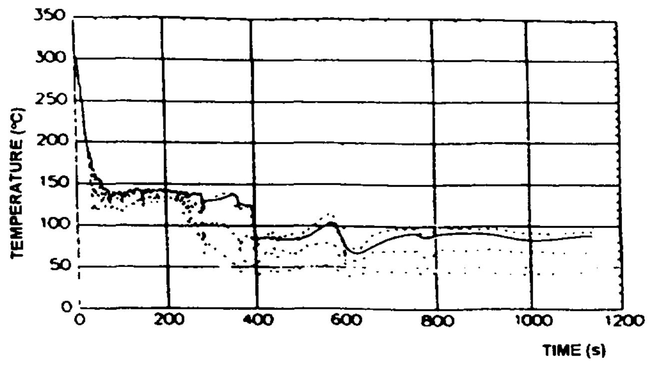

Fig. 15 - Break at the bottom of the RPV
Clad surface temperature at different elevations

7. MODULAR PLANT

The present international trend in the nuclear industry focuses on the simplification of the nuclear plants and on the reduction of the construction time. The reduced size of the most attractive modular reactors is dictated by the design target to remove the decay heat directly through the wall of the reactor vessel itself, thereby drastically reducing the number of safety-related systems.

The selected unit power of the ISIS Reactor Module (200 MWe) is consistent with this design target.

Layout studies of the ISIS power plant are in progress in ANSALDO, to optimize component arrangement and reduce the erection time of the Reactor Modules and of the Balance of the Plant.

8. ECONOMIC ASPECTS

At the beginning of the development of the ISIS concept (about seven years ago) it seemed reasonable to foreseen a moderate increase of the cost of the fossil fuels in the near future which would have improved the economic competitiveness of nuclear energy. Today, instead, two facts worsen this competitiveness:

- the fossil fuels price has remained low and stable,
- the efficiency of the modern electric energy generating fossil fuelled power plants has importantly increased.

The importance of the second fact is such that it will drastically affect the energy market, in particular the market of nuclear energy.

In the past, the efficiency of electricity production of the nuclear power plants was similar to that of the conventional power plants. Under that condition it was profitable to generate electricity by the large-size nuclear power plants that dominate the nuclear panorama.

Today, the efficiency of the modern Combined Cycle Turbo-Gas (CCTG) Power Plants has exceeded 50% and in the near future (before the year 2000) will reach and perhaps trespass 60%, while the efficiency of the nuclear water reactors stagnates at about 33%.

The higher efficiency of the CCTG will have two main consequences in the energy market.

The first consequence is that, at stable fossil fuel cost, the cost of electricity will be reduced while the cost of heat will remain substantially stable.

The capital cost of the nuclear power plants, at stable O&M and nuclear fuel costs, should be reduced to maintain the same level of competitiveness.

Fig. 16 shows how much the capital cost of a nuclear power plant would have to be reduced in the range of 50 to 60 % electrical generation efficiency of CCTG considered at stable capital cost.

The second consequence is that the fraction of power that can be extracted at low cost as useful heat for district heating or industrial use from a modern fossil fuelled cogenerative power plant reduces with the increasing efficiency in electricity generation.

A balanced mix of nuclear and fossil-fuelled plants can help to achieve the optimum ratio of thermal to electric energy generation for sites with high heat demand.

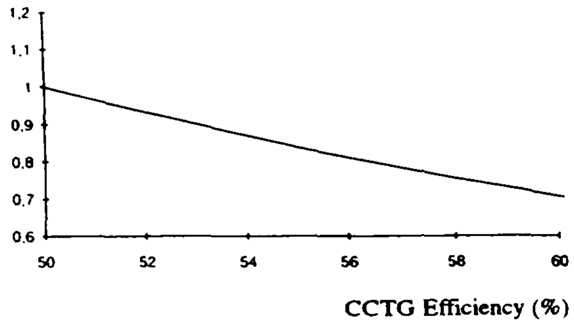

Fig 16 - Reduction of the capital cost of a nuclear power plant needed to maintain the competitiveness with the modern CCTG plants

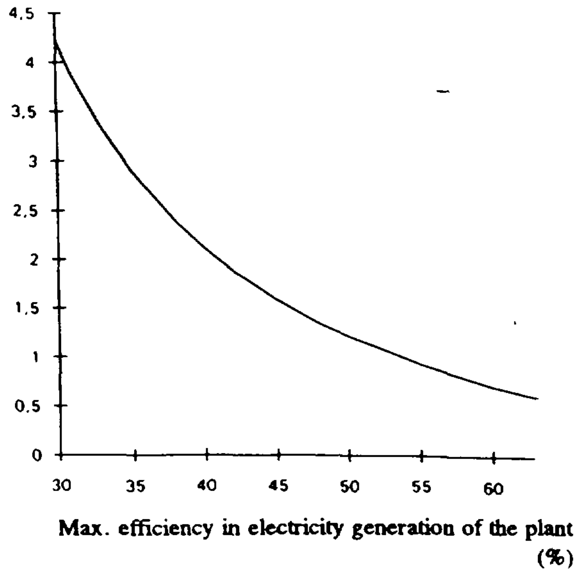

Fig. 17 - Ratio of the low cost heat to electric output of a cogenerative fossil fuelled plant

Fig.17 gives the fraction of the thermal energy that can be extracted at low cost from a fossil-fuelled plant vs the max. efficiency in electricity generation of the same plant when used only for electricity generation.

A preliminary economic evaluation carried out comparing 60% efficient CCTG, co-generative CCTG, conventional boilers and nuclear power plants, has shown that nuclear power plants could recover part of their economic attractiveness if used as co-generating or as thermal power plants.

The co-generative use appears attractive from 3000 hours/yr. upwards. That means an increase of capital cost of the economically viable nuclear plant of more than 50% over the capital cost needed for competitiveness with the CCTG plants in case of electric energy generation only, in other words an increase of the value of the cogenerative plant in the order of more than 50 %.

The increase of the value of the nuclear plant can even exceed 100% for specific site conditions where heat can be used during the most part of the year.

An obvious condition for interest of a prospective utility in a co-generative nuclear plant is that an adequate reactor design exists, that, besides featuring public-acceptable characteristics of radiological safety, be designedto overcome the unfavourable scale-effect on cost of downsizing, because the thermal power needed is in the order of hundreds of megawatts against the thousands of megawatts available from the today large nuclear reactors.

In the view of a designer, the smaller reactor can be competitive, in spite of downsizing, provided that:

- the number of the active safety related systems of the larger plants is strongly reduced,
- the mass of steel to installed power ratio is not significantly increased,
- the operation & maintenance costs do not become excessive.

The ISIS reactor has been designed to cope with these requirements.

The technical features and the results of preliminary analyses for an use of ISIS as co-generating reactor can be summarised as follows:

- no active systems are necessary to assure safety. All active safety systems can be eliminated.
- the specific mass of steel of the ISIS NSSS is comparable to that of the large modern PWRs. This is possible also because of the milder operating conditions of a reactor designed for co-generation (e.g., lower operating pressure).

Ongoing studies explore furthermore the possibility of reducing operating & maintenance costs, taking profit of the predicted simple operation of ISIS and of the modular approach that makes possible to share facilities, such as the fuel and component handling equipment, for servicing identical reactor modules of a multi-module ISIS NPP.

9. CONCLUSION

The ISIS is an innovative Nuclear Power Plant under development in ANSALDO. It is based on original ideas derived by ANSALDO experience on proven LWR and LMR technologies.

The main features of ISIS are as follows:

- Outstanding passively safe behaviour of the Reactor, which means core shutdown and cooling functions ensured in all accident conditions and no release of primary coolant outside the Reactor Building.
- Compact reactor layout and modular fabrication, made possible by the integrated design of the primary circuit.
- Flexible reactor concept for electricity generation or combined generation of heat and electricity, made possible by its modular solution and low cost sensitivity to downsizing.

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