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NATURAL CIRCULATION AND INTEGRATED LAYOUT PRESSURIZED WATER REACTOR

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ABSTRACT

A natural circulation and integrated layout PWR is here proposed by a preliminary feasibility study. Its main features are: low unit power and core power density, the same temperatures and pressures of current PWRs, no fuel melting scenario, passive protection systems, pressure suppression containment, long fuel cycle duration and no boron in the primary water. The power station can be designed for electrical power production (700 MWth - 225 MWe), or for combined electricity and heat production (200 MWth); this last solution seems viable even for urban siting. Here only the 225 MWe plant is described. The reactor startup is obtained by heating with a constant and low neutronic power, and a possible procedure is indicated and analyzed. The ATWS of a steam line break accident in hot standby conditions is studied by a suitable calculation program and the results are positive.

INTRODUCTION

The nuclear moratorium existing in practical terms in most countries of the world has helped us to reconsider the basic principles on which nuclear reactors were designed in the past. This process, still under way, has implied a certain reversal of some of these principles, which can be synthetically defined as: i) to increase the unit power, ii) to reduce the design margins, iii) to improve the safety by more complex process and protection systems. These principles were justified by the main goal of the cost reduction, taking into account the need to comply with the increasingly stricter safety requirements of the licensing Authorities. In practice, this choice resulted questionable, a posteriori, because the economy of scale, connected to high power ratings, was virtually offset by an exaggerated lengthening of the licensing process and the construction time, a reduced opportunity of a plant standardization and an increased difficulty in plant decommissioning. The economic gain of design margin reductions was practically reversed by the penalties of a less flexible plant operation, and a shortening of the fuel cycle duration, both implying a lower load factor, and the need to implement more sophisticated, complex and costly safety systems.

This trend is confirmed by the characteristics of the new nuclear reactor designs, proposed by different organizations in the world. These designs, founded on the already existing reactor technologies, have the aim not only of improving their inherent and passive behavior with respect of safety, but also of reversing the above mentioned principles, so as to render the nuclear product more sound from an engineering point of view and more palatable to the public opinion.

A further aspect concerns the need to withstand the severe accident scenario, which implies the extended melting of the fuel. A lot of studies has been recently carried out on this problem, providing a better insight on the overall picture, which shows that the consequences of this accident can be circumvented by a suitable design. However, the crucial point is the hard challenge to experimentally demonstrate the correctness of this conclusion by means of a significant test

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carried out using the irradiated fuel in a realistic plant accident scenario. If this is the case, it appears problematic the licensing by the Safety Authorities of plants conceived to withstand such an accident. Therefore, the best solution seems to be that to envisage a reactor design, in which this accident can be reasonably excluded by any scenario.

In conclusion, it appears that a more fruitful approach to reactor design is to reverse the above principles and then: i) to reduce the unit power, ii) to increase the design margins, and iii) to simplify the process and protection systems. This choice should also satisfy the cost reduction goal, which in the final analysis is the crucial point for any actual application.

The natural circulation reactor here proposed follows the above suggestions. The main features of this reactor concept are the following:

- low unit power;
- low power density;
- long fuel cycle duration;
- no boron poisoning in the primary system but adoption of burnable poisons inside the fuel;
- internal hydraulic driven control rods;
- no fuel melting scenario;
- natural circulation of the primary coolant;
- primary coolant flow outside steam generator tubes (no stress corrosion problems);
- integrated layout including the pressurizer;
- the same thermodynamic performances of current PWRs (pressures and temperatures);
- pressure suppression containment;
- simple and passive protection systems.

Two reactor designs have been conceived: i) a 700 MWth (225 MWe) modular reactor to be assembled in several units to form a big power station in the same site, ii) a 200 MWth reactor to be used for a combined heat and electricity production. This last solution can be adopted for urban heating, and this use can raise the interest by many countries, also because the heat and electricity ratio (approximately 2 to 1) satisfies the requirements of an urban use. A similar design of a natural circulation reactor for electricity or heat production was proposed by argentinean designers, who studied a 100 MWth reactor, named CAREM (CAREM, 1991).

Here only the 225 MWe reactor will be presented.

REACTOR DESCRIPTION

The main data of the plant are listed in Table 1, while sketches of the reactor vessel, the passive cooling systems and the overall layout inside the containment of the NSSS are shown in Figs. 1 and 2.

The maximum inner diameter of the reactor vessel is assumed equal to 5 meters, which is well within the fabrication limits (for instance South Texas vessel has 5.57 m), and it should not imply significant transportation constraints, also because the cylindrical portion of the vessel is divided in two sections, connected by a flange. The volume power density is substantially lower than that of current PWRs: 29 kW/liter instead of 105 kW/liter. A previous optimization study (Lombardi, Tenconi, 1991) shows that no economic penalty is associated to such a reduction in power density, while the operational and safety behavior of the overall plant is appreciably improved.

The fuel assembly is of 15x15 type instead of the more usual 17x17 one, in order to slightly reduce fuel fabrication costs, still obtaining an average linear power density much lower than that of the current 17x17 design (35 %).

The height over diameter ratio (H/D) of the reactor core is chosen in order to reduce to a minimum the overall height of the vessel, without yielding a reactivity penalty. The value of 0.93 adopted in this design is very close to the maximum reactivity condition, as it is shown by a parametric study carried out by the WIMS program (Askew et al., 1966). The boron poisoning of primary coolant is eliminated in order to avoid any coolant chemistry concern, to increase the negative moderator reactivity coefficient, and above all to avoid in the case of power transients any dangerous reactivity insertion due to some possible boron stratification. Then the reactivity control is made through conventional rods, driven hydraulically inside the pressure vessel, as already positively tested in a small Chinese PWR (Wang Dazhong, 1993). Burnable poisons (IFBA type: Integral Fuel Burnable Absorbers) are inserted into the fuel rods to reduce the reactivity span of the control rods. The scram is obtained by two independent sets of rods, possibly based on different activation device, in order to duplicate this safety function.

The fuel cycle length in equilibrium conditions is about 4.5 years, by assuming an average burnup per cycle of 12 MWd/kg and a load factor of 0.80. This long duration has surely positive effects on plant load factor. However, the attainment of these equilibrium conditions is undoubtedly a lengthy process, which can cover a significant portion of the plant life.

The twelve steam generators, located inside the reactor vessel in the peripheral zone, are of the recirculation type. This choice is not the final one, because the once-through type is still under consideration. The present choice is motivated by the need to reduce to a minimum the heat transfer surface; moreover the secondary coolant chemistry is made easier, as well as the control of the exit conditions. However, two steam drums are needed outside the pressure vessel, and this renders the overall layout more complex. The primary high pressure water flows outside the tubes, thus preventing stress corrosion cracking effects and obtaining safety advantages in the case of a tube failure. The geometry is the straight tube one, while special constructive solutions are to be devised to allow the removal of the steam generators for repair or substitution; these solutions should be reasonably implemented in this arrangement. The same solution is adopted for the twelve Heat Removal Exchangers (HRE), located just above the steam generators. Two sets of four HREs are directly connected to a pool, into which they discharge the hot water and are fed by the cold water from the pool bottom by suitable spargers (see Fig 1). The other two sets of four HREs are connected to an active cooling system in order to avoid the pool heat up and to

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simplify the temperature control of the primary system during the start-up operation, the operational transients, any planned cooling down operation, and the cold shut-down conditions. Each set, both passive and active, is designed to dissipate the whole decay heat, so as to obtain a double duplication of this safety provision. The material is Inconel 600 for all the components.

The pressurization of the primary coolant is obtained by a gas cushion inside the pressure vessel dome. The gas is a mixture of saturated steam at the reactor outlet temperature (partial pressure 12700 kPa) and nitrogen (partial pressure 2800 kPa). The pressure control is obtained by a feed of nitrogen and a bleed of the gas mixture, while the water level control is obtained by a feed and bleed of the primary fluid. The large gas volume (25 m^3) allows a significant dumping of the pressure oscillations during normal operation at constant power. The maximum pressure is obtained at the nominal power, while at lower power levels it is decreased accordingly with the lower reactor outlet temperature allowed by the control system through the variation of the secondary steam pressure, without varying the nitrogen content in the pressurizer volume.

PROTECTION SYSTEM

The containment is of the pressure suppression type (see Fig. 1). The water pool into which the steam-gas mixture of the dry well is discharged during a LOCA, is 15 m high, so as the minimum pressure inside the dry well in these conditions never goes below two atmospheres absolute (see below for the reason of this choice).

In the case of a steam generators loss as heat sink, the decay heat is absorbed by the heat exchangers HRE located above the steam generators. The operation of two sets of HREs is passive, being activated by opening a valve (passive system of fourth class according to IAEA classification). The thermal power is dissipated by heating the water contained in two out of four compartments of the pool inside the dry well. The pool water is brought up to the boiling point (100-120 °C according to the dry well pressure) and from this time on the produced steam is discharged to the wet well pool through the discharge ducts. The pool water can last several hours and can be replaced from an external reservoir through charging pumps.

In the case of LOCAs, owed to a pressure vessel breach or a failure of the servicing tubes connected to the vessel, or a steam generator tube rupture, the water level inside the vessel remains always well above the reactor core, by adopting a suitable out of vessel geometry in the lower dry-well zone (see Fig 1). The steam generated by the decay power is condensed in the passive heat exchangers and recycled into the reactor. The minimum absolute pressure of two atmospheres in the dry-well helps the condensation process. Even in the unforeseeable complete loss of water inside the vessel and the dry-well, the natural circulation of the remaining steam-air mixture through the core and the passive heat exchangers is capable to keep the fuel temperature below the melting point. However, this situation is beyond any conceivable hypothesis, because of the geometry of the dry-well lower zone and because further water is drained by gravity from the water pool located inside the dry-well, via check valves.

In the case of a steam generator tube rupture, the steam line geometry is such to avoid any siphon effect, which might yield a partial voidage of the vessel, if the secondary circuit remains open toward the discharge lines.

PLANT START UP

The reactor heating during start-up is obtained by nuclear power. This is a significant difference to current PWRs. In the initial cold conditions, the water level is realized by a nitrogen cushion brought to its nominal pressure (2800 kPa). Then the control rods are raised till a small positive reactivity value is obtained. The reactor power starts going up and the water inside the core is heated up: a slow natural circulation is initiated, thus spreading the water heating within the whole vessel. The reactor power is stopped at a very low value: in the present example about 17 MW. From this time on the reactivity is brought to zero and this value is kept constant. The cold sink is made by the two active sets of the HREs, activated when the reactivity is brought to zero; their inlet temperature is controlled in such a way to keep them always few degrees below the inlet primary temperature. Under these conditions the reactor power is higher than that extracted by the HREs, and the reactor is progressively heated up. Anyway, the reactivity negative temperature feedback involves a progressive control rod extraction, in order to keep a null reactivity value. The liquid level raises, due to the thermal expansion, and then the primary water is continuously bled. The steam partial pressure raises as well, and accordingly the overall pressure increases up to the maximum nominal value. This procedure assures that the heat up process occurs keeping limited the reactor power, so as to remain within the capabilities of the HREs, to slow down the dynamics of the operation and to reduce the wasted energy. Once the primary temperatures approach the nominal value, the steam generators are put in operation and the HREs are disconnected. Finally, the reactor power is progressively raised to the nominal value by acting on the control rods.

The above procedure was analyzed by means of the TRAP code (Brega et al., 1996) and the results of the transient lasting 34000 seconds are reported in Fig. 3.

SAFETY TRANSIENTS

The most crucial safety transient seems to be the steam line rupture. In principle, the lowering of the inlet core coolant temperature connected to the high negative reactivity coefficient of the moderator temperature might imply a significant reactivity insertion, before the neutronic power is transformed into thermal power, so as to reverse the downward trend of the coolant temperature. However, the small coolant speed along the downcomer and inside the core delays substantially this transient. As a matter of fact this speed is very low in hot standby conditions, which usually results the worst initial conditions for this accident.

The accident was studied in ATWS conditions, in order to verify its acceptability in these severe scenario. The transient

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was analyzed by the above mentioned TRAP program. The main initial conditions are: 1.5 MWth, - 60 pcm/°C, 289 °C. The main resulting parameters are displayed in Fig. 4 versus time, i.e.: break flow rate, steam drum pressure, total reactivity, thermal and neutronic power, core flowrate and maximum fuel temperature. The break in the common steam line is such to increase stepwise the steam flow rate from almost zero to 150% of the nominal value. The steam drum pressure first goes down rapidly, then it remains almost constant, because of the contribution of the reactor power producing new steam in the steam generator, and finally goes down steeply to zero, when the reactor power decreases. The break flow rate follows practically the same trend, while the reactor flow rate rises rapidly to a maximum slightly higher than the nominal value, then goes down to zero in a rather regular way. The reactivity has a very narrow peak of 600 pcm, which is translated in an analogous peak in the neutronic power of 1550 MW. The thermal power peak is substantially reduced and widened, so that the maximum value is only slightly higher than the nominal value (880 against 700 MW). In spite of this, the maximum fuel temperature is slightly lower than the nominal value (730 against 775 °C), because of its thermal inertia. The transient was repeated for different maximum values of the break flow rate: the results show a roughly constant value of the reactivity peak around 600 pcm, thus suggesting that this parameter is intrinsically limited by the doppler effect and the feedback of the thermal power on the core coolant temperature. In conclusion, it seems that the consequences of this severe accident are really limited, notwithstanding the conservative hypothesis adopted.

CONCLUDING REMARKS

The natural circulation integrated layout PWR seems to satisfy the criteria to avoid any severe consequences in the case of postulated accidents. This conclusion is to be verified in a more thorough way by an enlarged parametric study. The main uncertainties concern the steam generator mechanical design, the choice between recirculation and once-through steam generator, and the overall layout including the design and verification of the containment. This will be analyzed in the next future.

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Table 1. Main Plant Data

Core	
Thermal power	700 MW
Electrical power	225 MW
Pressure	15.5 MPa
Inlet/Outlet temperatures	289/329 °C
Flow Rate (bypass excl.)	3017 kg/s
Bypass flow rate fraction	0.06
Equivalent diameter	3.22 m

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Inner/Outer barrel diameter	3.49/3.57 m
Height (total/active)	3.4/3 m
Fuel assemblies (15x15)	177
Fuel rods per assembly	204
Rod diameter (outer/pellet)	10.72/9.29 mm
Cladding thickness	0.62 mm
Pitch	14.27 mm
Assembly equivalent diameter	13.57 mm
Overall assembly width	214 mm
Fuel weight	77100 kg
Volume power density	28.65 kW/l
Average fuel linear power density	6.28 kW/m
Moderator power fraction	0.03
Vessel	
Inner diameter (upper/lower)	5.0/4.0 m
Total height	24.20 m
Cylinder thickness (upper/lower)	0.26/0.20 m
Overall friction pressure losses	8.33 kPa
Overall volume	362.5 m^3
Steam Generators	
Number	12
Tube diameter (outer/inner)	10/8.8 mm
Pitch	14 mm
Equivalent diameter	14.95 mm
Tube height	3.82 m
Total number of tubes	49092
Total HT surface	5891 m^2
Steam drum number	2
Steam drum (diameter/length)	2.5/6 m
Steam drum pressure	6.2 MPa
Passive Heat Removal Exchangers	
Number	12
Tube diameter (outer/inner)	10/8.8 mm
Pitch	14 mm
Equivalent diameter	14.95 mm
Tube height	1.60 m
Total number of tubes	49092
Total HT surface	2466 m^2

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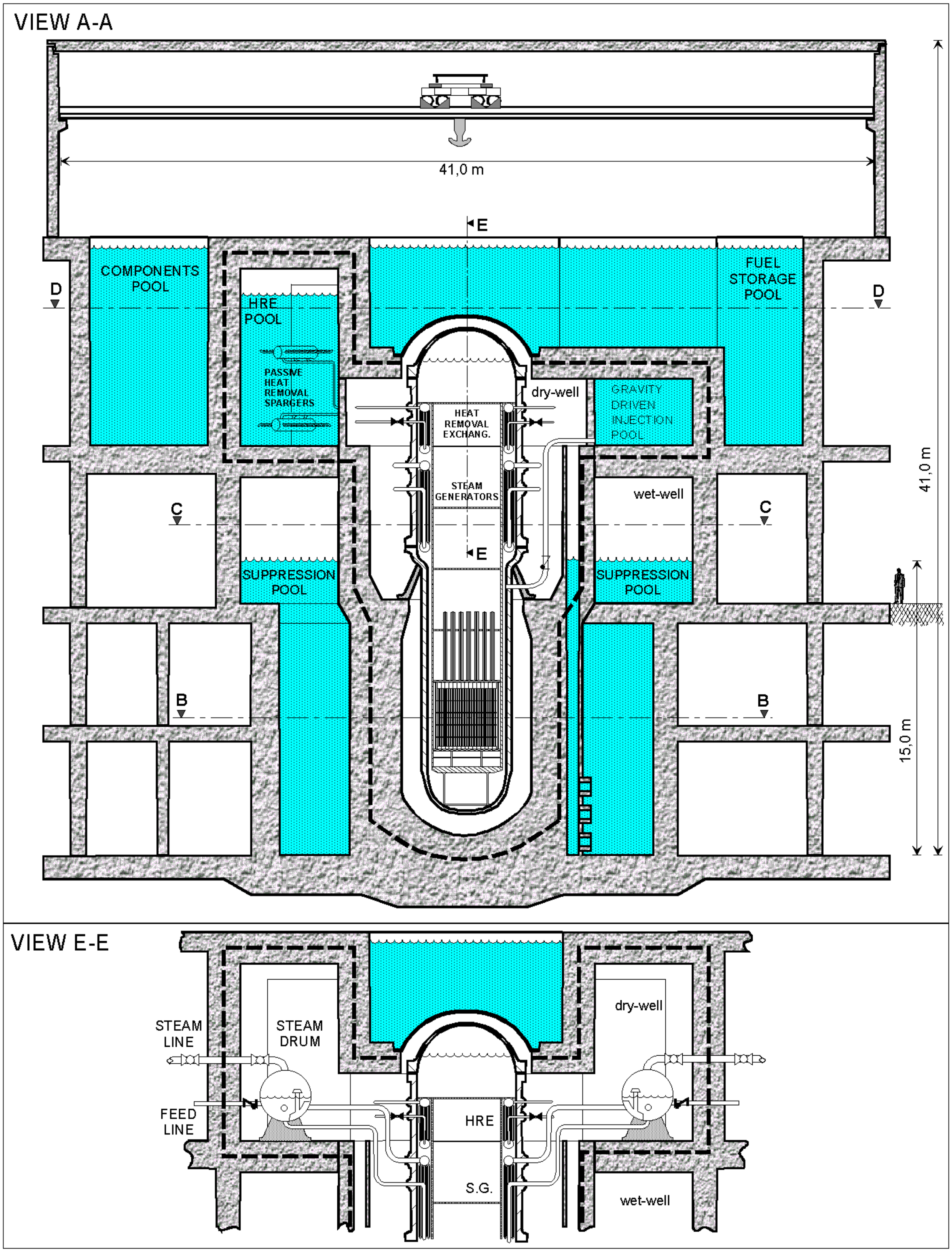

Figure 1. Layout of the Nuclear Steam Supply System: integrated pressure vessel with connected safety circuits.

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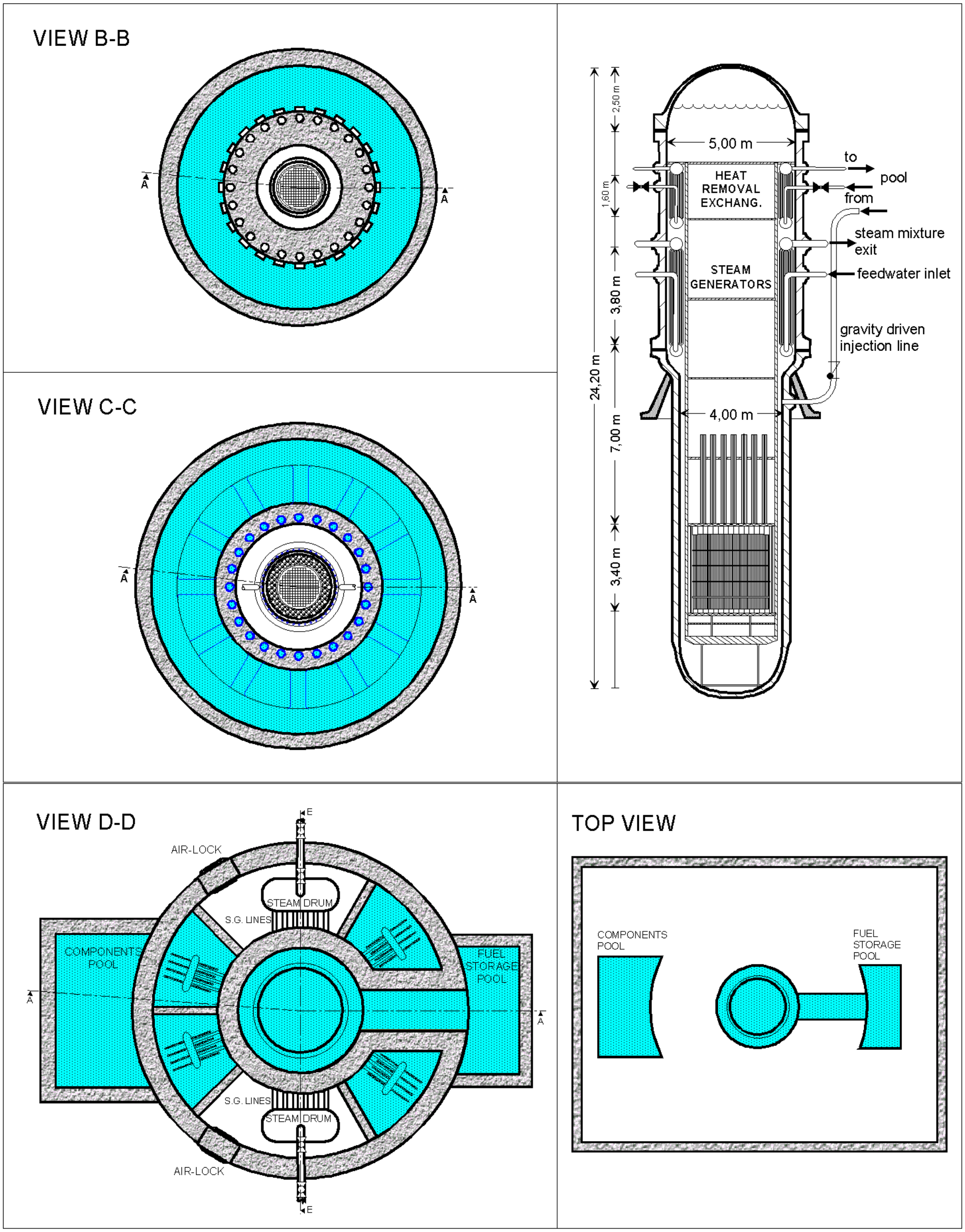

Figure 2. Layout of the Nuclear Steam Supply System: longitudinal section of the vessel and plant views at different elevations.

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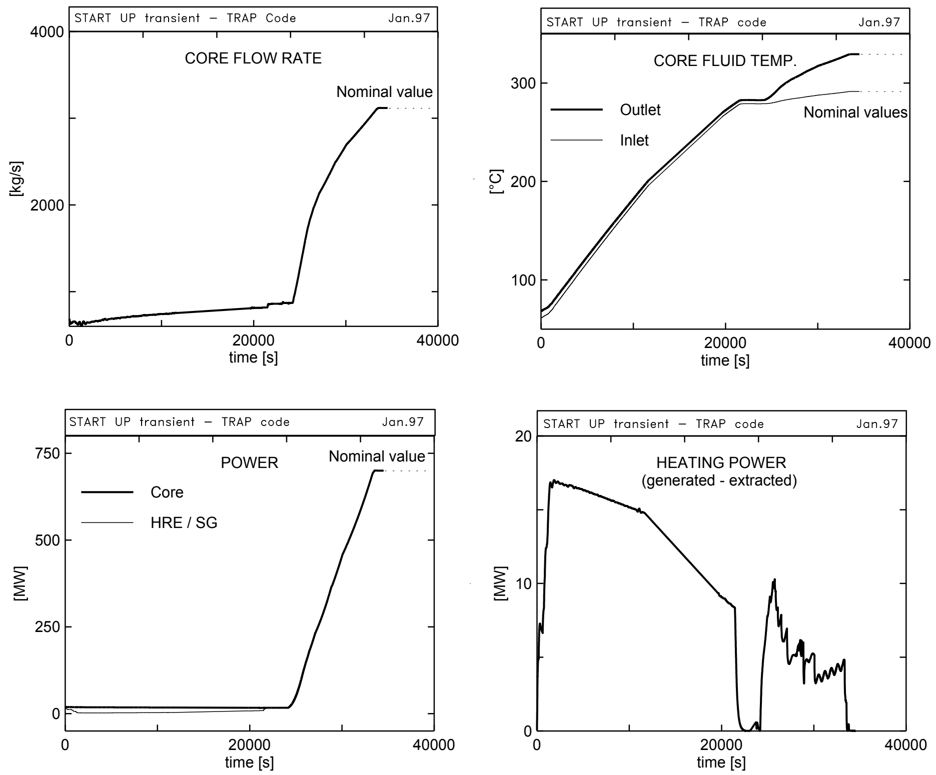

Figure 3. Typical Start up transient of the plant.

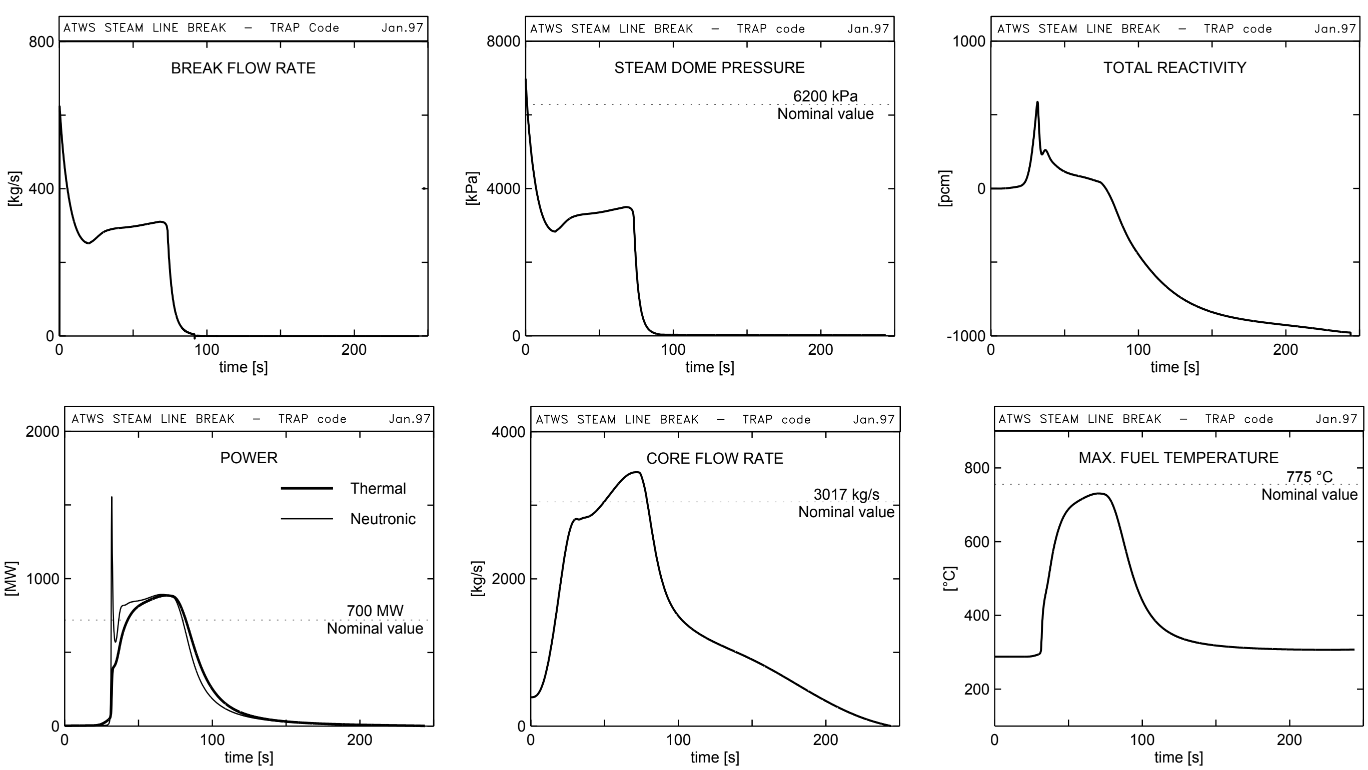

Figure 4. ATWS Steam Line Break accident in hot standby conditions.

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