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Engineering and Design of Fast Reactors [and Discussion]

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Engineering and design of fast reactors

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The paper deals with the basic engineering aspects specific to a fast breeder reactor. The characteristic design features are mainly determined by the coolant being sodium and the fast neutrons in the core.

Some particular properties of sodium, the high temperature and the large temperature differences impose severe requirements on materials and structures. Also the mode of operation, the protection systems and the auxiliary systems are strongly influenced by these conditions.

The resulting engineering implications will be illustrated by describing, firstly, the core layout and the core components, secondly, the main coolant system layout and its auxiliaries and, thirdly, the design aspects of main components such as pumps, heat exchangers and steam generators.

Introduction

This paper tries to illustrate some important aspects of the engineering science of fast breeder reactors (FBRs), concentrating on a few special design issues. The state of the art of present reactor designs and the safety-related engineering is treated in other presentations.

Even though the choice of the reactor coolant was, for many years, one of the most controversial issues, sodium, i.e. a liquid metal, is now the uncontested favourite. The other candidates – steam and helium – finally lost the competition, primarily because of the advantages of sodium in the areas of core physics and heat transfer. The latter plays an important role in emergency core cooling.

The design of such a reactor type is a challenge, demanding sophisticated engineering capability. But the experience with many sodium-cooled fast reactors, brought into operation in distinct development steps, has demonstrated that the conversion of fast fission and sodium technology into operable nuclear power stations has been accomplished.

The development went through more than 15 small test reactors with their power ranging from 1 to 100 MW_t. Furthermore, five prototype plants of about 300 MW_e were built to arrive, finally, at the Superphénix station, which has power of 1200 MW_e.

Dealing thoroughly with the science of FBR engineering would go far beyond the scope of this paper. Only the following three areas are illustrated as typical examples: core layout; general engineering of sodium systems; main sodium components.

1. Core layout

At the beginning of the FBR development U or U–Pu metal was used as fuel. Today all demonstration reactors use U–Pu mixed oxide (MOX) fuel. Recently, the progress in metallic fuel development in the United States (Argonne) has reopened the fuel question for future FBRs. However, in line with the development in Western Europe, this paper deals with a core of U–Pu MOX fuel.

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According to the basic physical process, the core consists of a central fissile zone of Pu–U mixed oxide and an enveloping fertile zone of natural or depleted uranium. In the central zone the fission of plutonium takes place by fast neutrons. In the fertile zone the surplus neutrons are captured by ^{238}U converting it into ^{239}Pu.

The absorber elements, equipped with boron carbide for the control of the fission process, are arranged in the fissile zone. This zone consists of an array of fuel pins grouped together in fuel elements. These elements form the individual handling units for core loading and unloading for the replacement of exhausted fuel.

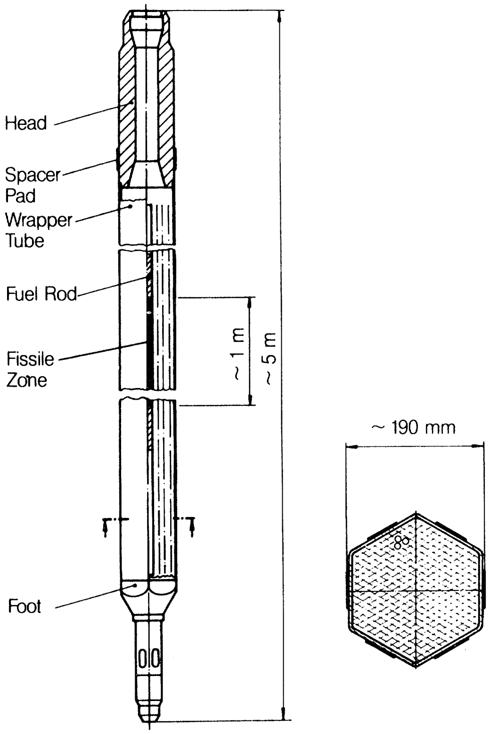

Figure 1. Fast breeder fuel element.

Figure 1 shows a representative FBR fuel element. The fuel pins are grouped into fuel pin bundles with the aid of spacers. There are two competing possibilities:

1. Grids that are axially arranged at intervals of about 20 cm. They are designed in such a way that an adequate cross section is available for the coolant flow.

2. Wires that run helically around the individual pins. By this they form support points for the neighbouring pins blocking up only a limited part of the cooling channel cross section.

Mainly as result of the differing power of adjacent fuel elements, an FBR, in contrast to a light water reactor (LWR), requires wrapper tubes as tight enclosures around the fuel pin bundles. The fuel pin bundle and the wrapper tube, together with the head and the foot, form the fuel element. Its geometric layout, such as pin pitch and spacer design, is basically governed by thermal hydraulic considerations. These are sufficient pin cooling, minimum pressure losses and limited coolant temperatures.

Irradiation to high neutron doses causes a substantial increase in volume, i.e. a decrease in

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density, of the material (swelling). Because of the flux and temperature gradients across a fuel element, swelling differs locally on account of its dependence on temperature and dose. This leads to fuel element bowing, which, in contrast to thermally induced deformation, is irreversible.

This swelling phenomenon is, for obvious reason, of great relevance for the core behaviour. After its discovery, specific development work was initiated on materials that retain a high structural stability under irradiation.

The setting-up of several hundreds of individual fuel elements to form a core structure has to take account of all deformations and interactions. The reactivity of a fast reactor core is sensitive to geometric changes. Unacceptable reactivity changes must be avoided and the stability of reactor operation must be assured under all conditions. Furthermore, the mechanical integrity of the element enclosures and the pin bundles must be guaranteed so that loading and unloading of all core elements can be accomplished.

To meet these requirements, fast breeder cores must be equipped with a core restraint system through which the position of the elements is defined. This can be accomplished either by engineering a barrel around the core or by sufficient stiffness of the core elements and their support in the grid plate.

Apart from the radial restraint of the elements, their axial restraint is also of importance. In all larger FBRs the sodium flows upwards through the core. As a result of fuel element pressure losses and buoyancy, the elements are exposed to an upward force that is higher than their dead weight. Therefore, a hold-down mechanism is necessary for axial fixing. Hydraulic hold-down devices with a special flow routing create a counterforce that compensates the upward forces.

The fuel pin consists of a steel tube and the fuel that has been filled into the tube in form of sintered pellets. The tube (clad) has a diameter of between 6 and 10 mm and a wall thickness of a few tenths of a millimetre. The clad is most important with respect to enclosing the fuel. It must keep its integrity up to the very end of its life, i.e. up to the start of reprocessing. During plant operation, this pin integrity is necessary to avoid fuel and fission product releases or contact between fuel and sodium. The integrity must be retained also in the case of accidents.

Temperature and, in particular, fission gas pressure are of importance for the clad. The anticipated burn-up of FBR fuel is 10–20% of the heavy atoms, i.e. it is almost an order of magnitude higher than for LWR fuel pins. Because of the high burn-up, fuel pins are equipped with an empty volume to accommodate the released fission gases. This fission gas plenum is arranged either above or below the fissile zone. Its size is such that the permissible cladding tube stresses caused by the internal pressure build-up are not exceeded, even at the end of the fuel pin's life. Vented fuel pins are being developed but have not come into real use yet.

Large increases in fuel volume must be avoided. Therefore, the fuel is generally designed such that there is no major fuel melting in the centre, even during an accident. A characteristic variable for this feature is the linear pin power. It basically governs the central temperature independent of the pin diameter. Depending on pin design and fuel density, the allowable linear pin power is of the order of 50–60 kW m^{-1}. To attain an even higher safety margin for anticipated failures of the safety system, considerably lower values are under discussion.

The diameter of the fuel pin is governed by the target of minimizing the fuel cycle costs and by breeding requirements.

The integrity of the fuel pin has to take account of material deterioration by the high neutron

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flux. The flux, primarily, leads to a drastic reduction of the ductility of the clad material at the end of life. In this respect, a typical effect for FBRs is clad embrittlement. It is caused by the formation of He bubbles in the steel at the grain boundaries.

2. General engineering of sodium systems

2.1. *Principal layout*

The system engineering of all large sodium-cooled FBRs is similar. The principal scheme is shown in figure 2. Two, three or four parallel sodium circuits are provided to transport the thermal energy out of the reactor core. These are the so-called primary circuits, each of which is equipped with one recirculation pump. The heat from the primary circuit containing radioactive sodium is transferred to a secondary circuit. This circuit contains non-activated sodium, and the heat transfer takes place via one or more intermediate heat exchangers (IHXs).

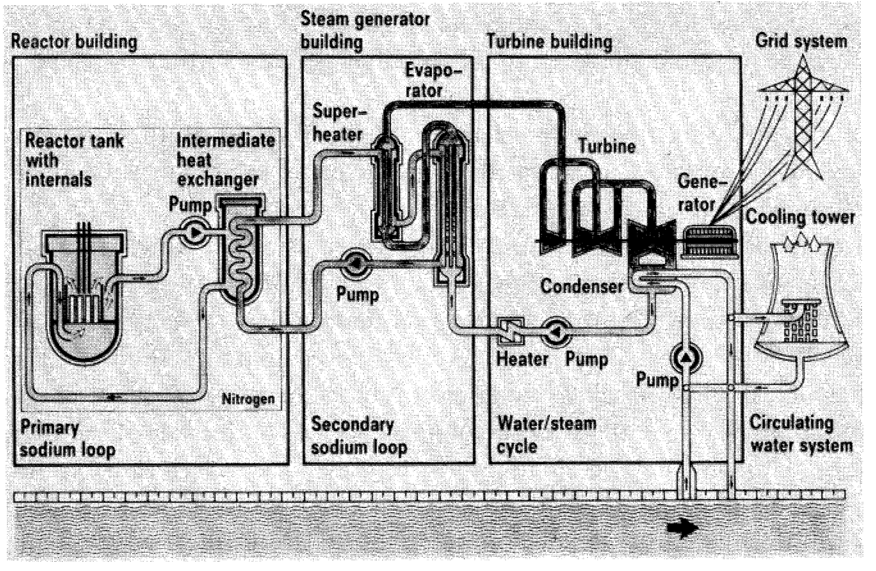

Figure 2. Basic arrangement (loop concept) of FBR.

In a steam generator, the sodium transfers its thermal energy to a water–steam system. There this energy is conventionally converted via a steam turbine and an alternator into electrical energy.

A key question concerning the primary system is secure core cooling. This is achieved by preventing unacceptable sodium losses in the event of a breach in the coolant enclosure.

Of the conceivable design variants, only the pool and the loop arrangements are of practical importance. The pool arrangement can be seen in figure 3. There not only the core but also the IHXs, the pumps and connecting pipes are located in the large sodium-filled primary vessel. The primary pumps directly take in cold sodium from the cold plenum of the vessel and transport it through pipes to the plenum underneath the core. From above the core, the hot sodium overflows into the IHXs and, having passed through them, reaches the cold plenum again.

The loop concept corresponds, in principle, to the normal design of LWR primary systems.

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It has already been shown in figure 2. The IHXs and pumps are installed in a separate compartment and are connected to each other and to the reactor vessel via pipes.

The pool concept controls an unacceptable loss of coolant, as the result of a leak in the primary vessel, by means of a guard vessel. More complex means are needed to solve this problem in the loop concept. All loop sections that are arranged outside the reactor tank below emergency level must be protected. This is done either by a second shell, by isolating systems or by leak-tight cavities with limited volumes. The reactor tank itself has a guard vessel, too.

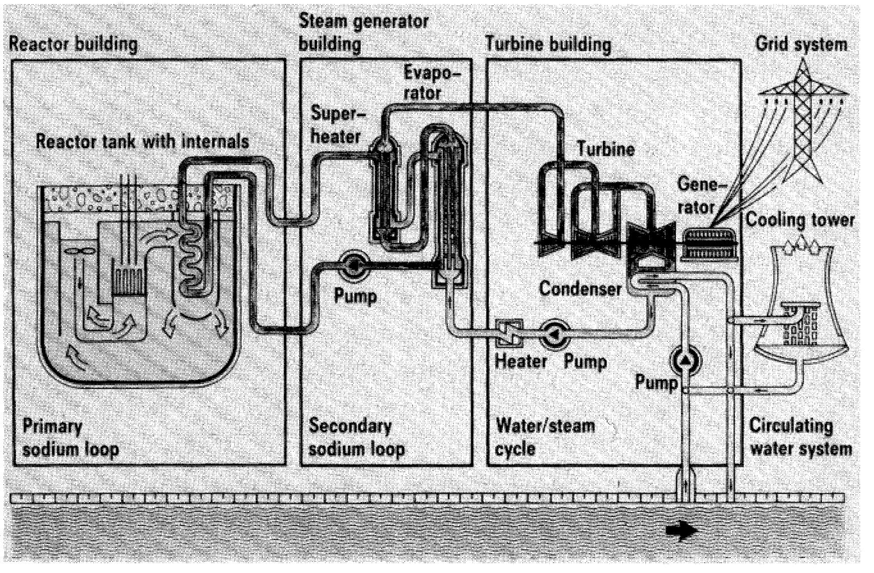

FIGURE 3. Pool arrangement of primary circuit in FBR.

The question whether a pool concept could have merits over a loop concept was a longstanding issue in FBR history. In essence, no decisive technical superiority of either system has been identified.

The difference in the two concepts mainly affects passive components forming the coolant enclosure. It does not affect the heat transfer components such as pumps and heat exchangers to the same extent.

Mainly in the interest of European harmonization, the pool concept was selected for the European Fast Reactor (EFR) project, a joint breeder design of Germany, France, United Kingdom, Italy and Belgium to be built in the late 1990s.

2.2. *Sodium properties and resulting engineering consequences*

The system engineering is strongly influenced by the properties of sodium. Sodium is the most abundant alkaline metal and the sixth most abundant element on Earth. Solid sodium looks silvery white and is of soft consistency. It is commercially produced by the dry electrolysis of NaCl (common salt). Its low neutron capture cross section makes it a suitable reactor coolant. But above all, the excellent heat transfer properties permit the necessary high power density of the fissile core zone. It has a boiling point of about 900 °C, hence a negligible vapour pressure at operating temperatures. This allows the application of almost pressureless coolant systems.

The always liquid and pressureless coolant assures permanent coupling of the core to the

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coolant by simple passive means. This is assured even in case of a breach in the coolant boundary, e.g. by means of a guard vessel around the primary system.

The low pressure of the coolant and the allowable high temperature rise on flowing through the core result in an almost inherently safe core cooling in emergency cases. This can be provided even in a fully passive way.

The specific heat capacity of sodium is approximately one third of that of water. To minimize the mass flow through the core and the systems, high temperature rises in the core of 150–180 K are chosen.

A high plant efficiency needs high temperature differences across the heat-exchanging components to minimize the heat-exchanging surfaces. In contrast to that, the temperatures must be limited to retain the structural integrity of the fuel pin and the heat transfer systems. As a compromise the core outlet temperatures are fixed between 500 and 550 °C.

Like all alkaline metals, sodium reacts vehemently with water because sodium–water reactions are exothermic. Violent reactions, in the case of steam generator tube failure, between radioactive sodium and water must be avoided. They would lead to releases of radioactivity into the environment. Therefore all FBRs are equipped with a non-radioactive intermediate circuit between the core cooling circuit and the steam–water side (see figure 2).

A sodium fire is like a fire of a burning liquid. Sodium burns with short flames of about 10 mm length. Sodium, discharged to the atmosphere, only ignites at temperatures above 150–200 °C. The exact ignition temperature depends on the atmospheric humidity. A high humidity favours fire ignition. Even when a pool of molten sodium is involved, the flames spread comparatively slowly. The calorific value of sodium and the burning rate are each between $\frac{1}{3}$ and $\frac{1}{4}$ of the respective values of gasoline.

Sodium does not react with nitrogen. Spaces into which radioactive sodium could be released in the case of leaks are therefore filled with this gas.

The gas spaces within the sodium system are filled with argon or helium as cover gas. It protects the free sodium surface, in case of a leak, from any contact with air. Any manipulation of components outside the sodium system must take place under an inert atmosphere and must be remotely controlled. This is particularly important for the handling of fuel elements outside the reactor.

In the non-radioactive sodium zones, provisions are made to limit and extinguish sodium fires that may have been caused by the release of sodium during accidents. Metal powders are used for fire extinguishing purposes.

Because sodium is liquid only at temperatures above 100 °C it is necessary to provide heating for sodium-carrying components. This is a specific feature of FBRs. For vessels and pipes, this is generally effected by electrical resistance heating cables, which are installed on the surface of the coolant enclosure under the insulation. In the case of larger components – such as the reactor vessel – heating is normally only necessary during initial commissioning. This is done with hot nitrogen.

Having a melting point significantly above ambient temperature the sodium allows 'freeze seals' for stem sealing of sodium valves. At a certain distance from the valve body the sodium in the vertical gap becomes solid by which the seal is formed.

Sodium forms active isotopes in the neutron flux. However, only the two strong gamma emitters ^{24}Na and ^{22}Na are of importance regarding activity release, shielding and accessibility. ^{24}Na has a half-life of 15 h, ^{22}Na has a half-life of 2.6 years. Because there are large quantities

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of sodium in the system, the primary system components must be heavily shielded. On account of the radioactive sodium isotopes, access is possible only 10–14 days after the reactor has been shut down. The decisive isotope ^{24}Na has, for the most part, decayed by then. However, accessibility could still be limited. This is because of activated material, stemming from defected fuel elements or fuel pin corrosion, which has been transported into the systems and deposited on their surfaces. These considerations are relevant primarily for a loop arrangement.

2.3. *Detection and limitation of sodium impurities*

Impurities in the sodium, in particular oxygen, carbon and hydrogen are of major importance for two reasons: firstly, because of their potential influence on the structural material; secondly, because there is the danger that the cooling channels in the core become blocked by their sodium compounds. To avoid these potential damages to the system the oxygen content of the sodium must be controlled and kept below 2–5 p.p.m. (by mass).

The solubility of Na_2O and NaOH depends on the temperature. This fact is used for monitoring and removing oxygen from the sodium system. The two compounds together with NaH can, therefore, be separated from the sodium by cooling it down. During the cooling process also other impurities such as calcium precipitate. This principle is used, on one side, in cold traps for purification purposes and on the other side in 'plugging' meters for measuring the oxygen content. In the cold traps, at low sodium temperatures, the oxide and other impurities deposit on packings.

Hydrogen, even if dissolved in sodium, will penetrate thin nickel or palladium membranes in contact with the sodium on one side, if a vacuum is applied on their rear side. The fact that only the hydrogen will penetrate the membrane is used to measure its concentration in the sodium.

The detection sensitivity of this method is far below 1 p.p.m. Apart from determining hydrogen impurities, this technique is applied for the early detection of hydrogen from small steam generator leakages. Early hydrogen detection enables measures to be taken for the prevention of larger sodium–water reactions. These would otherwise arise as a consequence of increasing leak size.

2.4. *Structural mechanics of sodium systems*

The FBR temperature conditions and coolant properties lead to special design requirements for the components. The selection of structural materials for FBRs must, therefore, take account of the following factors, which are in addition to the basic selection factors applied to LWRs:

(i) time-dependent strength and fatigue behaviour because of the high operating temperatures;

(ii) fast-energy spectrum of the neuron flux at high doses;

(iii) sodium-specific influences like material eroding mechanisms, depositing corrosion mechanisms and selective corrosion effects of individual alloying elements.

The mass transfer processes, governed by variables such as temperature gradients in the system and material quality, are strongly dependent on sodium impurities. Of particular concern are oxygen and carbon.

The products formed as a result of corrosive effects are distributed in the system by the sodium. There it acts as the carrier medium. Corrosive and deposition actions occur selectively for individual alloying elements of the materials. These processes can have, in some cases, a significant effect on the materials' properties.

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Stress and strain considerations must also take account of the additional loads resulting from temperature differences. There are, on the one side, steady-state temperature differences as, for example, the inevitable temperature drop between the hot coolant enclosure and its cold support points. On the other side, there are transient temperature differences caused by start-up and shutdown procedures or accidents. Because of the good heat transfer properties of sodium, changes in coolant temperature during transient operation are very quickly transferred to the structures. The fast transfer of the temperature into the structure is the cause for additional stresses.

Deformation-induced loads, caused by internal loading or forced deformations, can cause failures as a result of fatigue or creep damage. Depending on the number and duration of events, the yield strength may be exceeded for the combination of primary and secondary stresses. The calculation methods must therefore take account of possible interactions between time-independent and time-dependent material behaviour.

Consequently, for the stress analysis of highly loaded components, the elastic analyses in LWR design must be supplemented by inelastic analyses. In particular, relaxation and creep influences must be considered. To keep the loads within acceptable limits, component design is subject to specific design requirements.

(i) Consideration of clearances or specially designed expansion elements. This is to absorb thermally induced differential expansions at both steady-state and transient temperature conditions.

(ii) Avoidance of large material accumulations and sudden differences in stiffness. Points of inhomogeneity, being subject to fast temperature changes, must be protected by thermal sleeves.

3. Main sodium components

3.1. *Pumps*

Liquid sodium can be transported by electromagnetic pumps because of its electrical properties. A magnetic field applied around a sodium-carrying pipe makes the sodium flow. This principle is used for pumps with smaller throughputs and lower delivery heads for, e.g. auxiliary and emergency cooling systems. It is also applied for measuring the flow rate of sodium flowing through pipes.

Because the hydraulic properties of sodium are similar to those of water it is, basically, possible to use similar hydraulic machines.

Mechanical pumps – generally of single-stage design – are used for the main coolant flow. To avoid under-sodium friction problems, these pumps are equipped with a lower shaft supporting device. This device is equipped with a hydrostatic bearing. The bearing is supplied with sodium from the pump pressure side and leakages from it are returned to the low-pressure side of the loop. Such a pump may have, typically, a flow of some 5000 $m^3 h^{-1}$ at a pressure difference of 11 bar†.

3.2. *Intermediate heat exchangers*

For reasons of availability and safety, very stringest tightness requirements are imposed on the IHXs for all operating and accident conditions. IHXs in sodium systems are designed as counterflow components to keep stresses induced by temperature differences at a low level.

It is necessary to minimize the primary-side pressure loss and to absorb the secondary circuit

† 1 bar = 10^5 Pa.

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pressure arising from sodium–water reactions in the steam generator. Therefore, generally, a design is selected in which the primary sodium flows on the shell side and the secondary sodium on the tube side of the bundle. The heat transfer coefficient between primary and secondary sodium is of the order of 5 kW m^{-2} K^{-1}. The average logarithmic temperature difference is around 30–40 K. Heat transfer surfaces of the order of 5 m^2 MW_t^{-1} are therefore required.

The most important requirements for the IHX design are control of the thermal expansion between tube bundle and shell and a uniform temperature distribution inside the bundle. In addition, compensation of the expansion differences inside the bundle must be catered for.

3.3. *Steam generators*

With reference to plant availability, the steam generators are key components in power plants with sodium-cooled reactors. Apart from the basic requirements for heat-exchanging components, the consideration of sodium–water reac..ons and hydrodynamic and material stability on the water–steam side are of utmost importance. As in conventional systems, recirculation or once-through processes are possible on the water–steam side. In the once-through system, evaporator and superheated can either be designed as one unit or as separate units. In the latter case, water separators must be provided to prevent shifting of the evaporation point during part-load operation.

Figure 4 illustrates various types of sodium-heated once-through steam generators. The specific problem of sodium-heated steam generators is to control the sodium–water reaction in the case of a leak.

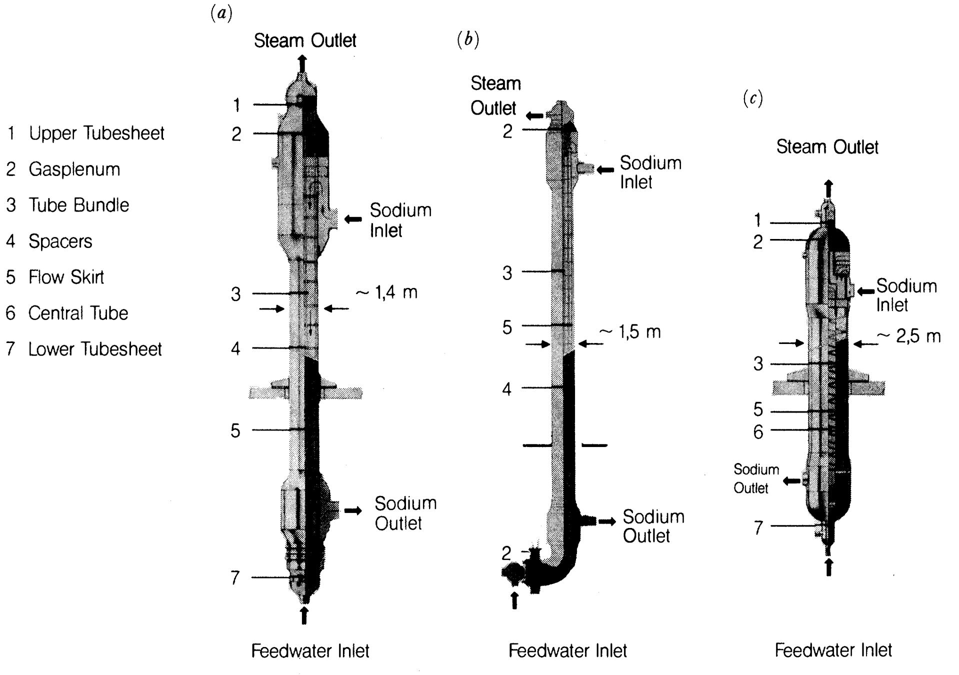

FIGURE 4. Three designs of FBR steam generator (SG) (*ca.* 600 MW). (*a*) Straight-tube SG, height *ca.* 36 m; (*b*) J-tube SG, height *ca.* 33 m; (*c*) helical-coiled SG, height *ca.* 24 m.

Thanks to the non-active intermediate loop and the spatially redundant design of the heat transfer systems, only the IHX integrity must be assured. The integrity is vital for avoiding nuclear safety problems, e.g. problems with residual heat removal and activity enclosure.

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However, damage resulting from sodium–water reactions should be limited for reasons of plant availability.

The reaction sequence is governed by the transport paths of the reactants. Because of the higher pressure on the water side, the water flows into the sodium. Large quantities of hydrogen are released during the reaction. Figure 5 shows typical pressure courses of two different experimental sodium–water reaction cases. The 'guillotine rupture' case is characterized by incipient high-pressure peaks of very short duration. After some 50 ms the pressure has dropped to a quasi-static value of a few tens of bars.

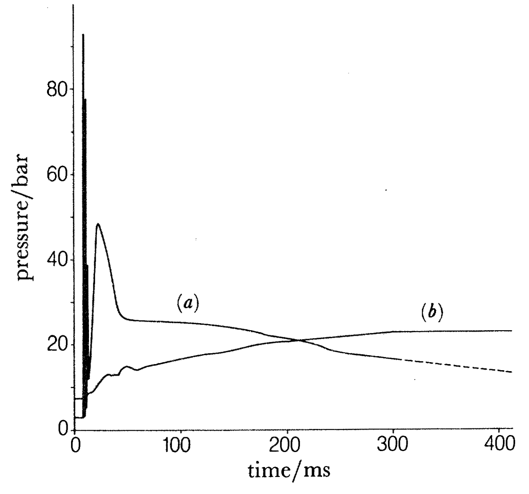

FIGURE 5. Pressure course during sodium–water reactions experiments. (*a*) Instantaneous tube 'guillotine' rupture; (*b*) tube rupture from preceding small leak.

In the other experiment a tube rupture is provoked by overheating. This overheating is caused by the sodium–water reaction from a preceding small tube leak. Here the hydrogen, already present from the small-leak reaction, has a damping function. So the pressure increases rather smoothly and there are no pronounced pressure peaks.

The experiments serve for the understanding of the phenomena and for computer code validation purposes. In a realistic steam generator the maximum pressure peak will be much lower. However, the secondary circuit must be protected against these kinds of pressure waves. This is especially true for the IHX that forms the boundary between the primary and the secondary sodium.

The steam generators must therefore be equipped with a pressure relief system. In all modern steam generators such a system consists of fast-responding rupture disks at the sodium side of the steam generators and relief lines connected to cyclones. In the latter the reaction products are separated and the hydrogen that has formed is allowed to escape.

The American sodium-cooled Experimental Breeder Reactor II is equipped with a different type of steam generator. It features double-walled tubes, the interspaces of which are monitored for leakages. Such a design may make, in the long run, the non-active intermediate sodium circuit superfluous. This approach is still under discussion in the U.S.A.

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Conclusions

There are, on the one hand, many engineering problems which must be tackled when designing a sodium-cooled FBR. On the other hand, this type of reactor has many features that facilitate the design. Experience up to now has shown that this technology is viable.

Because of the nature of this complex technology, some drawbacks have been experienced. Major R&D steps and considerable design efforts are still necessary to make the fast breeder a mature energy source which, regarding operability and economy, can compete with the existing LWRS.

Discussion

M. Y. H. Bangash (*Middlesex Polytechnic, U.K.*). Are there any practical measures that could be taken to avoid overpressurization of the primary sodium tank by, for example, a mini explosion. I note from the presentation that inelastic analyses were used to assess thermal creep effects but do not believe that this alone is adequate. Will some assessment of failure through additional causes such as overpressurization be necessary to estimate the safety margins?

M. Köhler. The only possible cause of overpressurization is the postulated 'Bethe–Tait' incident. The view held in Germany is that, with reasonable assumptions for the energy yield, this would be contained in the primary system even allowing for flaws. This view is based on analysis and experimental evidence. The inelastic analyses that I mentioned are related to lifetime analyses of the high-temperature components of the reactor. The primary vessel is maintained at temperatures below the creep range.

J. D. Lewins (*Cambridge University, U.K.*) What are the origins of the rapid changes of temperature gradients in fast reactor components?

M. Köhler. The most important shocks result from rapid shutdown, and are delivered to the above core structure (ACS). These are changes from normal core outlet temperature to something near the sodium inlet temperature, a difference of up to 170 °C. Thermal striping provides another mechanism, again affecting the ACS. This arises as incompletely mixed streams of hot and cold sodium impinge on the surface of a structure. Shifts in the relative positions of the streams or slight turbulence can lead to temperature variations of several tens of degrees with a frequency of the order of 1 Hz.

F. J. Barclay (*Energy Consultant, London, U.K.*). Returning to the subject of positive void coefficients in the core centre of large fast reactors, are any measures available to eliminate the possibility of voiding, other than the expensive approach of orificing all subassemblies and increasing pumping power? (The object would be to distance the system from flow instability, which could lead to voiding, and be triggered by a momentary flow reduction, due for example to a vigorous release of fission gas. Flow instability can be eliminated if, as in PWRS, subassemblies have no wrappers.)

M. Köhler. The inherent properties of sodium coolant require a large mismatch between power and cooling for voiding to occur. With channel blockage local sodium boiling and

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voidage is possible, but this would have little effect on overall reactivity. To achieve a large reactivity increase it would be necessary to void a complete zone of the core. This can only be envisaged for an uncontrolled reactivity excursion. However, diverse shutdown systems are incorporated to overcome this possibility. As a result the probability of this sort of event is so small that it is outside the design basis for fast reactors.

D. A. Davis (*CEGB*, *London*, *U.K.*). If as the paper implies there is no major advantage in favour of the pool design, why are most current designs of this type?

M. Köhler. The majority of design effort for large reactors has been invested in the pool reactor which has created a momentum in its favour. Specifically, the construction of Superphénix has resulted in a lead in the experience of pool reactors. In Germany the loop design was preferred for SNR300 but for SNR2, the pool was chosen to allow a European harmonization.

P. Dastidar (*IAEA*, *Vienna*, *Austria*). Light water reactors had become complex to take care of many phenomena, some thought of at the time of design and some based upon feedback from experience. Now there is an attempt to make them simple to achieve high reliability and safety.

Fast reactors are complex because of the use of sophisticated materials and processes which have to be taken care of by sophisticated systems. Is there an attempt to make systems simple and yet effective based upon experience?

M. Köhler. Good design makes fast reactors as simple as LWRs. The use of sodium coolant through its inherent heat transfer properties offers significant advantages in terms of design simplicity. Pressureless systems and decay heat removal through natural circulation are good examples.

J. Spence (*University of Strathclyde*, *U.K.*) About 10 years ago I was more involved with fast breeder structural mechanics – specifically, secondary piping systems – than I am now. At that time a main inhibiting factor was the difficulty of inelastic analysis and considerable effort was devoted to developing simplified (and consequently less certain) methods.

In the past few years our capability to perform inelastic analysis has increased manyfold; it is also cheaper. Has this made the design situation more straightforward and has it allowed more creative design? Or do the associated problems of basic material behaviour, constitutive relations, design limits or rules and possible failure modes pose even greater problems than the inelastic analysis? Where does Mr Köhler consider the key problems lie and what research might be targeted to alleviate any difficulty?

M. Köhler. For some components like the ACS the design target was only achievable by using inelastic analysis. For other parts, e.g. secondary piping, considerable design simplifications were possible by adopting inelastic design method.

Now the attention and the R&D needs are much more shifted to the real limits of the material, more specifically, to the welds for combined creep and fatigue loading together with the assumptions of postulated defects in the structure sometimes under irradiation.

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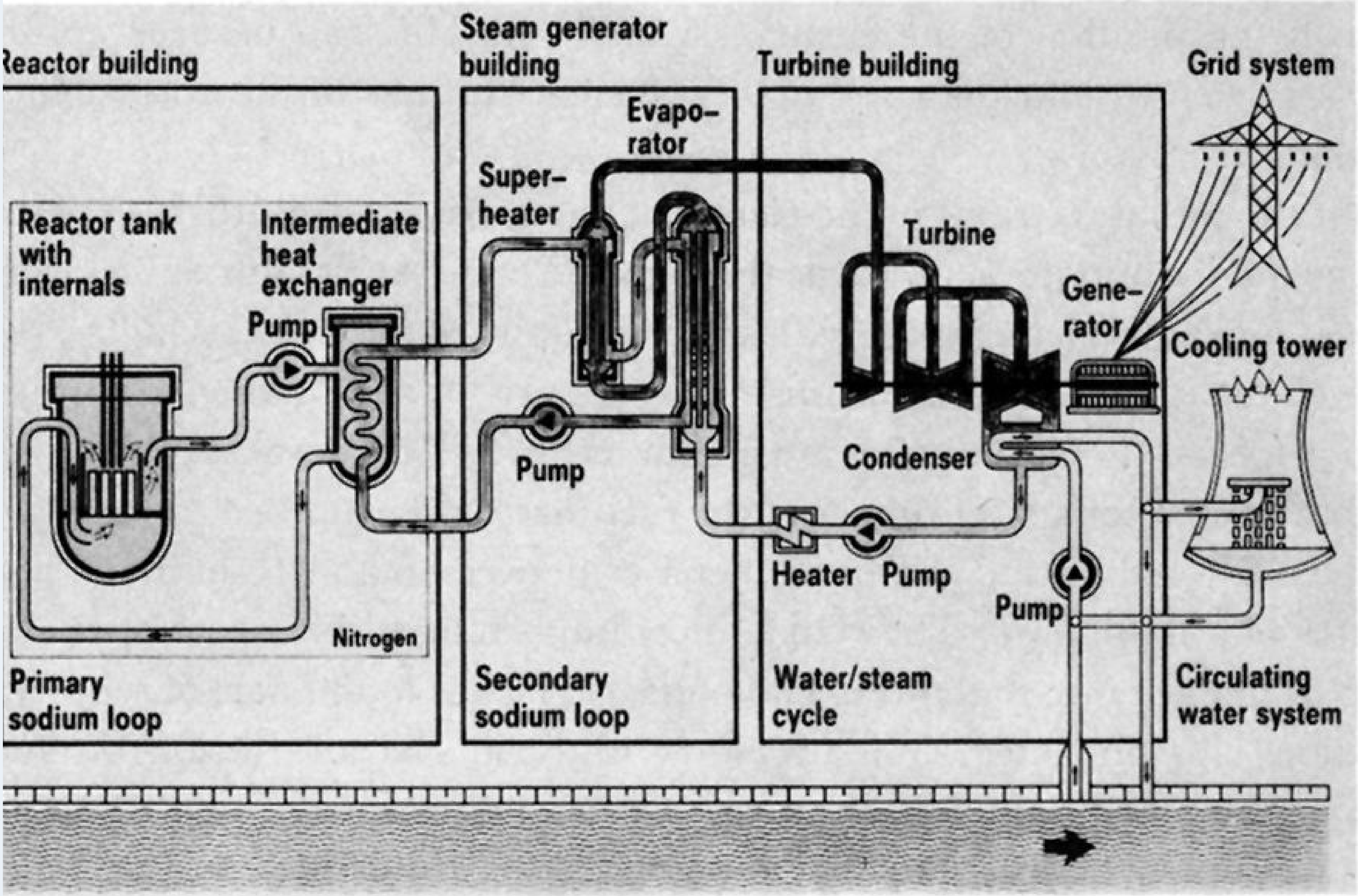

FIGURE 2. Basic arrangement (loop concept) of FBR.

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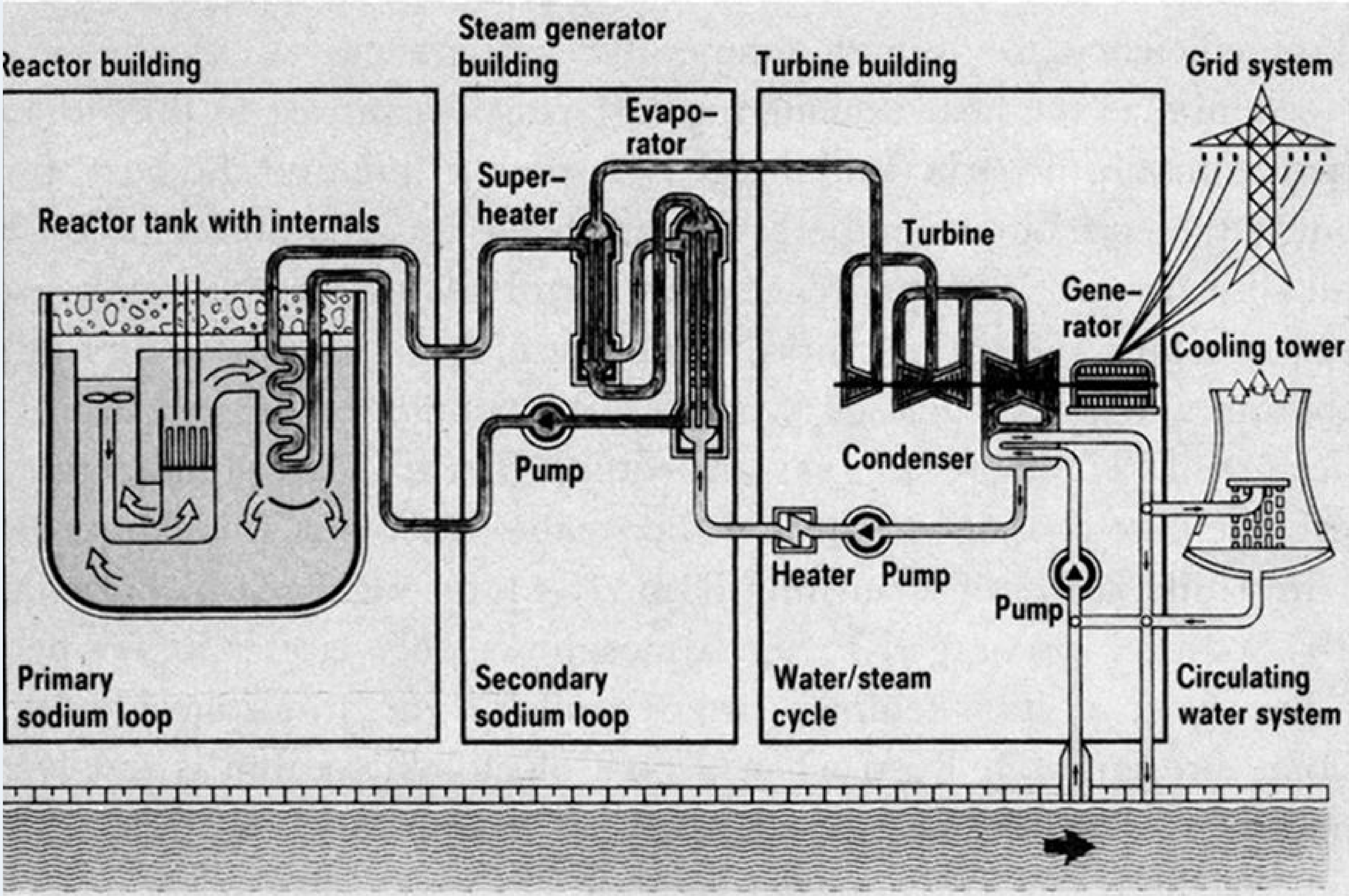

FIGURE 3. Pool arrangement of primary circuit in FBR.

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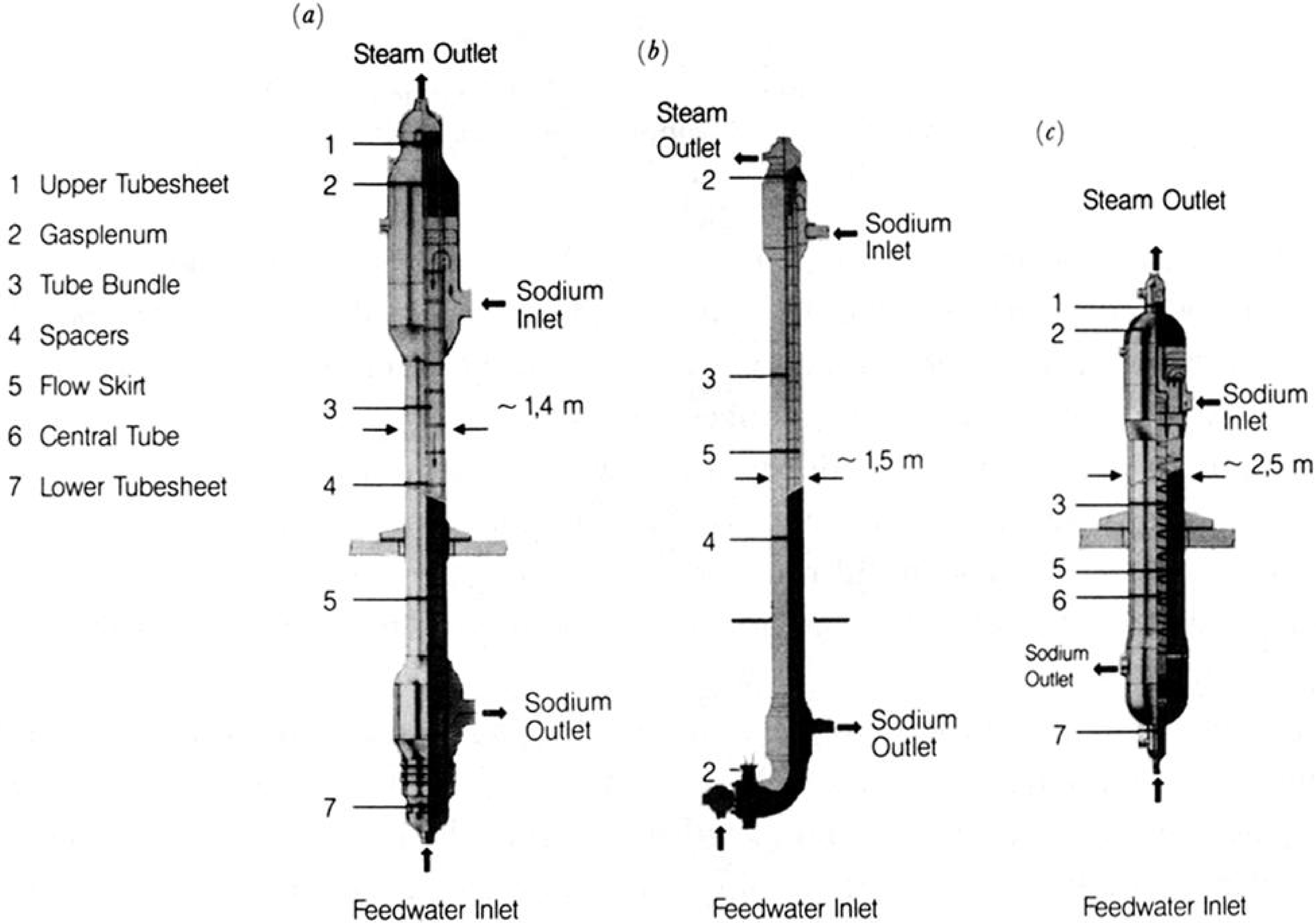

IGURE 4. Three designs of FBR steam generator (SG) (*ca.* 600 MW). (*a*) Straight-tube SG, height *ca.* 36 m; (*b*) J-tube SG, height *ca.* 33 m; (*c*) helical-coiled SG, height *ca.* 24 m.

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