LIQUID METAL COOLED FAST BREEDER REACTOR DEVELOPMENT AND ITS MECHANICAL-STRUCTURAL REQUIREMENTS AND PROBLEMS

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After a brief survey about the history and the present status of liquid metal cooled fast reactors the paper explains the specific properties of liquid sodium and the principal layout of sodium-cooled fast reactor plants. The most important mechanical-structural requirements and problems are broken down into fields of sodium containment, core structure, and special mechanisms to be operated in sodium and the cover gas atmosphere.

With respect to sodium containment, the main difficulties raised concern mechanical and thermal cycling stresses during transient operational conditions and creep effects appearing in the walls which are subjected to high temperatures. The requirements for the core structure have become extremely complicated due to swelling of the structural material at high neutron doses. In the field of mechanisms special attention must be given to all questions of material pairing with respect to diffusion welding, friction and wear.

1. Introduction

The development of liquid metal cooled fast-breeder reactors started with the construction and operation of quite a number of smaller plants. The first plant, EBR-I [1, 2] was commissioned as early as in 1951 in the United States of America. It was the first nuclear reactor from which the heat generated was converted into useful electricity. In 1959 EBR-I was followed by BR-5 [3, 4] in the U.S.S.R. and the Dounreay plant [5, 6] in the U.K. Four years later, in 1963, the reactors EBR-II [7, 8] and FERMI [9, 10] started operation in the USA. Then followed in 1967 Rapsodie [11, 12] in France, in 1969 SEFOR [13, 14] in the USA and in 1970 BR-60 [15, 16] in the U.S.S.R.. Finally, in 1971 KNK [17, 18] became critical in Germany. However, this reactor operates in the beginning with a thermal-neutron spectrum.

After this first phase of 20 years duration a number of larger-size liquid metal cooled fast breeders are now under construction, which are real forerunners of future central power stations.

These are PFR [19] in the U.K. and PHENIX [20] in France, both with a thermal power output of 600

MW, as well as in the U.S.S.R. BN 350 [21] with 1000 and BN 600 [22] with 1500 MW thermal power, respectively. Besides, FFTF [23] with 400 MW thermal power, which is being erected in the United States, must be included into this line. In addition, one of the big industrial companies in the United States will certainly start construction of another plant within the next years. Finally, the erection of the 300-MW(e) SNR [24] in Europe is near at hand which is part of the DeBeNeLux Fast Breeder Project.

It will depend largely on the reliability of these plants how fast this new reactor system will make its way into the electric power industry on a larger scale.

The technical problems to be solved in the development of fast sodium cooled reactors are completely different from those associated with the H_2O -cooled or gas-cooled reactors. In addition to the development of the reactor core proper, together with control rods and fueling machines, much more basic research work has to be done with respect to the components of heat-transfer systems and the numerous auxiliary systems. A completely new technology must be developed for the liquid metal, i.e. for the liquid sodium. This is a consequence of the specific properties of this kind of fluid.

2. Properties of liquid sodium

The predominant feature of liquid sodium is the very high thermal conductivity which is greater by a factor of 20 than that of water [25]. It has a decisive influence on the design of the system. Therefore, the high heat-transfer coefficients up to $13 \text{ W/cm}^2\text{ }^\circ\text{C}$, required for fast reactors, were realized at comparatively low coolant velocities. Consequently, liquid sodium yields very high heat fluxes even with small temperature differences. This is a decisive reason for the use of sodium as a coolant in fast reactors for which cores must be designed for very high power densities for reasons of economy. Typical values are 300–500 kW/l of core volume.

On the other hand, the high heat-transfer coefficient raises some new problems with respect to the cooling circuits. Any temperature change of the coolant can lead to serious thermal stresses in the components, also at low coolant velocities.

Another property of sodium is its high boiling point of 883°C at atmospheric pressure. This means that the heat-transfer systems can be operated practically without overpressure. This feature constitutes a basic difference as compared to LWR's, especially if one considers the high expenditure required in that case to avoid and limit the consequences, respectively, of a loss-of-coolant accident. Excess pressures in a sodium-cooled reactor are necessary only to overcome the pressure drops which are inevitably present in the cooling circuit. This absence of overpressure allows an almost unlimited extension of the plant size resulting in a remarkable cost decrease with large power outputs.

A certain drawback is the solidification point of about 100°C . It requires a preheating system for all sodium-carrying plant components. On the other hand, the high solidification temperature offers also some advantages. In sodium-cooled plants it is, for instance, an easy and safe measure to freeze a pipe locally in cases of repair, so that a perfect seal is achieved. Also, shafts or push rods, such as valve spindles, penetrating into the sodium filled space from the outside, can be sealed easily and safely by so-called freeze glands, even when moved. Values of larger sizes are designed almost exclusively on this principle.

Another feature of sodium is its high electric conductivity. This allows to use direct-contact instruments in some cases, e.g. level gauges or leak detectors, and, above all, the use of electromagnetic flowmeters and pumps. Although the efficiency of these pumps is rather low, they are a very simple and reliable device without moving parts and with very fine control-behaviour, which could be used successfully in a number of auxiliary systems of the reactor.

A clear drawback is the great chemical affinity of sodium to oxygen, e.g., to air, and to water and steam, respectively. This calls for a protective inert-gas atmosphere covering all sodium surfaces and for specific safety precautions in sodium-heated steam generators. On account of this property, the use of sodium on a technical scale still meets with a certain reserve in some quarters.

The great chemical affinity to oxygen has also the consequence that the oxide layers are dissolved which are normally present on the surfaces of structural materials. This is of great importance to all high-precision and mechanical parts operating in sodium, especially with respect to the friction, wear and diffusion welding behaviour of contact surfaces.

The corrosion rate in sodium is very low in the temperature ranges concerned up to about 600°C . It plays a major part only in the relatively thin-walled fuel-element cladding tubes.

In characterizing liquid sodium it should be mentioned that the density and viscosity are similar to water. Therefore, all hydraulic questions relating to sodium plants can be investigated experimentally in water.

Compared to water the opacity of sodium is also of some importance. If there exists an interest to carry out the refueling process under direct view, similar to that in LWR's, a special reactor design is required. It must allow to lift off the reactor tank cover and to lower the coolant level to such an extent that the heads of the fuel elements become visible. The already mentioned SEFOR follows this principle. In general, direct view is not considered for sodium-cooled reactors. The correct location of components in sodium, such as fuel elements or refueling tools, is checked with the help of mechanical devices or by using the ultrasonic technique. Especially the latter technique seems to be attractive for this purpose [26].

Fig. 1. Principal layout of sodium-cooled fast reactor plants and some mechanical-structural requirements.

3. Layout of sodium-cooled fast reactor plants and mechanical-structural requirements and problems

The principal layout of a sodium cooled fast reactor plant is shown in fig. 1; some of the most important mechanical-structural requirements are indicated there.

All present plants are characterized by the series arrangement of the primary and secondary sodium cooling circuits and the tertiary water-steam circuit. The non-radioactive secondary circuit penetrates through the wall of the containment; it protects the reactor from the consequences of faults occurring in the sodium-heated steam generator and keeps the effects of disturbances in the reactor away from the steam generator.

Some of such incidents can exert dynamic loads on the components of the plant. For example, a sodium-water reaction cannot be excluded with absolute certainty in the steam generator, since, for economic reasons, single-walled tubes must be used for heat transfer in large-scale plants. The pressure increase in the reaction zone is mainly determined in this case by the great thermal effect and the generation of hydrogen. In case of major leakages the liquid

sodium will be blown off the leak point. Burst membranes are provided to eject the sodium and the reaction products into a pressure relief system. Experiments carried out in tube bundles have shown that tubes adjacent to the leakage point are not affected significantly so that a failure propagation in the bundle can be excluded [27]. However, the pressure peaks produced by such a reaction must be taken into account in dimensioning the steam generator, the secondary circuit and the intermediate heat exchanger.

In designing and dimensioning the primary circuit, the overriding point is to make sure that leaks can be excluded which would result in such a sodium loss that the cooling of the core and the fuel elements, respectively, will fail. Also the tightness of the containment system must be preserved. Among others, this requires that the tube bundle of the intermediate heat exchanger can neither be disturbed by loadings exerted from the primary side. Up to now, it has been common practice to consider in addition to normal operational loading also such dynamic loadings which might be produced in a serious hypothetical accident in which the core is partially destructed. The time schedule of such excursions lie in the range

of milliseconds; therefore shock-wave effects must not be considered under any circumstances.

However, problems of a new type are encountered already in the design and calculation of the external shell of the sodium system for stresses under normal operating conditions. In addition to the usual problems associated with thermal expansions, the main difficulties raised here concern mechanical and thermal cycling stresses arising during transient operational conditions. Moreover, creep effects must be taken into account which appear in the walls subjected to high temperatures. Also the influence of sodium on structural material, such as carbon migration, which may act upon the properties of materials, has to be considered.

Another main area of interest associated with the mechanical-structural requirements is raised by the reactor core itself. A precise core structure of long-time stability with a predetermined thermal-expansion behaviour must be designed, since the reactivity of a fast reactor core is very sensitive to a change of volume, geometry, composition or density. This task has become extremely complicated as a result of swelling of the structural material which takes place at high neutron doses.

Finally, another unsettled complex of problems refers to the mechanisms to be operated in sodium and the cover gas atmosphere, respectively, which is saturated by sodium vapor. They include the control rod drives, refueling devices, and core instrumentation with their guiding and supporting elements. These systems require practically all the usual machine elements, such as bearings, guides, threaded spindles, but also piston rings for sealing etc. Special attention must be given here to all questions of material pairing with respect to diffusion welding, friction and wear.

3.1. Sodium containment

The requirements with respect to the primary sodium containment are different to some degree depending on its design concept [28]. It is well-known that, presently, two different systems are used for sodium-cooled reactors: the loop design and the pool design. In the loop design, which is represented in fig. 1, the reactor, the intermediate heat exchanger and the circulating pump are interconnected by pipes. In a 1000 MW(e) plant with three parallel circuits these pipes have a diameter of approximately

1200 mm and a length of more than 100 m per primary loop, in order to compensate for the thermal expansion between the three fixed points.

In the pool design the three main components — reactor, intermediate heat exchanger, and circulating pump — are arranged much closer in a common tank filled with sodium. At 1000 MW(e) the diameter of the tank is about 17 m and its height 14 m. Fig. 2 shows as an example the already mentioned British PFR [29] which is designed according to this principle. The sodium in the tank is guided through baffles and short pipes which are equipped with sliding connections to make up for thermal expansion and with a thermal insulation placed in sodium to make up for thermal stresses. These prerequisites as well as the large tank with its upper cover raise new problems too, which will not be discussed here in more detail.

Some additional information should be given about the large diameter sodium pipes which are necessarily required for the secondary circuits of both designs [30]. The extraordinary stress conditions call for more detailed calculations than usual in pipe work design. Besides the stresses due to internal pressure, the strain resulting from thermal expansion and the thermal stresses occurring with changing temperatures, have to be taken into account, and, above all, fatigue phenomena due to mechanical or thermal cycling stresses as well as creep effects appearing in the high temperatures sections. In three-dimensional piping systems operated under low pressure these creep effects may, e.g., result in structural instability, in particular, if a deformation of the circular cross section develops in the bends due to momenta, since this effect is accompanied by a rapid decrease of moments of inertia and resistance, respectively.

Up to now, only a stepwise calculation of the different problems has been possible. Individual results must then be interpreted together and added up; this applies also if the assumptions and approximations underlying the individual steps, strictly speaking, do not allow such treatment. For example, the methods used so far to calculate the flexibility of a pipework system generally assume elastic behaviour in all regions. A method would be much appreciated which allows to include the inelastic behaviour of individual sections, since this could significantly alter the distribution of forces and moments over the structure.

Fig. 2. The Dounreay PFR [29].

Fig. 3. Effect of temperature and time of exposure on the room-temperature impact properties of stainless steel [33]. Left-hand side: Type 304 (X5 Cr Ni 189); Right-hand side: Type 316 (X5 Cr Bi Mo 18 12).

The calculation of the stresses and strains occurring in such pipework system in a single comprehensive procedure is not yet possible. Such a calculation would have to take into account the relaxation of residual stresses with time in addition to plastic and creep strains, so that the true stress-strain-time history can be determined for every individual section. The present USA Nuclear Power Piping Code B 31.7 [31] with its subsequent Code Case Interpretation 1331 [32] is not adequate for this type of pipework which is expected to function for several decades at high operating temperatures and under extraordinary thermal stress cycles. This is verified by the Design Guide for LMFBR Sodium Piping published in spring 1971 [33]. It was prepared on behalf of the U.S. Atomic Energy Commission in an attempt to fill this gap in a first approximation. Among other recommendations, this guide postulates for the reduction of permissible high-temperature stress values indicated in the present codes, since these data had been extrapolated from experiments too short as against the actual service life of a power plant. Besides, the present code makes no difference between short- and long-term cyclic

stresses as may occur during operational transients and the start up or cool-down period as a result of differential thermal expansions. This distinction appears to become necessary under present conditions because the concept employed by the code permits some relaxation of stress during the hot portion of the thermal cycle. For sodium pipings subjected to numerous short-duration thermal cycles this may be a hazardous assumption. Cyclic strains of greatly different magnitudes are experienced here in varying and probably random order of application. This may have a significant effect on the overall fatigue life.

By contrast, the experimental fatigue data used so far for direct design procedures, e.g., according to the ASME Code Section III [34], were obtained by controlling the strain between fixed limits while cycling the specimen periodically until failure is observed.

Also a prolonged exposure at elevated temperatures of the piping material may result in gradual structural changes and thus influence material data responsible for the strength, creep and rupture behaviour of a piping. The room-temperature impact

Fig. 4. Bending and support of fuel elements in sodium-cooled fast reactors.

properties of the three materials, recommended for sodium pipings in the Design Code mentioned above should be considered here as an example: 304 SS; 316 SS, and $2\frac{1}{4}\text{Cr} - 1\text{Mo}$. While the toughness of the ferritic material $2\frac{1}{4}\text{Cr} - 1\text{Mo}$ remains unchanged or even increases after a prolonged time of operation at high temperatures, the austenitic steels are characterized by a remarkable reduction of the notch impact strength. This reduction is represented in fig. 3. After 10 000 hours i.e., about 15 months of operation at 650°C (1200°F) the notch impact strength of 304 SS drops to some 50%, that of 316 SS to about 25%. With regard to the long service life of a plant this decrease of the room-temperature impact strength must be duly considered.

These short descriptions have shown that the designer of a large sodium piping system designed for long-term operation at high temperatures, assumes an immense responsibility with respect to the layout, the selection of materials, the methods of calculation and skillful interpretation of individual results. For the time being, these difficulties can be diminished only by reducing or eliminating already in the conceptual stage as many problems as possible and by keeping the whole system very simple. It could be worthwhile also to oversize such components for which adequate stress calculation is not possible. However, the increasing influence of thermal shocks

with larger wall thicknesses must be taken into account.

In general, the design and construction of the large piping systems required for sodium-cooled fast breeders is possible, based on the knowledge and methods available today. However, further improvements or extensions of the presently available stress calculation methods and additional research concerning the material behaviour under actual operating conditions seems to be necessary. As an additional safety measure, a regular in-service inspection could be taken into account for piping systems of this type, which is similar to the in-service surveillance of reactor pressure vessels [35].

3.2. Core structure

The majority of problems to be solved with respect to the reactor core are of principal importance to fast breeder reactors, i.e., they are not only typical of sodium cooling.

The mechanical structure of a fast reactor core is, very roughly speaking, similar to that of the core of a thermal light-water reactor. Small stainless steel tubes of about 7 mm diameter, tightly welded on both ends and filled with U-Pu mixed oxide pellets, are used as fuel pins. The fuel pins are assembled in bundles — the fuel elements — and a larger number of these fuel elements constitute the reactor core. For reasons of neutron physics, the fuel pins should be

packed as closely as possible. Therefore, in sodium-cooled breeder reactors, a triangular arrangement has been generally adopted resulting in a hexagonal cross section of the fuel elements. The individual fuel elements are equipped with wrapper tubes, the so-called fuel element boxes, allowing the coolant flow to be calibrated individually. This is necessary because of the dissimilar amounts of heat released in the individual radial zones of the reactor core and in view of the large reheating ranges in order to obtain approximately identical outlet temperatures.

In fast reactors these fuel element boxes cause some structural difficulties. The radial fission rate distribution results in an unequal distribution of temperatures over the cross section of the fuel elements, and, hence, of unequal temperatures at the walls of the boxes. Consequently, the elements may be bent due to thermal expansion of the material. This mechanism is represented under a) in fig. 4. On account of the strong feedback of reactivity effect and safe functioning of the control rods, the resultant great deflection of fuel element heads cannot be tolerated. Therefore, fuel element deformation is limited by providing mechanical supports. Under b), the EBR-II system is shown with two supporting planes. The position of the supporting planes and the rigidity of the boxes in the axial direction is chosen in such a way that increasing reactor power, i.e., increasing deformation, moves the fuel to the outside, i.e., decreases the reactivity.

Due to the limit set to deformation, the natural results are mechanical stresses in the box. Thermal expansion in the horizontal direction is taken into account by providing initial gaps between the supporting points.

Designs according to this principle have been shown to give satisfactory results in fast reactors of smaller size, such as mentioned at the beginning.

For reactors of larger size, this design principle is no longer applicable automatically because of structural material swelling [36, 37]. Structural material swelling occurs in large reactors as a result of the lower fuel enrichment and the corresponding higher neutron flux which leads to a higher radiation exposure of the structural material also if the burnup of the fuel remains unchanged. The fluence to be taken into account in this case is about 3×10^{23} nvt or higher. This results in an increase in volume of

some percent up to a maximum of 5 to 8% for austenitic steels which are usually chosen for the core design. The amount of swelling depends on the dose, composition and preliminary treatment of the material and, above all, upon the irradiation temperature. The maximum values were observed at about 500 to 600°C, which is the temperature range applicable to many core components. Below 350°C and above 650°C swelling is not expected.

For the design shown under b) of fig. 4, structural swelling has two distinctive results: Because of the intermediate supporting plane provided the core will expand radially during the course of burnup, and due to the dissimilar temperatures and radiation exposures of the box walls, swelling leads to an additional bending which increases with time and is superimposed to the bending originated by thermal expansion. The resultant stresses and strains of this two modes would reach unacceptable levels.

Nevertheless, there exists also a process in the opposite direction, which has to be mentioned in this respect: irradiation-induced creep. However, only very few information is available at present about this phenomena [38]. An interesting indication results from the experience obtained with one test fuel assembly which was irradiated up to a fluence of 8×10^{22} nvt in the EBR-II, i.e., under the supporting conditions shown under b) [39]. Parallel to the expansion of the fuel element box by the swelling process an opposite creep process was observed, by which the separating buttons, originally pressed outwards from the box wall, were flattened by a corresponding amount without any rupture of the material. For the very complex structure of the reactor core, which consists of a large number of individual parts, it is unlikely, however, that it would be possible to balance the two processes, namely irradiation induced swelling and creep, precisely enough so that an adequate compensation could be expected. In this respect, one has to keep in mind also that during the refueling process of large reactors only a fraction of the total number of the fuel elements are replaced and it must be technically feasible to replace any defective fuel element in the core by a new one at any time.

The design proposals, which take into account the swelling of structural material from the beginning, are, therefore, characterized by two features: they provide larger gaps between adjacent fuel elements to

make up for transversal swelling and they shift the supporting planes out of the swelling zone. If the fuel elements are extended accordingly, a deformation mechanism is obtained which also causes a reduction of reactivity with increasing reactor power. An example is given under c) which represents the design discussed for the U.S. FFTF reactor [40]

For such a system, additional radial clamping mechanisms are often employed, which act upon the supporting planes [41, 42]. They can be actuated mechanically or hydraulically. These clamping systems serve the purpose of establishing a closely packed array of the fuel elements, so that all contact planes touch each other before startup of the reactor and the fuel elements are guided properly from the beginning also under transient operating conditions. The thermal expansion and residual amounts of swelling in the supporting planes must be accommodated by this system. In addition, any rapid core compaction initiated by the clamping system must be excluded for safety reasons, even in case of failures or ruptures in the core or clamping system or as a result of maloperations. The hexagonal shape of the fuel elements, which initially makes clamping more difficult, proves to be advantageous here. If a single box should collapse in the core under the influence of external clamping forces, the neighbouring fuel elements would prevent excessive shifting.

To overcome the problems of structural material swelling, a number of other proposals have been made so far [43]. The most important ones are: Use of different material for the fuel element boxes, e.g., Nimonic, which, according to present knowledge, shows a much lower swelling rate under the respective temperature and fluence conditions; Reduction of the fuel burnup; Periodic rotation of the fuel elements, which, however, could only compensate for swelling induced bowing; Regular heating of the entire core structure to temperatures above 760°C , because experiments have shown, that the major portion of swelling is reversed at this temperature [44].

These proposals are of special importance to those reactors presently under construction, the core designs of which were frozen at a time when the structural material swelling was largely unknown.

The calculation of the stresses, strains and move-

Fig. 5. Micrographs of stainless steel diffusion weldings. Top: 4981/4981 material pair; below: 4981/4301 material pair.

ments of the fuel caused in a core under the combined influence of pressure and temperature differences in the course of temperature and power transients and as result of irradiation induced swelling and creep during burnup are an unusual complex procedure.

At the supporting points of the fuel elements and on the surfaces gliding upon each other in the clamping systems mentioned before very high pressures are often encountered. Therefore, the risk of seizing and of the formation of diffusion weldings on contact surfaces requires special attention for these parts [45, 46].

The possible formation of diffusion weldings between contact surfaces in a technical apparatus is a

Fig. 6. Refueling systems for sodium-cooled fast reactors.

new phenomenon for the mechanical engineer. It has to be taken into account in designing sodium facilities for higher operating temperatures because of the high strength such connections could reach. In fig. 5 a ground cross section is shown of a connection between stainless steel 4981 with itself and with stainless steel 4301 [47]. These connections were exposed to a temperature of 700°C for about 10 days after a short flushing period. The connection with 4301 showed a strength of more than 10 kp/mm^2 after loading with a pressure of only 1.25 kp/mm^2 . In other tests strengths of more than 20 kp/mm^2 were reached. The ground cross section clearly indicates the typical transition structure formed at the interface.

The tendency to form diffusion welding of a material pair seems to be determined, above all, by the material with the lowest high-temperature strength. When, for instance, the ferritic material $2\frac{1}{4}\text{Cr}-1\text{Mo}$ is involved, the resultant compounds were mostly very resistant. The formation of diffusion weldings decreases by use of heat resistant materials with different thermal expansion behavior, e.g., Inconel 718 together with stainless steel 4981. Also, the peak-to-valley height of contacting surfaces exerts some influence; in general, very smooth surfaces have a stronger resistance.

In the core structure, diffusion welding at the supporting and clamping points of the fuel pins and fuel elements could falsify in an uncontrollable way the expansion and time behavior of the reactor core. Consequently, it must be avoided by appropriate pairing of materials at these points. According to the experimental results available materials such as Inconel 718, as mentioned before, or Stellites are particularly suited. However, the corrosion behavior of these materials has to be considered, too.

3.3. Mechanisms in sodium and cover gas

Diffusion welding and the friction and wear behavior in sodium in general also play a decisive part in the design of the other mechanical devices and mechanisms of the reactor. This includes, above all, the control rod system, the refueling equipment and the core instrumentation. Particularly severe conditions apply to the control rod system which have to function in an absolutely reliable manner over a long period of time at the full operating temperature of the reactor in the interest of the safety and availability of the plant.

In general, systems which are mechanically driven from outside, are used for the control rods. They are almost similar in design to the systems known from various thermal reactors.

Fig. 7. Pantograph manipulator of the Dounreay PFR [50].

To guarantee the safe shutdown even in cases of serious damage of the reactor, e.g., major dislocations of guiding parts, a second independent shutdown system is often installed in addition to the normal shutdown and control rods, which takes into account also these conditions to an extremely high degree. With SNR, for instance, absorbers are used for this second shutdown system, which consists of separate, short parts connected by joints and which are pulled from below into the guiding channels of the reactor during shutdown.

As in the primary system, several solutions are commonly adopted for the refueling machines with dissimilar requirements of the mechanisms. The most important systems are compared in fig. 6.

System a) is the classical rotating top shield design which is known also from D_2O -cooled reactors. Modifications of it have often been used. Through

two rotating top shields in eccentric arrangement relative to each other a charging hole in the inner top shield can be moved vertically over each position of the reactor core. Thus, the fuel elements can be withdrawn from the core by a simple hoist. Evidently, the control rod drives must first be disengaged from the absorbers. This system is proposed for the 300-MW(e) SNR.

In system b) a fixed reactor top is used. It is penetrated by three manipulators each of which is capable of covering 1/3 of the core surface. In this design, the control rod systems and the instrument support plate must be disengaged also and either turned sideways or moved upwards. The U.S. FFTF will use this system.

Finally, in the third system according to c), the entire stationary reactor lid is lifted from the reactor tank together with the control rod drives and the core instrumentation, as in LWR's. However, due to the radiation field and the great chemical affinity of sodium to oxygen a shielded cell filled with an inert gas, e.g., argon, must be provided over the reactor. After removal of the top the whole inner space of the reactor tank is freely accessible within this refueling cell. Consequently, a similarly simple hoist as in the design with a rotating top shield can be used. Fueling systems with hot cells are favored above all in the United States [13, 48, 49].

As an example for a typical refueling mechanism, fig. 7 shows the pantograph manipulator envisaged for the British PFR according to fig. 2 [50]. This design allows a radial displacement of the charge chute in addition to the sweeping and lifting movements. The figure shows clearly the different mechanisms required, such as: the drive unit and the joints for radial displacement, the universal joints and the screw spindle for the lifting of the gripper. The gripper proper consists of a number of additional mechanical elements.

Mechanical parts as a whole have worked satisfactorily in sodium after initial difficulties had been overcome. The behavior of sliding and rolling surfaces is relatively well-known [51, 52]. A satisfactory behavior has been shown by material pairs such as Inconel 718 with Colmonoy 6 or Hastelloy C with Colmonoy 75.

The refueling machines, in particular, should present no major problems, because these systems oper-

ate at reduced temperatures and can be overhauled and inspected when not in use.

In the design and assessment of the mechanical equipment mentioned above the environmental condition of the cover gas atmosphere overlying the sodium level must be included in the considerations. In any case, parts of the control rod drives, refueling devices and instrumentation lines penetrate this space, and the rotating top shield has to operate under these conditions as a whole.

At a sodium temperature of 560°C the partial pressure of sodium vapor amounts to 0.016 atm. This means that in practice a two-component mixture of cover gas and sodium vapor is present in the cover gas space. This cover gas atmosphere is polluted by traces of oxygen and moisture, which are introduced either together with the supplementary cover gas, through small leaks or during insertion of charging tools, fuel elements and irradiation specimens. Of course, oxygen and hydrogen will immediately react in the cover gas atmosphere with sodium to give Na_2O , NaH , NaOH , and Na_2O_2 . These compounds may stay as aerosols in the cover gas atmosphere but may also deposit on cold walls. Especially, if a convective cover gas circulation exists, considerable local deposits may be formed preferably in the narrow gaps of the gas space. The cover gas circulation is enhanced due to the temperature differences existing between the center and the wall of the reactor tank and, in a very strong manner, due to the temperature difference between the sodium surface and the top cover, if the cover gas space is used intentionally for the temperature reduction in this direction.

The deposits may greatly impede the functions of mechanical components, above all of the rotating top shield [53]. Experience has shown that the metal fraction contained in these deposits is often so low that they cannot be eliminated by mere heating. Therefore, if deposits in critical areas cannot be excluded, e.g., by flushing the affected zones with clean cover gas, special precautions are necessary to enable the removing of these deposits from time to time. For instance, this could be realized by washing these areas with liquid sodium or with the help of adequate openings for the introduction of special cleaning tools.

It cannot be excluded that the environment conditions of the cover gas space overlying a high-tempera-

ture sodium surface will be of greater importance in the long run with respect to the design and construction of mechanical components than liquid sodium metal itself.

4. Conclusion

The introduction of sodium on a large scale and under the aggravated conditions of a nuclear reactor raises quite a number of problems some of which are difficult to solve. There is no doubt about this. The expenditure in terms of money and time required to develop these new technologies is correspondingly high. On the other hand, considering the progress made as a result of worldwide effort, there is just as little doubt that the problems raised can be solved. It can be assumed that for the time being all problems are well-known and have been solved to such an extent as to fully justify the erection of large-scale plants. Many questions, including the complicated problems relating to the reactor core, will be answered better by the practical experiment.

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