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6. THERMOHYDRAULICS OF REACTOR CORE AND SOME ASPECTS OF FAST REACTOR ENGINEERING

6.1. INTRODUCTION

High power rating and temperature impose special requirements to fast reactor thermohydraulics. Analysis of thermohydraulic issues assumes reliable hydraulic and heat transfer relationships to be worked out. As a result, distributions of the coolant flow rate, its velocity, and finally, fuel and core structure temperatures would be obtained. All mentioned parameters are required for evaluation of core integrity and mechanical behaviour.

The fuel subassembly life is affected by peak cladding temperature and temperature distributions in the coolant, fuel and subassembly structures. These temperature distributions should be such that the fuel burnup goals can be met without seriously impairing core life and performance. For constraining temperature levels by design measures, it is important that the fuel subassembly temperature distributions be accurately predicted and the governing thermal-hydraulic phenomena be understood. An example of temperature affected performance is that the estimate made for existing LMFR designs indicates that fuel burnup can be increased by about 10% (in MW^D/t) per one Kelvin degree reduction of hot spot cladding temperature.

Method developed for predicting temperature distributions should be capable of accounting for the effects of geometrical differences and the wide range of operating conditions of the core and blanket subassemblies, as well as control rods of LMFRs.

For prediction of subassembly coolant flow rate and temperature distributions a wide range of coolant flow and thermal convection regimes must be considered including laminar and turbulent flow; natural, forced and mixed (forced + natural) convection; and steady state and transient reactor conditions.

Thermohydraulic performance prediction for the core subassemblies begins with calculation of the subassembly flow rate necessary for determining the limiting design parameter (e.g. lifetime or maximum local cladding midwall temperature) in the limiting subassembly of each flow orificing zone.

Total calculated subassembly flow rate, as well as design data on core neutronics are input to subchannel analysis codes that predict coolant flow and temperature distribution in the subchannels of the core subassemblies. The peripheral subchannel temperatures and flow rates used for duct temperature prediction, as well as peak subchannel coolant temperature used for prediction of hot spot temperature of the hottest fuel elements are of particular interest.

For detailed cladding and coolant temperature information the distributed parameter codes are used, which calculate two-dimensional temperature fields in the coolant, cladding and fuel.

Within this Report it is impossible to discuss all details and strategies of LMFR thermal hydraulic design. Therefore, only the most important aspects will be treated in the following subsections.

6.2. BASIC ASSESSMENTS

LMFR core orificing is important because fuel lifetime strongly depends on maximum cladding temperature, which is determined primarily by the core coolant temperature rise (ΔT) and inlet temperature.

Subchannel analysis is commonly used for thermal hydraulic analysis of single fuel subassemblies. Bulk average values characterize the fluid dynamics and thermal coolant conditions in each subchannel. Thermal and hydraulic interactions between subchannels are taken into account.

All subchannel codes are used to solve mass, momentum and energy transport equations for turbulent fluid flow. Since the governing equations are very complicated, in each code simplifying assumptions are made to expedite solutions. Empirical input by either experimental data or analytical results is required for the codes to obtain reliable computer results. The correlations strongly depend on the geometry considered, the spacer concept (grids or wire-wraps), etc.

Experimental investigations, therefore, are always necessary on the actual geometry for an accurate design. However, some correlations, which are widely used, are mentioned in the following subsections.

Qualitative comparison of some thermohydraulic characteristics of sodium and lead coolants is presented in [6.1] with BREST reactor design taken as a reference [6.2]. Results of optional assessments performed by the author [6.2] are in a good agreement with the data presented in [6.1]. In the analysis, the following parameters are assumed identical: reactor power, average temperature in the circuit, cross section for coolant flow, coolant temperature rise in the reactor, primary circuit pressure drop and coolant volume.

Analysis has shown the following ratios:

Coolant velocity in the core	$W_{Na}/W_{Pb}=3.5-6.0$
Core pressure drop	$\Delta P_{Na}/\Delta P_{Pb}=0.16-0.2$
Pumping power	$N_{Na}/N_{Pb}=0.23-0.3$
Heat accumulated in the primary circuit prior to achieving boiling temperature	
$T_{Na} = 880^{\circ}\text{C}$, $T_{Pb} = 1745^{\circ}\text{C}$	$Q_{Na}/Q_{Pb}=0.7-0.4$

The latter assessment has no special sense for just logical reason, namely that lead boiling temperature cannot be achieved in LCFR, in contrast to that of sodium in LMFR, because reactor vessel would be necessarily broken before lead boiling onset (melting temperature of Cr18Ni10Ti, Cr15Ni11MoNb and other steels is about 1500°C). Because of this, there is no reason for considering high boiling temperature of lead as an advantage.

Thus, in terms of thermohydraulics, sodium cooled reactor parameters such as pressure drop (5–6 times lower) and required pumping power (3–4 times lower) are preferable.

Thermohydraulic parameters of BREST type plant should follow the following four requirements:

- Maximum permissible coolant velocity (based on corrosion and erosion conditions) is ~ 2.5 m/s;
- Low primary circuit pressure drop (under accident conditions, 15% flow rate should be provided);
- Temperature of the fuel element ferritic steel cladding is ~ 923 K (650°C);
- Minimum release of gaseous fission products (maximum fuel temperature should not exceed 1000°C).

6.3. HYDRODYNAMICS

As a rule, hydrodynamics of Na and Pb is quite similar, since both sodium and lead are Newton liquids. Therefore the data available on hydraulic resistance of the reactor core incorporating pin bundles can be used for adequate evaluation of hydraulic friction loss in the core [6.3–6.6].

Data on pressure drop for pin bundles were determined by summarizing a great number of experimental and analytical results. The following relationships have been proposed:

$$\Lambda_{tr} = 0.57 + 0.18(x-1) + 0.53[1 - \exp(-a)], \quad (6.1)$$

where $a = 0.58\{1 - \exp[-70(x-1)]\} + 9.2(x-1)$, and

$$\Lambda_{sq} = 0.59 + 0.19(x-1) + 0.52\{1 - \exp[10(x-1)]\}, \quad (6.2)$$

which is valid for $1 < x < 10$; $2 \cdot 10^4 < Re < 2 \cdot 10^5$.

In this case, Λ is ratio between friction factor of a bundle (λ) and that of round tube (λ_o), subscripts "tr" and "sq" refer to triangular and square arrays respectively, and $x = s/d$ is relative pitch of the pin bundle.

Equivalent hydraulic diameter values are calculated as follows:

$$d_{tr} = d \left(\frac{2\sqrt{3}}{\pi} x^2 - 1 \right), \quad d_{sq} = d \left(\frac{4}{\pi} x^2 - 1 \right)$$

Instead of equation (6.1), more simple relationship can be used:

$$\Lambda_{tr} = 0.21/Re^{0.25} [1 + (x-1)^{0.32}]$$

for $1 < x < 1.5$; $6 \cdot 10^3 < Re < 2 \cdot 10^5$.

Equation (6.2) is based mainly on the results of analysis and limited experimental data. For more specific consideration, additional studies and new results are required, for example those obtained by other authors.

Local pressure drop in the subassemblies (for instance that due to spacing elements) can make significant contribution to the total pressure drop [6.7, 6.8]. It is unlikely, that spacing elements used in the fuel subassemblies of sodium cooled reactors (wire, etc.) could be applied for lead cooled reactors. Therefore, if the new spacers are designed, the available data cannot be applied. This would require special-purpose experiments not only for determining hydrodynamics, but to test structure for vibration and attrition.

There is one point which is most important for the development of lead cooled reactors. When carrying out studies on the models of lead-bismuth cooled reactor flow path (including full-scale model), it was found that in parallel with the toroidal eddies appearing at the core diagrid inlet, sufficiently stable horizontal eddies may arise near the bottom of the

diagrid. These eddies result in reduction of local inlet flow rates and flow stability, thus facilitating local concentration of particles suspended in the core diagrid area and, hence, offering risk of their penetration into the core.

6.4. HEAT TRANSFER

6.4.1. Basic relationships

In general, comprehensive studies have been made on heat transfer of different liquid metals (Hg, Na, NaK, PbBi, Li, etc.). Unless special measures are taken for liquid metal purification, heat transfer coefficients follow the relationship (for tubes):

$$Nu = 3.3 + 0.014 Pe^{0.8}. \quad (6.4)$$

Careful purification makes heat removal more intensive, as it is described in [6.12–6.14]:

$$Nu = 5.0 + 0.025 Pe^{0.8}. \quad (6.5)$$

Low heat transfer coefficient (6.4) is caused by thermal contact resistance at the wall-liquid boundary. Thermal resistance is influenced by surface wetting with coolant, oxide films formed on metal surface, as well as deposits of oxides and other impurities. Thermal contact resistance can hardly be evaluated, only possible upper limit can be specified [6.15, 6.16]:

$$\left(R_c \frac{\lambda}{d} \right)_{\max} = 200 \cdot Re^{-0.75}. \quad (6.6)$$

Fig. 6.1 shows the data available and the relationship presented.

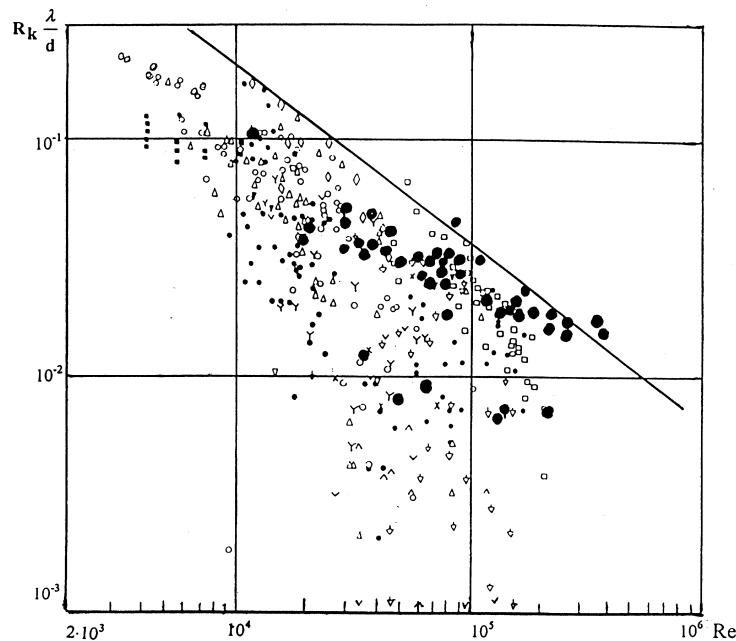


FIG. 6.1. Dimensionless thermal contact resistance as Re function [6.15, 6.16]:

● — PbBi; other sympols: Li, Na, NaK, Hg

---- — equation 6.6.

Thus, common conclusion cannot be reached on value of thermal contact resistance. As for lead, this question remains open and the data must be checked.

Normal purification of liquid metal leads to almost complete elimination of thermal contact resistance due to depositions.

Typical thickness of passivation film in PbBi alloy is about 10 μm and 20 μm respectively for austenitic and perlitic steel, though scattering within the range from 10 μm to 100 μm . The film thermal conductivity is about 5 W/(mK). There is a good reason to believe that the film is not continuous because of its saturation with liquid metal. Thus, thermal resistance is rather low and temperature difference across the film is about 5K.

It is clear that the coolant must be purified. While this is not a problem for sodium, in case of lead one should bear in mind presence of permanent oxides in the circuit taking into consideration corrosion activity of lead and necessity of protective oxide films formation on the metal surface. Comparison of heat transfer in tubes with sodium and lead-bismuth described by (6.1) and (6.2) equations is shown in the Table 6.1.

TABLE 6.1. COMPARISON OF HEAT TRANSFER COEFFICIENT FOR DIFFERENT COOLANTS ($\alpha - \text{w}/(\text{M}^2\text{k})$, $T = 400^\circ\text{C}$, $D = 10 \text{ mm}$)

Na

Re	Pe	Nu ₁	Nu ₂	α_1	α_2
20000	90	3.512	5.915	25018	42131
30000	135	3.709	6.265	26416	44628
50000	225	4.066	6.904	28964	49178
100000	450	4.856	8.315	34593	59229

Pb

Pe	Nu ₁	Nu ₂	α_1	α_2
348	4.511	7.699	7723	13181
522	5.091	8.733	8715	14951
870	6.146	10.618	10522	18177
1740	8.477	14.781	14513	25305

PbBi

Pe	Nu ₁	Nu ₂	α_1	α_2
294	4.321	7.358	6135.4	10449
441	4.827	8.262	6854.0	11732
735	5.749	9.909	8163.4	14070
1470	7.786	13.547	11056	19236

6.4.2. Pin bundle heat transfer

Lead cooled fast reactor design assumes the subassembly with fuel pins arranged in square lattice with large pitch-to-diameter ratio ($s/d \sim 1.4-1.5$). In LMFR a more tight arrangement of fuel pins is adopted ($s/d \sim 1.1-1.18$):

Reactors	BR-10	BOR-60	BN-350	BN-600	BN-800	Phenix	Super Phenix
$x=s/d$	1.10	1.10	1.16	1.16	1.17	1.178	1.18

Both experimental and analytical studies of heat transfer in Hg, PbBi, NaK and Na coolants have been carried out. The following relationships for evaluation of heat transfer in triangular lattice bundle were obtained experimentally:

1. V.M. Borishanski, et al (triangular arrays) [6.20]:

$$Nu1 = 24.151 \lg[-8.12 + 12.76x - 3.65x^2] + 0.0174[1 - \exp(-6(x-1))](Pe - 200)^{0.9}$$

2. H. Graber, (triangular arrays) [6.21]:

$$Nu2 = 0.25 + 6.2x + [0.32x - 0.07]Pe^{(0.8-0.024x)}$$

3. H. Calamai, et al [6.23]:

$$Nu3 = 4 + 0.16x^5 + 0.33x^{3.8} (Pe/100)^{0.86}$$

4. A.V. Zhukov (rod bundles with spacers, square arrays) [6.22]:

$$Nu4 = 7.55x - 14x^{-5} + 0.09Pe^{(0.64+0.264x)}$$

A.V. Zhukov (rod bundles without spacers, square arrays) [6.22]:

$$Nu4 = 7.55x - 14x^{-5} + 0.07Pe^{(0.64+0.264x)}$$

It has been demonstrated that Nusselt number for square lattice is lower than that for triangular lattice bundles [6.17, 6.18]. This is illustrated by the data in Fig. 6.2.

In Table 6.2 and Fig. 6.3 relationships $Nu=f(Pe)$ taken for triangular and square lattice with fixed pitch-to-diameter ratio ($x=1.4$) are compared.

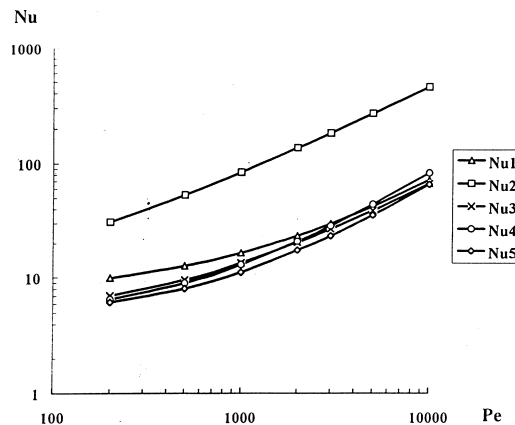


FIG. 6.2. Correlation for liquid metal heat transfer in rod bundles ($s/d = 1.4$).

TABLE 6.2. COMPARISON OF HEAT TRANSFER IN ROD BUNDLES WITH TRIANGULAR (TR) AND SQUARE (SQ) ARRAYS, $X=1.4$

Pe	100	200	300	500	1000	2000	3000	5000	10000
Nu(tr)	11.26	11.98	12.64	13.86	16.61	21.46	25.90	34.08	52.44
Nu(sq)	8.62	9.25	9.89	11.24	14.25	20.4	26.50	38.61	68.60
Nu(sq)/Nu(tr)	0.766	0.772	0.782	0.804	0.858	0.950	1.023	1.133	1.308

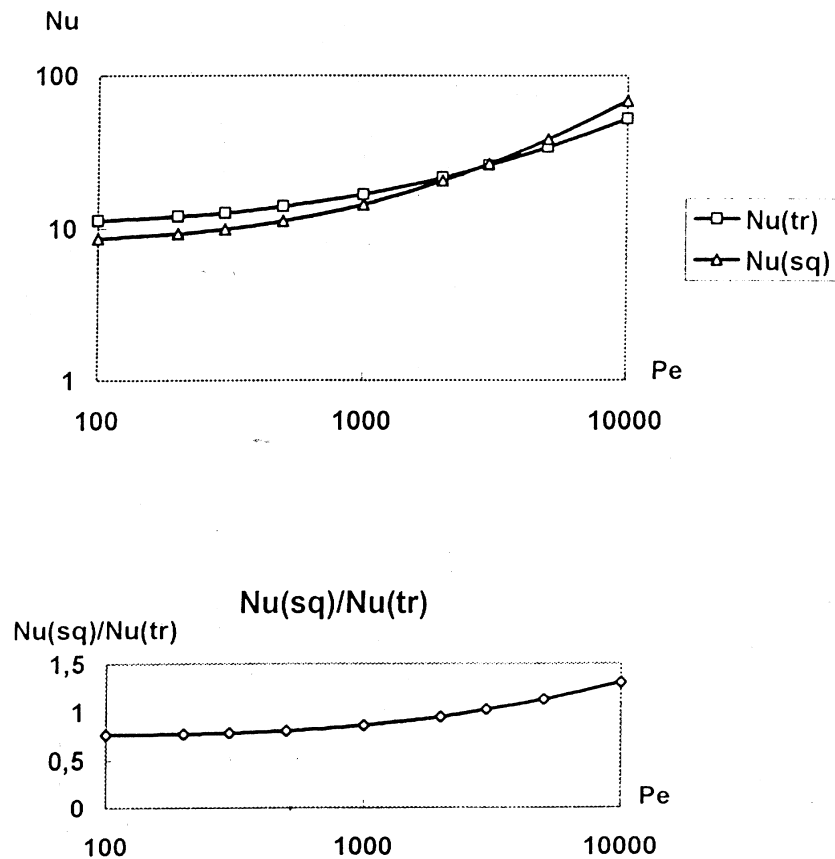


FIG. 6.3. heat transfer in rod bundles with triangular and square arrays.

Heat transfer in square bundles is roughly estimated, so special experiments are needed for verification. By now there are no convincing methods of extrapolation of the data on temperature non-uniformity in the area around fuel pin to the square grid bundle. As a first approximation, it can be assumed that values increase in the same way as in case of laminar flow. Unfortunately, this cannot be adopted for the peripheral pins where temperature pattern non-uniformity is significant. Therefore, additional experimental studies are needed.

Available experimental results on temperature distribution in plugged SA were analyzed to determine the main characteristics of velocity and temperature profiles of single-phase flow, namely: recirculation zone length, coolant flow distribution in SA, coolant temperature rise in the wake, etc. At the same time, it is clear that data currently available provide only rough estimations of SA thermohydraulic characteristics in case of blockage as a function of Reynolds number and blocked flow area.

In-pile and out-of-pile test results show that boiling occurring behind the blockage does not lead to rapid propagation of failure in the fuel pin bundles, while the gas release taking place as a result of pin failure behind the blockage may cause more pin damages within the SA. Studies of various factors influencing temperature profile (flow rate, power rating, Reynolds and Peclet numbers, blocked fraction of flow cross section area) call for carrying out additional comprehensive experimental and analytical studies. Special attention should be given to studies of thermohydraulics of SA with porous heat generating blockages, initial dry-out conditions with boiling occurring behind the blockage and SA cooling limits.

As for local thermohydraulic characteristics, such as hot spot factors, temperature non-uniformity in the irregular arrangement of pins etc., of critical importance are the processes of inter-channel exchange. Comprehensive studies of inter-channel heat and mass transfer in the bundles of smooth pins with ribs or wires provided a basis for development of calculation techniques.

Mixing factors in case of wire-to-pin contact in the bundle of wire spaced fuel elements turned out to be higher than those for wire-to-wire contact. Counter-direction wires providing lateral coolant flows would be most effective.

Issue of spacing elements development for lead cooled SA is rather complicated because of high dynamic pressure head, possible vibrations, etc. This is the subject of further detailed studies. A delusion still exists that rather high thermal conductivity would smooth temperature non-uniformity. It is only true for stagnant liquid metals that are equivalent to the solid body. Experiments and evaluations show the necessity of considering conjugate tasks of heat removal from fuel elements, i.e. taking into account properties of fuel elements.

As it was demonstrated by experimental studies, considerable temperature fluctuations take place in both coolant and channel walls. Preliminary estimations showed that such pulsation would not impact structural strength. However, this does not relate to the situation when two flows of different temperature are mixed. In this case thermal stress can result in structure failure, that happened at Phenix IHX.

The problem cannot be resolved without implementing a comprehensive study program, including analysis of thermal stripping, experimental studies on computer code verification, development of codes and techniques for measurements of the wall surface temperature fluctuations, etc.

6.5. FAST REACTOR ENGINEERING

6.5.1. Sodium-water interaction

Because of sodium radioactivity in the primary circuit and the possibility of sodium-water interaction in the steam generators, LMFRs are equipped with non-radioactive intermediate (secondary) sodium circuit. Intermediate heat exchangers thus constitute an internal containment barrier. In case of failure of water filled pressurized steam generator tube, water is injected at high pressure into sodium, and violent sodium-water interaction occurs. This sodium-water interaction would give rise to peak pressures that must be accommodated by straining the steam generator shell and the pipelines in the secondary circuit. Moreover, if the pressure in the system becomes too high, a special rupture discs in sodium and/or gas sections will be broken, thereby relieving the pressure and ejecting sodium

hydroxide, sodium oxide and hydrogen into a special collector vessel. This pressure relief system is very important, since it protects intermediate heat exchanger from damage. The sodium-water interaction is immediately detected by instrumentation and the plant would be shut down if a large leak occurred. The basic physical processes are discussed below.

Sodium-water steam generator design features. One of the principal problems in the development of the sodium-water steam generator (SG) design is the choice of the tube bundle characteristics.

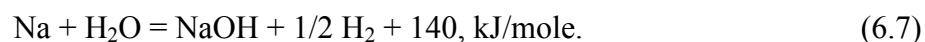
Specific requirements are also given to sodium-water SG structural materials. Corrosion-erosion resistance of materials contacting with sodium-water reaction products becomes of great importance. From this point of view, high-nickel alloys (30–40% Ni) have the best characteristics; chromium-nickel austenitic steels of X18H9 type being somewhat inferior to them. Ferritic steels, such as 2¹/₄Cr1Mo and chromium steels with 9–12% Cr content are most susceptible to corrosion-erosion failure caused by reaction products. It should also be taken into account that ferritic steels are low-cost and relatively workable but have low corrosion resistance and are prone to pitting on the steam-water side.

Besides, there are certain limitations on temperature conditions for some steels (for example, stability of mechanical properties of 2¹/₄Cr1Mo steel can only be guaranteed at temperature not exceeding 520°C). It should be also noted that practically all of the above steels have adequate corrosion resistance in pure sodium and all corrosion problems take place on the steam-water side. So it is seen that choosing tube material is rather complicated problem.

Taking into account various properties of steels and contradictory requirements imposed to them, the SG design features, such as unit heat capacity, tube bundle characteristics, maintainability, etc. sometimes play decisive role in the choice of structural materials. To date, there are examples of implementation of all the above steels.

Speaking about design features, small tube-to-tube distance in the tube bundle should be mentioned, since micro defect present in one tube would result in rapid failures of the adjacent tubes.

Physical phenomena of sodium-water reaction. Sodium-water chemical interaction proceeds in two stages: at the first stage the reaction proceeds at a high rate with release of gaseous hydrogen and heat:



This reaction is practically irreversible because of a very high equilibrium pressure of hydrogen.

At the second stage chemical interaction takes place between the products of the first stage of the reaction and excess sodium (excess sodium is a characteristic of water leak in the steam generator):





The total reaction is described by the following equation:



Thus, sodium-water interaction is rather complicated multistage process including successive reactions with sodium hydroxide and hydrogen formation followed by their interaction with sodium. The final concentration of water-sodium interaction products is determined by thermodynamic equilibrium conditions and the time of reaching equilibrium state in the course of the reaction.

Small water leaks into sodium occur in the tubes with formation of chemical reaction products jet due to coolant interaction with water. The rate of corrosion-erosion damage of the adjacent tube material by reaction products formed in the jet depends, to a considerable degree, on the tube-to-tube gap value.

When choosing tube-to-tube gap value, the designer should be governed, on the one hand, by the limitations on the overall bundle dimensions, and, on the other hand, he should take into account the consequences of accidental contact of coolants caused by small water-into-sodium leaks. Practice has shown that for the single module design of small heat capacity, for which repair problems can be solved by simple replacement of failed module by the new one, optimum tube-to-tube gap value is within 12–15 mm range. For larger capacity SG using corrosion-resistant steels (for instance, Incolloy 800), tube-to-tube distance is sometimes increased up to 30–40 mm.

The second design feature comes from a desire to ensure the integrity of the steam generator vessel and sodium circuit under conditions of large water-into-sodium leaks (appreciable tube failures). As experimental studies have shown, an efficient means to reduce the rate of hydrodynamic effects in case of contact of large amounts of sodium and water is the use of expansion gas volumes. For the designs of large unit heat capacity the expansion (damping) gas volume is provided, as a rule, within the SG vessel.

It should be noted that the problem of sodium circuit hydraulic resistance is of vital importance due to some features of accident processes. Accidental pressure in the sodium volume in the vicinity of large water leak into sodium depends not only on the leak rate but also on the hydraulic "compliance" of the sodium circuit. The experience shows that, in view of the above circumstances, sodium circuit pressure drop within the boundaries of "SG-expansion gas volume inlet" section should not exceed 0.3 MPa value.

In some experimental and prototype NPPs with sodium-cooled fast reactors water/steam-sodium contacts took place in SG. The requirements to sodium-water steam generator from the standpoint of its tightness should be much higher than those to any other heat exchanger. When considering specific problems, on the basis of experience obtained on some NPPs the following conclusions can be drawn:

1. Operating experience of "Enrico Fermi" NPP SG (the first SG with single wall separation of sodium and water) has shown that tubes vibration led to their very rapid failures and, as a result, to large leaks. Insufficient attention was given to design optimization of some units and systems. It was also concluded that the choice of tube wall thickness of 1 mm

proved to be inappropriate solution and it cannot be adopted for the sodium-water steam generators.

2. As it is known, the cause of a number of large and small water leaks into sodium in the BN-350 NPP steam generators in 1972–1975 was their poor quality of fabrication. Nevertheless, the experience gained from incidents in the BN-350 NPP steam generators has also some positive results. First (and this is most important), it has been shown that in principle an ingress of large amount of water into sodium (in one of the incidents, about 800 kg of water got into the sodium circuit) does not lead to catastrophic consequences. Therefore, integration of sodium and water within one plant is quite justified from the engineering point of view. Second, steam generators and sodium circuits, even under conditions of substantial water-into-sodium leaks, remain repairable. Third, steam generator safety systems, even being under severe accident conditions, are capable of ensuring plant safety.
3. Water-into-sodium leaks that occurred in the steam generators of the BN-350 and other reactors were of much less scale than those occurred on the PFR NPP. However, these leaks have also confirmed the main conclusions drawn on the basis of the BN-350 and PFR NPP experience from the viewpoint of emergency conditions caused by chemical activity of sodium to water.
4. Operating experience of the BN-600 NPP steam generators has revealed the necessity of more careful choice of materials and refinement of fabrication techniques (in particular, of weld joints). However the main conclusion is that sodium-water steam generator design can be developed that would be practically fully capable of eliminating impact of steam generator leaks on the operation of the plant as a whole. This can be sectional-modular design, with single modules of the evaporators, superheaters and reheaters integrated into several sections, and each section can be isolated while all other sections are kept in operation. Of course, such design is more expensive as compared to the integral concept. However, during 20 years of operation this design fully paid for itself.

6.5.2. Burning and freezing of coolants

Sodium burning. Metallic sodium in its pure form does not exist in nature. The reason for this is its high chemical activity, especially with respect to air and water. Sodium interacts with water very intensively in the open air. In contrast to it, sodium reaction with air is slow. Sodium is solid at normal temperature, its surface being coated with the oxide. It is not possible to define the exact value of sodium ignition temperature because it depends on impurities content in sodium, air humidity and sodium-air interface condition. According to the results of studies performed by different investigators the ignition temperature values lie in the range from 140°C to 320°C.

As for burning process intensity, sodium takes one of the last places among other combustibles. This can be illustrated by comparing sodium and petrol pool burning characteristics [6.24]. Temperature distribution in the area above burning sodium and petrol is presented. Burning conditions were equal in both cases, burning surface area was about 1 m². Sodium combustion rate was 45 kg/m²hour. In the point 1 m above the sodium surface the temperature was lower than 100°C. Reaction zone (flame zone) was situated close over the sodium surface. In the case of petrol, the flame was as high as 4 m. Time averaged temperature measured in the point 2 m above the petrol surface was over 600°C, and petrol

combustion rate was 4 times higher than that of sodium. This is owing to different physical properties of sodium and petrol. Sodium boiling temperature is 880°C at atmospheric pressure, and evaporation latent heat value is 4340 kJ/kg. This results in its relatively low evaporation rate.

On the other hand, petrol boiling temperature is 80°C and its evaporation latent heat value is 12 times lower as compared to that of sodium. Therefore in the burning petrol boiling point is achieved soon resulting in sufficient vapour amount over its surface. That is why high flame is formed over the petrol. Sodium-air reaction heat is 10900 kJ/kg that is 4 times lower as compared to that of petrol. Thus, energy release rate during sodium burning is approximately 15 times less than that in case of petrol.

The intensity of processes in the reaction zone is determined by two mechanisms: (a) air oxygen diffusion into reaction zone, and (b) opposite direction sodium vapour diffusion from sodium surface into the reaction zone.

The former mechanism basically controls processes at temperatures lower than 650°C. With temperature increase the latter mechanism becomes more and more significant. The higher sodium temperature, the more vapour generation rate. The reaction zone extends from the sodium surface resulting in the decrease of heat transfer to sodium. This, in its turn, decreases evaporation rate. Studies have shown that steady state is achieved at the pool sodium temperature of 720°C to 745°C. During burning process 15 to 25% of the combustion product mass leaves the reaction zone in the form of fume. Nevertheless sodium mass remains almost constant because of simultaneous mass increase due to sodium oxidation.

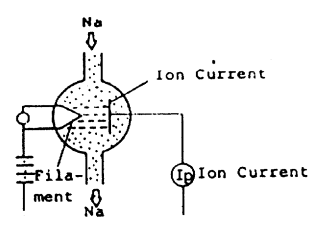
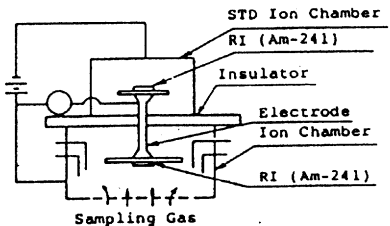
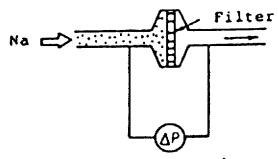
All this concerns burning of sodium spilled over the horizontal plane. However sodium drop burning can also take place. Specific rate of falling sodium burning is higher than that of sodium stationary pool. This also applies to the other burning effects. However in order to spray outflowing sodium, definite conditions are needed which are seldom met in practice. Drop burning effect should be taken into account in case of outflowing jet spraying caused by some barrier (e.g. floor or walls of the room). Continuous sodium spray does not burn because of time deficiency. As in any fire, smoke is one of the most dangerous factors in the sodium burning process. Sodium fume consists of combustion products that have become aerosols. These aerosols are sodium oxides (NaO and Na₂O) coming immediately into interaction with atmospheric components, steam and carbon dioxide producing sodium hydroxide-alkali (NaOH) and sodium carbonate (Na₂CO₃).

The process of oxide into hydroxide transformation takes few seconds after particle formation. Sodium carbonate is formed more slowly. This process rate depends on atmosphere humidity and takes several minutes. Both chemical and physical transformations of particles take place during the fume propagation, as well as conglomeration of particles formed in the burning process. With relative air humidity over 35%, dry particles are transformed into fluid drops. There are changes of their density and size distribution mode. Sodium particle radius varies within the wide range: from several hundredths of micron to tens of microns. In the process of aerosol propagation they deposit on the floor, ceiling and walls of the room, as well as on the surfaces of equipment and ventilation piping. This particle precipitation takes place mainly because of gravity: up to 80% of all particles are deposited on the floor. Aerosol particles can cause equipment damage, concerning first of all electrical equipment and instrumentation. These particles are also dangerous for people.

Sodium leak detection methods. Sodium metal leaking through a small crack in the component or piping wall in LMFR may drop off or flow out along the wall, producing aerosols as a result of its interaction with atmospheric oxygen and moisture.

There are two methods of sodium leak detection. One of these methods is based on detection of certain minimum amount of leaking sodium accumulated in one location. The other method is referred to as gas sampling method, based on aerosols detection in the gaseous atmosphere evacuated continuously by pumping. Table 6.3 illustrates gas sampling method and detection principle. Laser Type Sodium Leak Detector is based on gas sampling, and if sensitivity of gas sampling detector is adequately increased, it is feasible to provide monitoring throughout the whole plant.

TABLE 6.3. TYPE OF GAS SAMPLING AND DETECTING PRINCIPLE

Type	Principle of detection	Sensitivity
Sodium Ionization Detector (SID)		10^{-10}g/cm^3
Radiative Ionization Detector (RID)		10^{-10}g/cm^3
Differential Pressure Detector (DPD)		2mg

Heavy coolant freezing. One of the most dangerous events resulting in coolant freezing is the secondary steam header depressurization. Rarefaction wave propagates at sound velocity in the secondary circuit on the initial stage of such accident.

Secondary circuit water boiling up and abrupt increase of heat removal from the primary coolant occurs to result probably in its freezing. Steam turbine drive of feed water pump and by-pass pipelines connecting steam generator inlet and steam header through orifices are provided in the secondary circuit to decrease consequences of such accidents.

Evaluations have shown that coolant freezing does not occur in presence of by-pass pipelines in the secondary circuit. If there are no such pipelines, lead freezing begins at 15th s of emergency discharge.

Engineering features are provided in the reactor design to prevent lead freezing due to malfunctions or personnel errors in start-up and transient modes.

Relatively low melting temperature of PbBi eutectic (125°C) required study to be carried out on the primary circuit components and reactor as a whole in case of a change of phase state of the coolant under conditions of accident or scheduled cooling down and the posterior primary circuit heating up («freezing» and «unfreezing» conditions).

Variation of the eutectic volume during its solidification and comparability of its mechanical properties with those of structural materials leads to additional loading of the primary circuit components.

Study of change in PbBi eutectic density within the limits of possible deviation from eutectic within the temperature interval from -50°C to +130°C has been carried out at the IPPE.

It has been demonstrated, in particular, that in case that Pb content deviates by +5% from that of the eutectic composition the volumetric effect of fusion neither varies significantly, nor deviates from the corresponding value for eutectic by more than +0.17%.

Variation of the eutectic volume during its solidification and comparability of its mechanical properties with those of structural materials leads to additional loading of the primary circuit components.

The experimental studies have shown that due to limited solubility of components in the solid state, PbBi alloy is subject to phase-structural transformations, in the course of which its properties change. It has turned out that it is possible to describe these changes using kinetic relationships that have been determined for such most significant properties as density, elasticity and creep.

Experimental studies of features of the alloy-structure interaction were carried out by tests of mock-ups, models and real primary components. Fuel subassemblies, control and safety system components, primary pumps, steam generators, sections of main and auxiliary pipelines and their valves have been tested. In the course of tests requirements have been worked out for the temperature conditions of heating up and cooling down modes, as well as for separate structures designed for avoiding damages at multiple condition changes.

On the basis of experimental and analytical studies algorithms have been developed to implement conditions of particular plants.

The research work carried out and the experience gained in the field of realising freezing-unfreezing conditions facilitate solution of the problem of different type plants to be applied for various purposes and to realise the conditions repeatedly according to the schedule.

CONCLUSIONS

1. Analysis of experimental results on LMFR has served the basis for approximate evaluation of the main characteristics of coolant flow velocity and temperature profiles in the fuel subassemblies with blockage of single-phase flow, i.e. recirculation zone length, coolant flow distribution in SA, coolant temperature rise in the wake, etc. On the other hand, it is clear that currently available data provide only rough estimations of SA thermohydraulic characteristics in case of blockage depending on the Reynolds number and blocked flow cross section area.
2. Results of in-pile and out-of-pile tests demonstrate that boiling occurring behind the blockage does not lead to rapid propagation of failure within the pin bundle. Gas release taking place as a result of pin failure behind the blockage may be the cause of more pin damages in the SA.
3. Studies of various factors influencing temperature profile (such as flow rate, power rating, Reynolds and Peclet numbers and blocked fraction of flow cross section area) call for carrying out additional comprehensive experimental and analytical studies. Special attention should be given to the investigation of thermal hydraulics of SA with porous power generating blockages, as well as initial dryout conditions with boiling occurring behind the blockage and SA cooling limits.
4. Pressure drop in the pin bundles can be evaluated using relationships available.
5. Evaluation of local pressure drop in LCFR core (spacing grids and other structural elements of SA) requires additional studies to be carried out on each modification stage.
6. Heat transfer ratios for square/triangular lattice of the pin bundles (LMFR) are as follows:
 - for laminar flow, $Pe=100$, $x=1.4$: $\alpha(sq)=0.6\alpha(tr)$;
 - for turbulent flow, $Pe=1000$, $x=1.4$: $\alpha(sq)=0.65\alpha(tr)$

Taking into account velocity profile and properties of different coolants (Na, PbBi and Pb) the inherent Peclet numbers in LCFRs are 3 times higher as compared to those in LMFRs, thus assuring approximately equal heat transfer coefficients for these reactor cores (provided coolant is purified). If impurities are present in lead or PbBi coolant, one should take into account possible impact on heat transfer, and additional thermal resistance can be evaluated by (6.3).

7. Hot spot factors associated with (a) uncertainty of coolant parameters distribution at the SA inlet and (b) probable distortion of SA configuration during operation, that may impact LCFR parameters, should be taken into consideration, particularly in case of high values of pin bundle relative pitch.

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7. RADIOACTIVITY AND TOXICITY OF COOLANTS

7.1. SODIUM COOLANT RADIOACTIVITY

“Operational” radioactivity. The natural isotope of sodium is ^{23}Na (abundance 100%). Neutron capture processes in sodium (n,γ reaction resulting in 1.4 MeV γ -quanta emittance) lead to formation of ^{24}Na isotope with half-life time of 15 hours, while sodium is flowing through the core. Besides, there is $(n, 2n)$ threshold reaction producing ^{22}Na with 2.6 years half-life time. ^{22}Na emits 1.3 MeV γ -quanta; its activity is proportional to thermal power of the reactor plant.

Neutrons are not practically generated by sodium radionuclides with half-life time exceeding 2.6 years value, even after 50 years of exposition to intensive fast neutron flux.

^{24}Na is the main isotope giving rise to requirement of protection against γ -radiation. Approximately 10 days after reactor shutdown the primary circuit activity is mainly determined by ^{22}Na .

This feature, along with the fact that sodium interacts chemically with water and air, results in three circuit reactor power plant design, including:

- primary circuit containing radioactive sodium heated up in the core,
- secondary non-radioactive sodium circuit coupled with the primary circuit by the intermediate heat exchanger,
- tertiary water circuit producing steam for electricity generation.

Within the primary circuit radioactive coolant is protected against air by the steel barriers and cover argon gas. Radioactive sodium of the primary circuit is separated from non-radioactive sodium of the secondary circuit by the steel tubes of the intermediate heat exchangers.

The operating experience of different reactors has shown, that the coolants separated by the wall of heat exchanging component of the reactor heat removal circuit (steam generator in PWR, turbine condenser in BWR and intermediate heat exchanger in LMFR) should be the same to assure nuclear plant safety and reliability (or at least highly compatible to each other in order to avoid plugging of the core flow channels by the products of coolants interaction). The rise of the total NPP cost owing to this measure (application of identical coolants in the primary and the secondary circuits) and use of tertiary water-steam circuit is insignificant, since standard operation and maintenance of steam generator, water circuit and turbine plant are applied.

Operational experience gained on LMFRs, such as BN-350, Phenix, PFR, BN-600 etc. has proved, that sodium is practically non-corrosive to stainless steel, and content of impurities, mainly oxygen and carbon, are maintained at acceptably low level by the cold traps installed in the bypass of the main coolant circuit. Radioactive hazardous isotopes (caesium, tritium, strontium and iodine) are retained by sodium.