# The Fast-Neutron Breeder Fission Reactor: Safety Issues in Reactor Design and Operation [and Discussion]

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# The fast-neutron breeder fission reactor: safety issues in reactor design and operation

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Today's fast breeder reactors contain mixed uranium–plutonium oxide fuel and are cooled with liquid sodium. Their normal operational behaviour, including power transients, is similar to that of thermal reactors.

The fact that the sodium density coefficient is positive is of no importance at normal operating temperatures because negative coefficients like Doppler or fuel expansion coefficients have compensating effects. Dangerous effects may arise when sodium boiling at much higher temperatures occur. It is shown that sodium boiling in most cases can be avoided by proper design of the reactor core. Energy releases associated with partial destruction of the core are discussed.

The safety features of metallic fuel are briefly discussed, resulting in the statement that in principle, use of metallic fuel does not promise more positive safety features.

## Safety-related technical features in thermal and fast reactors

Fast breeder reactors (FBRS) of the present generation are characterized by unmoderated reactor cores, i.e. the use of fast neutrons, by mixed uranium–plutonium oxide (MOX) fuel, and by liquid sodium for heat transfer. These main features imply certain differences in the design, safety philosophy, plant dynamics, and operating mode of FBR plants as compared with the various types of thermal reactors, e.g. light-water reactor (LWR) power stations. These differences will be treated in the following sections.

However, it is to be noted that a number of basic safety-related features are essentially the same in either case. For example, the time available for neutron-absorber rods to be actuated for a scram is comparable for breeder and light-water reactors, in spite of some numerical difference between the contributions of delayed neutrons to the neutron spectra (Häfele 1963; Fröhlich *et al.* 1969; Kessler 1983; IAEA 1985; see also tables 1 and 2). A sudden increase in reactivity (starting from normal operating conditions) resulting in a prompt-supercritical power excursion would cause a temperature increase that has a negative feedback on the neutron flux and thus would automatically reduce the reactor power to a stable level. For FBRS, these safety features have been forecast by theoretical calculations related to the so-called nuclear Doppler effect. They have been confirmed by operating experience and by experiments especially in the Southwest Experimental Fast Oxide Reactor (SEFOR) reactor in 1970, cf. figure 1 and Caldarola *et al.* (1972).

It should also be mentioned that the annual quantity of fission products of reactors of equal power is comparable, so that the problems of final waste disposal are roughly the same for FBRS and LWRS.

There are also many formal parallels in the safety approach to breeder and light-water reactor plants. The lines of assurance in either case include basic design with suitable choice of materials and welding techniques, quality assurance programmes and in-service inspections,

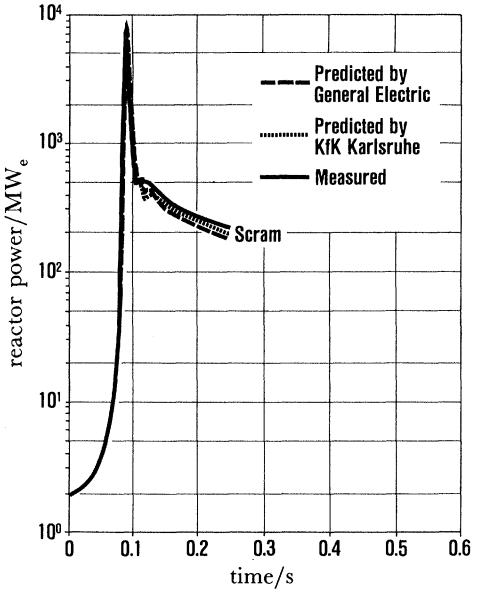


FIGURE 1. Superprompt critical power transient experiments at the SEFOR plant.

TABLE 1. COMPARISON BETWEEN NUCLEAR CHARACTERISTICS OF PRESSURIZED-WATER REACTOR (PWR) AND FBR CORES

| | PWR (1300 $MW_e$) | FBR SNR300 |
|---|---|---|
| fuel | $^{235}U$–$^{238}U$ | $^{239}Pu$–$^{238}U$ |
| Doppler coefficient/°C$^{-1}$ | $-(2\text{–}3)\times10^{-5}$ | $-5\times10^{-6}$ |
| prompt neutron lifetime, $l_{eff}$/s | $2.5\times10^{-5}$ | $4.1\times10^{-7}$ |
| effective fraction of delayed neutrons, $\beta_{eff}$ | 0.005–0.0065 | 0.0033 |
| mean decay constant of delayed neutron precursors, $\lambda$/s$^{-1}$ | 0.077 | 0.087 |

TABLE 2. COMPARISON OF REACTIVITY EFFECTS AND DESIGN FEATURES OF CONTROL AND SHUTDOWN SYSTEMS FOR PWR AND FBR

| | PWR (1300 $MW_e$) | FBR SNR300 |
|---|---|---|
| control rod insertion velocity/(mm s$^{-1}$) | 1 | 1.5 |
| control rod insertion velocity/($10^{-2}$ \$ s$^{-1}$) | 2.5 | <4 |
| shutdown rod insertion velocity/(cm s$^{-1}$) | 156 | 85–190 |
| dropping range/m | 4 | 1 |
| delay period for reaction of protective system/s | 0.2 | 0.2 |
| dropping time of shutdown rods/s | 2.5 | 0.6 |
| reactivity of shutdown system/\$ | ≈11 | ≈10 |
| reactivity of burn-up/\$ | ≈19 | ≈8 |

protective instrumentation and diagnosing systems for timely detection of failures that might initiate more severe incidents, engineered safeguards and accident management to mitigate the consequences of beyond-design accidents. Probabilistic risk analyses and the concept of ‘residual risk’ have come to play an important role in licensing procedures as well as public debates. In addition, principles such as redundancy (multiple lay-out), diversity (against common-mode failures), ‘fail-safe’ design and ‘passive’ safety have become typical of all safety-related nuclear subsystems. Several recent incidents have also made it generally urgent to improve the control-room design to create a better man–machine interface.

### Typical aspects of breeder reactor safety

Most of safety features characteristic of breeder reactors are connected with the sodium coolant. To begin with, let me list the main safety-related advantages of the use of this liquid metal.

(i) Sodium does not require a high pressure to keep it in a liquid state at normal operating temperatures (up to 550 °C), its boiling point being as high as about 900 °C. Hydrostatic events (breach of piping, etc.) cannot cause voiding of the core (see, for example, Vossebrecker 1980).

(ii) Sodium cooling allows for relying on natural convection to assure decay heat removal. This has been confirmed by special experiments, e.g. in the Dounreay Fast Reactor, the Prototype Fast Reactor, the Fast Flux Test Facility, Rapsodie, Phénix and Superphénix reactors. Further investigations with a view to the European Fast Reactor are to be performed at out-of-pile facilities such as RAMONA, NEPTUN and ILONA in the F.R.G. A complete reliance on natural convection might render it possible to do without active components such as pony motors for pumps. Sodium makes for a high degree of passive safety, with accidents similar to Three Mile Island extremely unlikely (Vossebrecker *et al.* 1987). Sodium has a heat conductivity about 100 times as high as that of water.

(iii) Irradiation exposure of operating personnel has turned out to be very low in fast reactor plants, because of chemical properties of pure sodium (Broomfield *et al.* 1988).

On the other hand the following drawbacks and problems of sodium have to be faced.

(i) Sodium reacts chemically with water and with air.

(ii) Local sodium boiling can occur as a result of fuel failures and subassembly blockages. A failure propagation from pin to pin and possibly from subassembly to subassembly cannot altogether be excluded.

(iii) Hypothetically, sodium boiling might result in a major accident, if all the shutdown systems and coolant pumps fail at the same time. Certain types of large sodium-cooled reactor cores have a positive sodium void coefficient, i.e. the reactivity tends to increase once large parts of the core have been voided as a result of large-scale boiling.

Let us discuss some of the particular design features, engineered safety systems and research programmes used for FBRS to prevent accidents related to sodium.

#### *Chemical reactions of sodium*

The well-known chemical reactions of sodium with water and air call for extensive engineered safeguards, such as inertization of the primary cell and other provisions against sodium fire, non-radioactive sodium secondary circuits with rupture disks for pressure relief, hydrogen detectors, and cyclones for removing hydrogen.

Failures of steam generators due to tube leaks especially at welding spots have demonstrated that the choice of suitable materials, tests in out-of-pile facilities with subsequent design improvements, and extensive quality assurance programmes are vital for the safe operation of steam generators.

The effects of sodium spray fires are clearly more pronounced than those of pool fires. The Karlsruhe KfK has summarized in a report with comments (Cherdron 1986; Cherdron *et al.* 1988) the results of more recent spray fire tests on its own FAUNA facility and the corresponding programmes in the U.S.A., Japan and France. Unlike pool fires, spray fires cause the sodium to be oxidized mainly in the airborne mode.

Although the spray fire experiments described differ greatly in terms of experimental conditions, the effects do not differ too widely.

(i) The highest temperatures are achieved in the spray cone, i.e. in the direct flame zone. The temperatures encountered here are around 1000 °C. Peaks of around 1300 °C will occur briefly, but are not uncommon.

(ii) Outside the direct influence of the spray cone, the temperatures will be around 600 °C and even higher in some spots, e.g. right above the sodium outlet.

(iii) As a function of the sodium discharge rate, peaks of 1–2.5 bar (1 bar = $10^5$ Pa) overpressure are measured in the first few seconds. At low discharge rates and in large containments, the overpressure will be lower.

In spray fires, at least 30 % of the sodium must be expected to react within a few seconds. For the aerosol generation rate, levels of 150–200 g $m^{-3}$ of mass concentration are not unusual. These high mass concentrations drop to 5 g $m^{-3}$ within approximately 10 min.

It should be mentioned that KfK is also carrying out investigations of sodium–concrete reactions, which might become important for sodium leaks if a steel liner of the concrete fails or is non-existent.

### *Local sodium boiling*

International operating experience of breeder fuel subassemblies has been very encouraging in the last two decades. More than 300000 fuel pins have been irradiated especially in France, the U.K., the U.S.S.R. and the U.S.A. up to considerable burn-ups, with only a few dozen pin failures (Lallement *et al.* 1988). These defects have given valuable hints as to what happens to the cooling conditions in the respective subassemblies. As a rule, these subassemblies could continue operation for a while without melting down. The entrainment of fuel through defective cladding into the sodium coolant was limited to small amounts.

Nevertheless, it was considered advisable to study failed fuel behaviour more systematically in specially instrumented experiments. In this way particular failure mechanisms and phenomena have been and are being elucidated, and failure detection devices have been tested successfully. In some of these in-pile experiments, the fuel rods to be studied carried intended pin cladding defects. In other in-pile experiments, artificial blockages were installed in pin bundles to investigate failure propagation. Perhaps the most important result of these tests is the strong effect of burn-up on fuel failure propagation. Local boiling was detected immediately by the instruments. Out-of-pile experiments with thermite simulating the nuclear energy release serve for supplementing the studies, particularly with a view to materials relocation in an accident. In this way, effects even under rather extreme conditions have been shown to be controllable.

### *Hypothetical whole-core accidents* (HCDAS)

Public debates on breeder reactors especially in Germany have focused on the question: What is the worst-case scenario for an accident? (In the early days of breeder reactor development, a first preliminary study of this question had been performed by Bethe *et al.* (1956).) Therefore, HDCAS moved into the centre of interest in a parliamentary enquiry commission some years ago as well as in the current licensing procedure for the SNR 300 plant on the lower Rhine (Maschek 1982; Maschek *et al.* 1983). In this connection, 'hypothetical' means a probability of occurrence smaller than about $10^{-6}$ per reactor year. The reactivity coefficients of temperature of FBRS are important data for such assessments, see figure 2.

What is the chain of events that might be imagined to result in a HDCA? As a major possible





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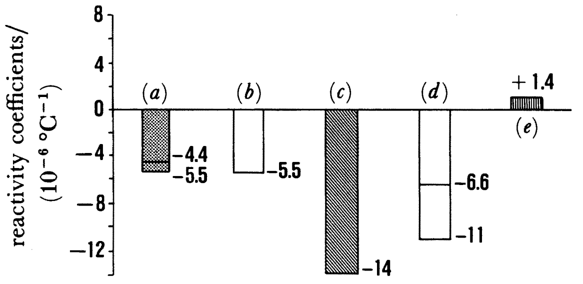


FIGURE 2. Reactivity coefficients of temperature in FBR plants (SNR300). (*a*) Doppler; (*b*) fuel expansion; (*c*) expansion of core grid plate; (*d*) expansion of control rod linkage; (*e*) coolant expansion.

initiator we consider an unprotected loss-of-flow (ULOF) accident in the extremely improbable situation where cooling as well as all scram systems fail to work. This might come about by loss of power to the primary pumps and, simultaneously, failure of the reactor automatic protection systems. Another route for an ULOF would be a local subassembly incident. It would have to propagate in such a way that neighbouring subassemblies are severely damaged and coolant blockages formed to such an extent that the temperature rises by about 400 °C in large parts of the core, finally resulting in sodium boiling on a major scale.

Only in such a phase, the so-called sodium void coefficient will become important. For a large MOX core this coefficient is positive, unless the core has an extremely flat (pancake-shaped) design. Under the extreme condition described above, the positive void coefficient can cause a nuclear excursion. The mechanical energy released will first of all depend on the equations of state of the media converting the thermal and radiation energy into mechanical work, particularly on the vapour pressure of fuel, sodium and steel at temperatures ranging between 3000 and 5000 K. Secondly, the energy release will depend on the time after which the core becomes subcritical. The dispersive effects of the core disruption will lead to a core configuration which stops the chain reaction within milliseconds. This result is typical of various extensive calculations performed with code systems adopted especially from Argonne National Laboratory. The calculations include also effects of a second criticality, which might come about in the course of events.

It should be mentioned that, because of basic differences of design, any comparison between safety features of fast reactors and the RBMK (Chernobyl) types would be out of place (cf. Vossebrecker 1987 and figure 3).

In connection with the SNR300 plant, thorough surveys were done of the international literature on the maximum energy release in 1982 and 1985 (Fischer *et al.* 1985). They gave unambiguous support to corresponding calculations done at the Karlsruhe KfK. They show the following.

(i) The HCDA continues to be a hypothetical accident, i.e. there is no argument in the literature which would give rise to the need to assume a higher probability of occurrence than had been postulated earlier on. With respect to its probability and its consequences, this accident is still covered in the description of the ULOF.

(ii) Recent experimental findings allow the range of parameter studies in boundary case assessments to be restricted much more severely.

(iii) For the SNR300, mechanical energy releases of more than 100–150 MJ still cannot be substantiated by any physically conclusive line of arguments. Mechanical energy releases





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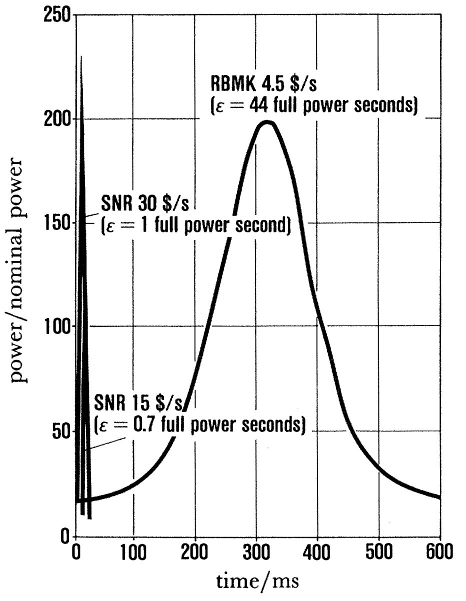


FIGURE 3. Comparison between time dependence of SNR and RBMK excursions.

above 370 MJ (the design basis of the SNR300 primary system) can therefore be excluded to all practical intents and purposes.

The presently available computer codes give a good qualitative description of fuel failures observed in experiments. As far as the initiating mechanisms are concerned, theoretical results are also in good quantitative agreement with experiments, whereas further improvement of the codes is desirable for describing subsequent failure phenomena.

With these reservations, the tools are available to calculate the behaviour also of large fast cores. Estimates on this basis indicate that mechanical energy releases due to HCDAs can be contained with appropriate designs of the primary system of a 1500 MW plant (Gouriou *et al.* 1982; Royl *et al.* 1986, 1987; Nissen *et al.* 1986; Wider *et al.* 1986).

## INTEGRAL FAST REACTORS AND PASSIVE SAFETY

The extensive engineered safeguards and R&D programmes required for safe design and operation of large fast reactors have prompted American laboratories to seek new types of breeder reactor design including integrated fuel cycle processes. Therefore, a new generation of small-sized plants with metallic fuel is under development (Berglund *et al.* 1987). It is to be characterized by an enhanced emphasis on passive, 'inherent' safety. Simplifications in the design, construction and the fuel cycle are expected to be more favourable for public acceptance and costs.

One of the inherent safety features envisaged are passive shutdown capabilities. They shall be discussed here (cf. Hennies 1987) with regard to the dynamic behaviour of fast reactor cores against the following accident-initiating events:

(i) loss of primary coolant flow with subsequent failure of reactor shutdown (ULOF);

(ii) loss of heat sink in the tertiary circuit (steam generators) with subsequent failure of shutdown of the reactor power (ULOHS, see figures 4 and 5).

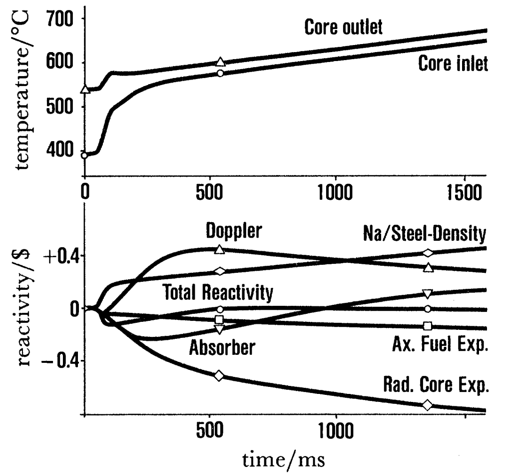


FIGURE 4. Temperatures and reactivities for LOHS in SNR2.

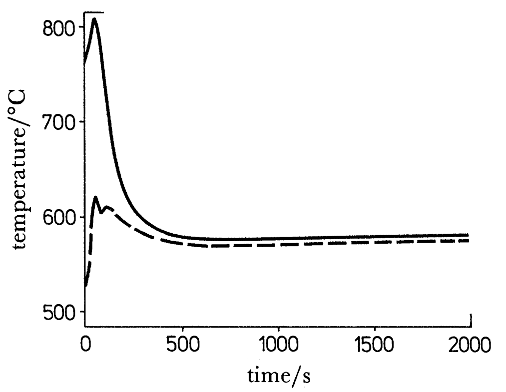


FIGURE 5. PRISM: LOHS without scram: ——, peak fuel; ----, peak coolant outlet.

To drastically mitigate the consequences of these unprotected transient events and to avoid sodium boiling with subsequent core meltdown (HCDA), the main power and temperature reactivity coefficients (Doppler, sodium void, core structural coefficients and the axial expansion of the absorber rod guide structure) are designed such that in case of an ULOF or ULOHS incident the total reactivity of the core becomes negative and the reactor power shuts itself down inherently, while the sodium temperature in the core and in the pool tank increases. Sodium boiling is hereby avoided, if at the same time the primary pumps have a relatively long coast-down characteristic.

The existence of the negative reactivity effect caused by the expansion of the control rod guide structures was measured already during early operation of Phénix and Prototype Fast Reactor. This knowledge was later also applied to the safety analysis of Superphénix (SPX). It is now part of the safety concept of the European Fast Reactor.

In comparing MOX cores and metallic fuel cores with regard to inherent shutdown capabilities, metallic cores at first sight appear to have advantages since they can attain a lower Doppler coefficient and a higher axial expansion coefficient of the fuel, which brings the total reactivity to negative values more reasily. Even more important is the higher thermal conductivity of the metallic fuel which leads to smaller maximum fuel temperatures and fuel temperature differences, which again acts more in support of the inherent shutdown capability. As a consequence, detailed safety analyses show that in the metallic core the coolant temperatures increase up to about 600 °C, whereas in the MOX core, e.g. SPX, the coolant temperature tends to attain a higher temperature level (740 °C at the core outlet and 650 °C in the reactor pool) still rising after $\frac{1}{2}$ h.

However, in both cases the reactor power shuts itself down to about 10% within 10–20 min and no sodium boiling occurs in the hottest core channel. In addition the expansion of the reactor vessel, which appears somewhat delayed as a consequence of the rising coolant temperature, begins to counteract the negative effects of the expansion of the control rod guide structure. This requires a neutronic shutdown after a time period of about half an hour or more, by either absorber rod injection or some other means ($^6$Li injection) actuated by the operator. Because of the somewhat higher temperatures this requirement holds in particular for the MOX core case. The time period for the actuation of shutdown is longer in the case of metal core fuel, but also in this case a final shutdown must be assured.

For MOX fuel cores the situation can still be improved by designing the core such that the Doppler and sodium void coefficients become smaller by a factor of nearly 2. This is possible with heterogeneous or modular core configurations. Furthermore a lower Doppler reactivity can be reached by using smaller values for the linear heat rating. The time period for the pool temperatures to exceed 650 °C will then be longer than $\frac{1}{2}$ h.

At this point the question is whether the still lower pool temperatures and the better inherent shutdown capabilities of metal fuel cores justify a transition from MOX fuel to metal fuel. The European Fast Reactor project considers the still remaining advantage of metal cores as marginal because it has to be balanced against the vast amount of experience available with oxide fuel. In addition the problem of eutectic alloy formation of metal fuel with the cladding at higher temperatures still remains to be carefully examined.

More experience is needed before a final decision can be taken as to which of the concepts turns out to be the best. The European Fast Reactor Steering Committee examined the situation and decided not to modify the strategy of improving the MOX fuel and the corresponding PUREX reprocessing technology. The parallel work going on in Europe, Asia, and the United States is considered as a positive development in the way to commercializing FBRS in the future.

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## *Discussion*

C. B. Cowking (*UKAEA, Risley, U.K.*). Can Dr Hennies comment on the effect of reactor scale on safety? Small reactors appear to have good safety characteristics individually but, of course, more units would be required for a given output. For example, in the U.S. proposals outlined by Dr Griffith the PRISM design would require a nine-fold increase in reactor numbers. Perhaps Dr Griffith would also like to comment.

H.-H. Hennies. The case for small reactors depends in the first instance on their economics and this will vary according to local circumstances. Three main arguments are usually made for small systems.

1. They can provide improved matching to a national grid system. This, of course, depends on the size of the grid. The benefit will only be real with a relatively small grid, such as in the U.S. It may be less relevant for the large grids in Europe.

2. Small reactors have a short construction time. Therefore incur less interest during construction. This will be of regular importance where interest rates are high.

3. There are cost advantages in factory fabrication. This will depend on the size of unit. Reactors of say one tenth of the output of present large designs (i.e. *ca.* 150 MW) are never likely to be economic, whereas those at about 600 MW, which is considered to be about the limit for LWRs, are probably too big for factory fabrication.

In comparing the relative safety of small and large systems, arguments are put forward that it is possible to improve inherent safety in small systems from the point of view of a core disruptive accident. What concerns the public is the consequences of an accident in terms of release of radioactive materials. Improved containment of large reactors can lead to a preferable solution overall.

J. D. Griffith (*DOE*, *U.S.*). The hypothetical accidents that we considered in our comparison of systems were power transients without scram. We believe it is easier to achieve self-limitation with smaller systems.

F. J. Barclay (*Energy Consultant*, *London*, *U.K.*). There are many practical things which can be done to improve safety systems. For example, in the early days of nuclear power in order to improve the shutdown response of small LWRs the design of electromagnets was improved to shorten their response time from hundreds of milliseconds to a few milliseconds, and to reduce the chance that the magnet and armature could stick together (via serrated faces).

D. Broadley (*NNC*, *Risley*, *U.K.*). I want to emphasize that the reliability of shutdown of fast reactor systems is not a problem, as Dr Hennies demonstrated in his paper.

P. Dastidar (*IAEA*, *Vienna*, *Austria*). In my opinion the Chernobyl accident became more serious and widespread because of the consequences of the graphite fire, though according to design the graphite was in inert atmosphere. Is the possibility of extensive fires in sodium being taken into account in the safety assessment of fast reactors?

H.-H. Hennies. There is a great deal of experience in handling sodium and fires in themselves are not a major cause of concern. The main anxiety is a transfer of radioactive species to the outside world. All spaces containing primary sodium are nitrogen filled.




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