FR8001769

International meeting on fast reactor safety technology.
Seatle, usa, August 19 - 23, 1979.
CEA - CONF 5150

FBR'S SAFETY : MAIN RESULTS, PROBLEMS TO SOLVE AND CORRESPONDING PROGRAMMES IN FRANCE

F. JUSTIN, A. MEYER-HEINE, M. NATTA, F. PENET

Commissariat à l'Energie Atomique
B.P. n° 6, 92260, France

ABSTRACT

The safety demonstration of the first large fast breeder reactor in FRANCE was based on existing tests and on a programme of studies. The condition to license was the regular providing of some intermediate results to authorities.

This continuous procedure is developed and examples are given of complementary measures when results are not in compliance with previsions. The start-up of the plant and its operating conditions will depend on the availability of necessary information.

GENERAL

Concerning the present state of LMBR safety studies in France, the overall programme generally follows from the final version of the safety criteria which were used for SUPER-PHENIX. After PHENIX experience on safety evaluation, a first version was proposed by a CEA expert group to licensing authorities ; taking account of amendments suggested by a large group of experts including utility and industry minister experts, the final version was officially sent by authorities to the future owner of the plant in July 1973.

The Preliminary Safety Report was submitted for CREYS-MALVILLE site in February 1974. Taking account of the prototype nature of the plant and the lack of operating experience (PHENIX being in the power-raising-tests phase), the demonstration of criteria application could only stand on partial results. So, a future programme of studies was proposed in the PSAR to complete the demonstration.

After the examination of the PSAR, taking into account the pessimistic character of the overall approach, a favorable advice on the construction of the plant was given provided that :

- the programme of studies be completed on some points
- the results of the studies and related features and eventual modifications be approved by the safety authorities during the construction stage according to a defined schedule.

The main safety features of SUPER-PHENIX [1] and the corresponding key safety issues [2] were presented in the CHICAGO Conference of 1976. Here we shall insist on the main results obtained since that time and on the state of the remaining problems and corresponding programmes. In practice, the main items of this paper have a link to the question of core disruptive accidents either from the point of view of :

- the prevention of thi. type of accident [§ 2,4,5] on which the safety criteria put the most emphasis
- or their analysis [§ 3 and 6] and corresponding mitigation of their consequences by appropriate containment design [§ 6] or complementary safety features [§ 3 and 6].

RELIABILITY AND DEVELOPMENT OF THE SHUT-DOWN SYSTEM

For the shut-down system, the safety authorities insisted on the presence of an auxiliary shut-down system which was already envisaged by the utility and which is now adopted [3]. Further a study of the reliability of the overall protection system was asked for.

The auxiliary shut-down system which is composed of 3 articulated absorber assemblies is now in the in-pile test phase in PHENIX reactor. The first aim was to have a mechanical diverse system, much less sensitive to eventual mechanical stresses due to earthquakes or fuel coolant interaction in a sub-assembly. Moreover, the electromagnet of the prehension system is used for an intrinsic actuation to the nearest point of the absorber rods by abnormal vibrations or sodium outlet excessive temperature which should be very efficient in most of the sequences without scram because of the long delay [see § 3] available before irreversible faulted situations. As all parts of the auxiliary shut-down system are replaceable, future emprovements are possible.

The reliability study of the shut-down system has been completed in 1978 by the utility. The conclusions which can presently be drawn from this study are the following :

- for frequent transients (1 to 10^{-2} per year) the probability of failure to scram is less than 10^{-8}, this taking into account common mode failure in a conservative way and only automatic shut-down. Since the figure which would correspond to the main protection system taken alone is of the order of 10^{-5}, it appears that, in order to achieve the figure of 10^{-8} the presence of the auxiliary shut-down system is essential ;

- for unlikely conditions like earthquakes, large sodium fires, the further design specifications of the shut-down system are such that the reliability of the system remains very high.

Further it appears that even in the case of an unforeseen event , the presence of the auxiliary shut-down system should be a very efficient ultimate safeguard.

As a general rule, it appears that in order to achieve in any case a very high reliability of the shut-down system, the auxiliary shut-down system is essential. For that reason a further systematic reliability study of the mechanical parts of this system is now in progress.

This type of study will be determinant in the future to decide on whether the hypothesis of failure to scram should be kept or not for the containment design.

ACCIDENTAL SEQUENCES

Accidental sequences, which represent general classes of accidents, were studied in detail [8] with parameters relative to SUPER-PHENIX arrangement. Independently of protection system, it was generally supposed that no induced absorber movements took place during the studied period. Consideration has only been given to physical phenomena like :

- thermal inertia of fluids and structures
- run-down inertia of primary pumps
- reactivity feed-back, including thermal expansions.

Four main accidental classes were studied, without scram :

- loss of external power leading to a loss of flow
- loss of water inlet to steam generating units
- control rod movements
- failure of one or two pipes from a primary pump to core inlet.

The parameter of interest was the available time to reach a loss of integrity due to sodium boiling or excessive structure temperature. Some uncertainties have been detected, but it appears that the minimum time to reach irreversibility is at least 10 minutes and could be several hours. The process to reduce uncertainties is underway, the main problems being to validate computer codes on reactor transients and assessing material properties at high temperatures. These favourable results, if they were confirmed, come mainly from the high thermal inertia of the system due to the integrated concept, inertia which is further combined with the inertia of the electrical supplies of the primary pumps. The very long time available makes a simple operator action, manual scram,highly reliable ; however it is necessary to check that in all cases the operator is adequately informed. After scram, a long time is also available to start decay heat removal, as will be seen in the following chapter.

NATURAL CONVECTION IN THE PRIMARY CIRCUIT

From the beginning of the design, a complete loss of normal heat rejection system has been taken into account, i.e. a loss of all secondary sodium circuits. Verifications of the solution extrapolated [1] from PHENIX design revealed insufficient performances. In particular, sodium stratification inside the primary vessel gives only half of the external surface of vessels for radiation towards external cold pipes. So, decision was taken to add a decay heat rejection system for which the following criteria were adopted :

- reactor is scrammed
- primary sodium is circulated at the beginning with inertia of primary pumps, then with natural convection
- the system is independent of secondary sodium circuits
- the system gives a temperature distribution compatible with vessels leak-tightness, which gives practically about 650° C, a subsequent restarting being not guaranted
- the system can be inspected
- the system works with a single failure and in the case of a leak on main vessel
- the system withstands accidental conditions like safe shut-down earth-quake or mechanical energy release in the core region.

The design of the chosen system comprises 4 identical loops with an immersed sodium-sodium heat exchanger connected to an outside sodium-air cooler. During normal operation, electro-magnetic pumps insure uniform temperature distribution, and sodium-air coolers are isolated by air-gates. The pumps can be supplied by electric emergency power.

Calculations and model tests have demonstrated that the chosen system satisfies the above mentionned criteria. Particular tests have shown that assymetry due to the use of only two circuits among four gives acceptable temperature distribution. Actuation of the system needs only opening of air-gates of sodium-air coolers, but the thermal inertia of the primary sodium in an integrated design allows a delayed action, some tens of minutes at least. And if electromagnetic pumps could not be used, the extreme case of complete natural convection can be envisaged : in primary sodium, intermediate sodium and external air.

So, a particular effort has been given to decay heat removal reliability. Until now, a value cannot be given as for PHENIX case [4], but the present design leads us to the conclusion that the decay heat removal function is sufficiently reliable for SUPER-PHENIX arrangement.

SUB-ASSEMBLY ACCIDENT PROPAGATION

Initiation of a local blockage inside a sub-assembly can be imagined from mechanical ruptures, errors, neutron induced strains, foreign debris; nevertheless, efforts made in material selection, controls, tend to decrease the possibility of an important blockage.

Also, individual sub-assembly detection like sodium outlet temperature, and overall signals like burst clad detection or reactivity variations make automatic protection a reliable means to stop the evolution of a defect. The low sodium pollution due to "clean core" concept is intended to allow sufficient sensitivity to burst clad detection. Moreover long term studies are underway to detect local defects by noise analysis of neutron, acoustic or temperature measures [5].

Indenendently of prevention measures, taking the defense-in-depth principle, a complete and immediate blockage of a sub-assembly was assumed. Then, overall detection is not sure because of a possible long delay between fission products emission from the clad and the measure. The detection by temperature increase in neighbouring sub-assemblies is probable, giving an automatic shut-down, but extensive sodium boiling [6] and fuel melting inside the hexagonal wrapper of the defected sub-assembly could then not be excluded. The first scheme of extension passes through a fast sodium vaporization when contacting melted fuel oxide and giving mechanical energy [7]. In the present status of knowledge, it appears that the corresponding probability to distort the core and preventing scram is reasonnably small. But extreme cases could give such deformations on neighbouring wrappers that correct cooling of these sub-assemblies cannot be guaranted. The main uncertainty is the dynamic mechanical properties of materials after irradiation, for which tests are prepared with PHENIX wrappers. The second scheme of extension is a slower one, supposing that mechanical energy is low, the danger being a "thermal" propagation from melted fuel. Out of pile tests and a part of SCARABEE N Reactor Programme aim at giving precisions on the stability of the solid crust of fuel containing melted materials. If the crust is sufficiently strong, taking account of sodium cooling between sub-assemblies, "radial thermal propagation" could be excluded. Even in this case, attention has to be given to possible fragmentation and entrainment of fuel particles into other sub-assemblies : this is called "axial thermal propagation". Out of pile tests are prepared to precise particle distribution and corresponding blockage evolution.

To conclude on this point, it seems that prevention and individual monitoring of sub-assemblies should be sufficient to give a very low probability of a fast extension of a local defect. Assumed extreme cases could give slow thermal extension to other sub-assemblies but we are confident that on-going programmes will give us confirmation that the extreme hypothesis are not plausible. Nevertheless, pessimistic envelopes are taken on molten fuel retention after an extensive accident.

WHOLE CORE ACCIDENTS RISK ASSESSMENT

As it was said for accidental sequences, a big deal of recent studies has shown [8] that a loss of flow accident from loss of power to primary pumps should not result in sodium boiling in a short term. Waiting confirmations from out of pile tests and SCARABEE-N Reactor results, we continue to examine the case of an extensive fuel melting after sodium boiling [9] and disruptive phase, for which calculations [10] will be based on increasingly precise results from CABRI Reactor [11] and fuel equation of state [12] .

Fuel Coolant Interaction and Energetic Core Disruptive Accident

After an extensive fuel melting, an important mechanical energy could come from melted fuel-sodium interaction. Laboratory experiments have shown that defined conditions are needed to deliver some measurable mechanical energy [7]. Some changes are being done on the experiments to make easier code calculations. Also, in pile future experiments will give the effects of fuel irradiation : CABRI will give molten fuel ejection in circulating sodium, SCARABEE-N will give condensing of sodium back on molten fuel, and SILENE reactor will give high and fast energy input into fuel immersed in sodium. Up to now, from out of pile experiments and simulation materials, we have introduced in calculational models the effects of thermal radiation, cooling fluid condensation and fission gas release. Application of these models to reactor case for extensive fuel melting shows that the first estimations of the mechanical energy release are always conservative : a safety factor of two at least is found.

Let us consider now the molten fuel mass during an accident. As was said earlier, we need to neglect favourable feed-back effects to reach sodium boiling before ten minutes. In this case, reactivity ramp depends of materials movements. First results of CABRI runs showed that fuel movements in pre-disassembly phase tend to decrease reactivity ramp. If the dispersion effects from fission gases is confirmed at the beginning of fusion (i.e. solidus instead of liquidus), this will decrease thermal energy. But other important effects have to be known for irradiated materials, like clad rupture moment and localization, melted clad movements, sodium movements, fuel dispersion, ... This confirms the necessity of on-going programmes.

To sum up, an active effort is continued on energetic accidents, but we are confident that mechanical energy release will remain much lower than energies used for containment test validations [13].

Post Accident Heat Removal

Considering accidents where melted fuel can leave the core region, we followed the way of developing the best internal core catcher possible into the available volume. A number of out of pile experiments [14] were made to choose the shape, the level, ... of the core catcher. The studies are not finished, but the general tendencies are :

- the simplicity of the design gives us confidence on overall strength against mechanical energy, localized shoks, limited deformations
- delays due to diagrid strength are important in decreasing decay heat power
- performance evaluations of a core catcher seem in rough agreement with dynamic of accidents resulting in fuel melting.

So, the general behaviour of an internal core catcher seems to add significant containment possibility against improbable accidents, but numerous studies and tests are needed to confirm this [15]. We have to mention particularly programmed tests in SCARABEE N for fuel crust strength and molten pool behaviour, and a possible European test for particle bed parameters.

Attention is also necessary for local point behaviour in core catcher structures, and are under consideration.

CONCLUSIONS

From the original PSAR studies in 1974, a number of evolutions concerning safety have been assessed. An adverse one, insufficient cooling from outside vessel, has been fairly compensated by a positive safety characteristic by adding efficient, redundant system, working possibly with no energy. Auxiliary shut-down system and molten fuel catcher have been improved and are tested. Calculated accidental sequences appear to show that an integrated system is little sensitive to transients, giving numerous possibilities of scram and cooldown. We are confident that present uncertainties on local defects of core cooling and on molted fuel behaviour will not impair containment function.

REFERENCES

1. E. ROBERT, G. LUCENET, A. CHALOT and J. LEDUC, "Main Safety Features of the SUPER-PHENIX Project", in *Proc. International Meeting on Fast Reactor Safety and Related Physics*, Chicago, Oct. 5-8 1976.

2. P. TANGUY, J. PETIT and F. JUSTIN, "Key Safety Issues for Fast Breeder Reactors", in *Proc. (see above mentioned reference).*

3. B. MAGNON et al., "Safe Shut-down Systems Design of the CREYS-MALVILLE Fast Neutron Plan", presented at *ANS/ENS International Meeting on Fast Reactor Safety Technology*, SEATTLE, August 19-23, 1979.

4. G. JUBAULT, A. CARNINO, "Reliability Analysis of PHENIX Decay Heat Removal Function", *(see above mentioned reference).*

5. J. GOURDON et al. "The Detection of Anomalies in Fast Reactor Cores", *(see above mentioned reference).*

6. C. GIRARD et al., "French Studies on Local Blockages in LMFBR Fuel Assemblies", *(see above mentioned reference).*

7. M. AMBLARD, G. BERTHOUD, "UO_2-Na Interactions : the correct 2 experiment", *(see above mentioned reference).*

8. H. FRESLON et al., "Analysis of the Dynamic Behaviour of the PHENIX and SUPER-PHENIX Reactors during some Accident Sequences", *(see above mentioned reference).*

9. C. GIRARD et al., "Sodium Boiling Studies at CEA. State of the art", *(see above mentioned reference).*

10. Y. BAFFOLLET et al., "Calculations of the Loss of Flow Accidents in Large LMBFR Reactors : Influence of Core Parameters". *(see above mentioned reference).*

11. J. DADILLON et al., "The CABRI Facility. Its Objectives and Programme. Results Obtained Until Now", *(see above mentioned reference).*

12. P. COMBETTE et al., "Equation of State for LMFBR Fuel", *(see above mentioned reference).*

13. Y. BLANCHET, "Validation of the Containment Code SIRIUS : Interpretation of an Explosion Experiment on a Scale Model", *(see above mentioned reference).*

14. D. TENCHINE et al., "Experimental Investigation of Heat Transfer by Natural Convection. Application to Reactor PAHR Situations", *(see above mentioned reference).*

15. C. LE RIGOLEUR, G. KAYSER, "An Internal Core Catcher for a Pool LMFBR", *(see above mentioned reference).*