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SPX SIGNIFICANT EVENTS AND WHETHER IT WOULD HAVE HAPPENED ON EFR

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Abstract

In the 13 years since commissioning of the Creys-Malville nuclear power plant, exactly 100 unusual events were recorded on the French and later on the INES scales. The resultant ratio is slightly lower than the French PWR average. This is a noteworthy accomplishment, considering that the plant is a prototype, went through significant design changes, was repeatedly put to test in operating transients and, in addition, holds roughly twice the number of components as a PWR of comparable power. It may be inferred that fast reactors are not more difficult to operate than PWRs, which is also the opinion of most people having taken shifts in both types of reactors. Although Superphenix was labelled a white elephant by public opinion makers, this little known characteristic should remain part of its legacy.

In this period 7 events are registered at the level 1 of the French and INES scales, owing either to misconception, material or operational failure. At the level 2 of the scales, 2 events are registered, which is admittedly quite high. The first one was the sodium leakage from and pervasive cracking of the revolving "drum" of the fuel handling line, in retrospect the result of the choice of steel grade not fully compatible with sodium, which questions the designer's decision making process. The second level 2 event started as a massive air ingress in the primary circuit atmosphere, bringing on a pollution of the sodium up to 15,5 ppm of oxygen (although its significance in terms of corrosion was shown to be minimal). Although this event originated from a maintenance mix-up, it revealed a lack in understanding of sodium chemistry and the inadequacy of the instrumentation.

The operational feed-back of the Superphenix reactor was thoroughly combed for clues to potential anomalies by a working group comprising representatives of the operator, utility, designer and R&D bodies. All the gathered information (together with experience gained from other FBRs, most notably PFR and Phenix) was then analysed in periodic project reviews validating the European Fast Reactor project, mainly in the areas specific to FBR technology (referring to Superphenix' second level 2 event, it led to the addition of a gas analyser). That feed-back process, with complementary contributions and mutual checks of designer teams of various backgrounds, allows an optimistic view of EFR's seaworthiness.

Introduction

Between the start of fuel loading in July 1985 and the start of definitive shutdown operations in July 1998, the Superphénix reactor which powers the Creys-Malville NPP has been effected by 101 anomalies and incidents. These events are analysed based on different characteristic criteria. In particular, we noted:

- the period during which the event occurred, notably in order to highlight the role of the start-up period and to distinguish between operation and shutdown periods,

- the method of detecting the original event, in particular to highlight latent anomalies,

- the safety function affected by the event, together with deterioration or otherwise of this function,

- the original causes of the event to notably differentiate between human and material causes, but also to separate the events due to design or construction errors from those due to reactor operation errors,

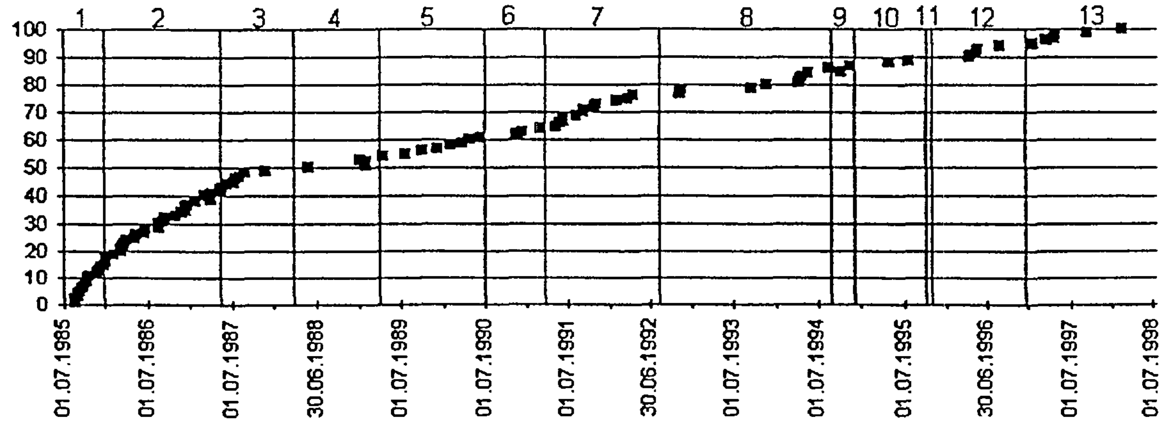
- the importance of the event as graded on the international nuclear event scale (INES).

The main results of this examination are described below and a few graphs given. The events are then listed following a thematic classification.

Lastly those events which are specific to the LMFR technology are again considered from the angle of the concepts retained for the European Fast Reactor (EFR).

1. HISTORICAL TREND

The figure below gives the breakdown of significant events over time.



The main periods of reactor operation are as follows:

1• second half of 1985: reactor loading, first criticality and zero power tests. This period runs from 19 July 1985 (start of reactor loading with fuel assemblies) to 14 January 1986 (first connection to the grid),

2• first half of 1987, limited to 25 May 1987 (reactor shutdown following on from a fuel storage drum incident) : tests on build up to nominal power,

3• May 1987 to March 1988 : reactor shutdown as a result of the drum incident (no.40).

4• March 1988 to April 1989 : reactor shutdown during the administrative procedures prior to restart,

5• April 1989 to July 1990 : operation,

6• 25 June 1900 to March 1991 : reactor shutdown following primary sodium pollution (incident 61), 7• March 1991 to July 1992 : works intended to reinforce the reactor defence against sprayed sodium fires, and administrative procedures prior to restart,

8• July 1992 to August 1994 : public inquiry into restarting by government decision

9• August 1994 to December 1994 : operation

10• December 1994 to September 1995 : reactor shutdown following on from an argon leak from an intermediate exchanger bell (event not in declaration criteria and described in annex 1)

11• September 1995 : administrative procedures prior to restart

12• September 1995 to 24 December 1996 : reactor operation

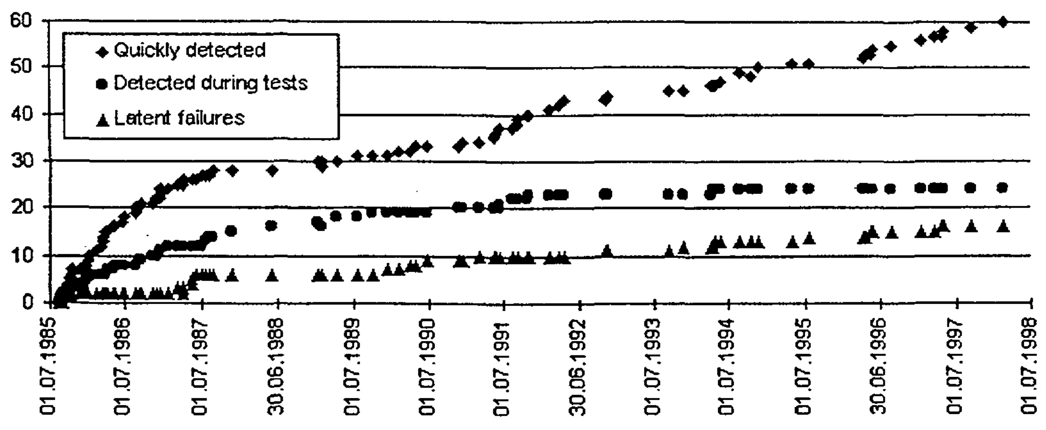
13• since 24 December 1996 : reactor scheduled shutdown followed by legal cancellation of its operation licence followed by definitive decision to shut down the reactor by the government.

The frequency of the events was 7.8 per year over the entire period considered. In this respect three periods can be identified :

- loading, first criticality and zero power testing of the reactor, reactor power build up testing and start of the drum shutdown until August 1987, during which there were 48 events, i.e. an annual average of 24 per year. Although this figure is high, it can be explained by the discovery of a certain number of design and construction anomalies on start-up testing, and by installation take-over,
- long reactor shutdown periods as a result of the drum leak (no.40), pollution of the primary sodium (no. 61), argon leak on an intermediate exchanger and cancellation of the decree authorising operation : 36 events can be listed on these shutdown periods which total 7.75 years, i.e. 4.6 events per year.
- operation of the reactor during the periods noted 5 - 9 - 12 on the figure above : 16 events over a total of 36 months, i.e. on average 5.3 events per year.

It is noted that the overall frequency of the events is similar to that observed on the PWR reactors, i.e. around 8 events declared per year, covering all units after commercial start-up. It is remarkably low after the initial period of start-up and take-over of the installation, whether in the period of shutdown or operation.

2. DETECTION METHOD



60% of the anomalies were detected quickly. These were equipment or operating anomalies the consequences of which were felt in the short term, notably by emergency shutdowns, start-up of safeguard systems[1], control rod lowering. These anomalies continued to appear at a regular frequency until the end of the period considered.

24% of the anomalies were detected on routine inspections : essentially detected during periodic tests (of which it is one of the aims). This detection method was only effective during the first years of operation.

16% of anomalies, of various degrees of seriousness, were discovered by accident or due to their long-term consequences. They are therefore worth more detailed examination:

- no.5 : encountered on account of an inadvertent signal due to incorrect setting of a threshold. The anomaly discovered is the absence of counting systems for detecting fuel element cladding failures on one of the two reactor protection channels;

- no.12 : shutdown without inertia of the two reactor coolant pumps as a result of a loss of one of the two external power supply lines : this incident revealed a common mode defect on the 4 reactor coolant pumps resulting from a design fault in their control system after a first modification was carried out;

- no.40 : sodium leak from the fuel assembly storage drum ; this fault was compounded by not taking into account the alarms in the control room and involves the unnoticed development of a vast network of cracks in a liquid sodium tank;

- no.42 : error in calculating and setting the radioactivity thresholds leading to dome isolation ;

- no.43 : break of sodium-air fan blades discovered by a visitor, which might have however been noted during a routine test;

- no.44 : inoperability of the automatic standby diesel on a safety-related switchboard due to opening of a manually operated circuit-breaker and non-indication of the signal;

- no.57 : multiple failures of the feedwater sodium loop drainage valve positioning systems (generic fault);

- no.59 : absence of a system of auto-blocking support on the residual heat evacuation sodium circuit : the contractor although aware of the anomaly had not declared it;

- no.61 : air intake into the sodium circuit as a result of an error in the maintenance procedures : the incident also highlighted the absence of means for checking the purity of the circuit cover argon;

- no.64 : uncontrolled discharge of slightly irradiated metal waste : the normal inspections did not detect their presence at the site exit;

- no.77 : inoperability of automatic standby by one of the two diesels on a train due to a maintenance error on the electrical batteries;

[1] decay heat removal, reactor confinement, diesel generators

- no.80 : inoperability of the sodium leak detection alarms in the components submerged in the reactor coolant system due to erroneous determination of the signalling thresholds;

- no.83 : inoperability of the sodium leak detection on a feedwater loop storage tank due to inaccurate transmission of data between workers;

- no.89 : definitive loss of a sealed radioactive source (1 μCi) used to monitor the accesses, thrown into a workshop bin by a repairer who was unaware of its nature;

- no.92 : melt of the liquid metal seal on a rotating plug with the reactor critical due to an error on the operation sheet;

- no.97 : inoperability of the sodium leak detection on several sections of a feedwater sodium loop auxiliary circuit due to an error in an operating procedure.

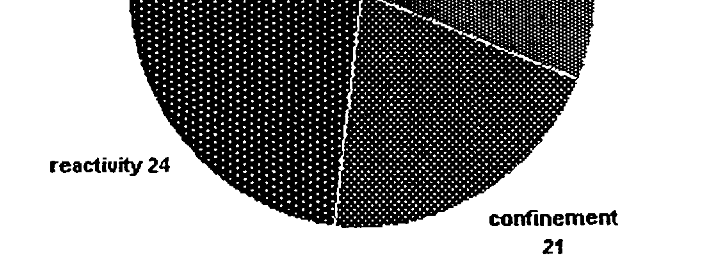
In all, seven of these events involve a design error, three an erection or equipment adjustment defect and eleven indicated faults in the plant operation organisation.

3. SAFETY FUNCTIONS INVOLVED IN THE EVENTS

3.1 Functions generally involved

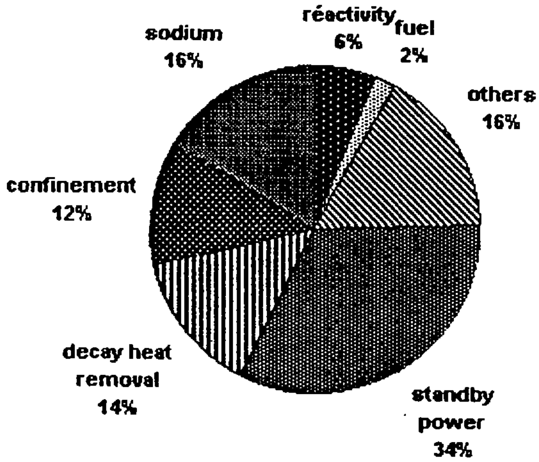
The events are here classified according whether they concern the fuel, radioactivity control, residual heat evacuation, emergency electricity supplies, confinement, control of nuclear material or other functions. In this paper we will first of all examine only the functions involved even if the event does not indicate any deterioration of this function.

We observe that almost two thirds of the anomalies involve the reactivity control function (command of control rod positions), the containment (notably dome isolation and reactor building ventilation) and standby supplies (diesel generator sets, batteries and switchboards that they supply). The anomalies concerning sodium only involve eight events



3.2 Impaired functions

If we exclude a half (51) of events which have not led to a deterioration (real or potential) of a reactor safety function, it becomes obvious that the largest number of really significant anomalies is, like on other installations, associated with the loss of standby electricity supply (which did not have important consequences for Superphenix since there are 4 identical sets), but also that the second cause of safety function deterioration is linked to sodium. This illustrates that the two, sodium-related, incidents which were classified level 2 on the INES scale (nos. 40 and 61) were not isolated events.



Note : in this paper we have covered deterioration of a safety function and not the consequences for safety : in general, for a deterioration of a safety function to impact the safety of the installation, this function must also be required. If we adopted this criterion, the relative importance of the anomalies associated with the sodium and its cover gas would further increase since these must be permanently confined owing to :

- their contribution to the transport of fission and activation, volatile or gaseous, products;

- the risks they engender (sodium fires, anoxia).

3.3 Absence of a safety function degradation

In more than half the events declared, no reactor safety function was degraded. This category includes:

- inadvertent emergency shutdowns, dome isolation, ventilation configuration of the reactor in the reinforced confinement position,
- lowering of control rods,
- three cases of non-compliance with procedures for checking the protection functions,
- three events involving fuel handling,
- two events involving uncontrolled exit of radioactive matter involving a very small amount of activity,
- two cases of human errors making the equipment assuring a safety function inoperable without operator action,
- one case of inadvertent start up of a residual heat evacuation circuit.

This large number of events non-directly significant for reactor safety could lead to the conclusion that the criteria which cover the declaration of significant events are excessively strict. However, these events reveal problems of the plant operation organisation.

3.4 Construction defects without a direct impact on safety

Event no.63 would not be included in the list of those associated with a loss of a safety function if its initiator (collapse of a part of the turbine hall roof under snow weight) had not caused a long loss of one of the external electrical sources.

The argon leak from an intermediate exchanger bell, an event which was outside declaration criteria, caused 8 months' shutdown.

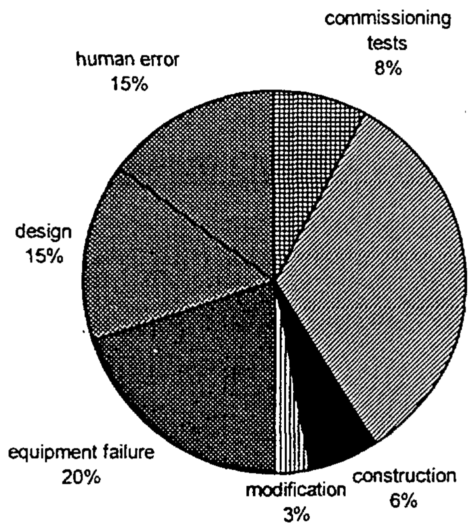
These two events are directly linked to malpractices, undetected through the quality assurance system. The management structure chosen for building the plant (mainly in the view of distributing the contracts between participant countries) rendered the quality control probably more difficult. It should be noted yet that, as a whole, building deficiencies contribute only marginally to the number of significant events.

4. CAUSES

The causes have been listed according to their origin as follows :

- installation design,
- installation construction,
- execution of start-up tests,
- modifications executed since start-up,
- methods associated with normal operation of the installation,
- equipment failure,
- human errors during execution of a task.

The following graph gives the global percentages, while the table indicates, for each given period, the annual frequency of events.



Period (operation shaded)	1	2	3	4	5	6	7	8	9	10	11	12	13
Equipment	6	5.3	0	2.2	1.3	1.3	1.5	1.8	3	0	0	0	0
Design	8	3.8	1.2	1.1	2.5	0	0.8	0.9	0	0	0	0	0
Construction	4	0.8	0	0	2.5	1.3	0	0	0	0	0	0	0
Commissioning	2	3.8	2.4	0	0	0	0	0	0	0	0	0	0
Modification	4	0	0	0	0	0	0	0.9	0	0	0	0	0
Operation	8	2.3	3.6	2.2	1.3	1.3	5.3	3.4	3	2.7	0	3.2	1.7
Human error	4	3	0	0	1.3	0	0.8	0.9	3	0	0	0.8	3.4

This classification is partly subjective since in most cases there is not a single cause of the incident. Nevertheless, out of all the events occurring at Creys-Malville, the following trends can be identified:

- Events attributed to the design, the construction of the installation, and its commissioning are preponderant at the start of the plant life, but only represent 29% of the total. The phase of installation take-over has therefore correctly fulfilled its role.

- Equipment failures and human errors together caused one third of the events. Although precautions (as a matter of quality assurance) can be taken in this respect (selection of equipment, preventive maintenance, training, preparation for works, etc.), these two types of events are nevertheless difficult to avert totally.

- The definition of modifications on the equipment brought about 3 events:

    • no.4 : deformation of a new fuel assembly (before loading) : the modification executed as a result of incident no.1 (addition of a pointer to the biological protection hood for new fuel sub-assemblies) have caused this incident,

    • no.12 : no-inertia coast-down of reactor coolant pumps : a modification to avoid the alternator supplying their power operating in motor mode had been previously implemented on the four drive sets by using information supplied by a non-emergency supplied electrical source,

    • no.80 : inoperability of the sodium leak protection alarms in the gas spaces : the thresholds validating the alarms had been set erroneously by the team responsible for designing the modification and the error was detected thanks to the critical observations of an operator.

    The increasingly strict approach to the management of modifications has, since 1994, avoided such events recurring in spite of the many modifications made.

- Lastly, 33% of events can be attributed to the methods or the usual organisation of the reactor operating staff, including maintenance. Out of these, 7 occurred during reactor shutdown for purification of the sodium. However, there is no apparent correlation with the reactor status.

These events, which are not, generally, specific to the Fast Reactors technology, could be reduced in frequency thank to the progress made in matters of man-machine interface and equipment testability.

5. SCALE OF SERIOUSNESS

All the significant events occurring on Creys-Malville have been classified using the criteria on the INES scale of nuclear events. It must be observed that for events occurring prior to application of the French scale, the subsequent classification cannot take account of the discussions which have taken place in certain limit cases. However, this classification benefits from all the experimental classification work which had been carried out by IPSN and EDF when defining the French scale.

The following 2 incidents characteristic of the fast reactors are classified level 2:

- sodium storage drum leak (no.40)

- reactor coolant system pollution (no.61).

These two incidents fully reflect the prototype nature of the installation.

The following six events are classified level 1:

- no.12 : emergency shutdown with loss of inertia on two reactor coolant pumps
- no.18 : heating of a fuel assembly,
- no.56 : fall of a temporary lifting crane,
- no.63 : collapse of the train A turbine hall roof,
- no.88 : simultaneous opening of several containment barriers,
- no.92 : fusion of the liquefiable metal seal on the large rotating plug, reactor critical.

Three of these events are associated with erection of the installations, the other three with operation.

The analysis of these 8 events according to the criteria examined previously are given in the following table.

It will be observed that none of these events did endanger reactor safety. On the other hand one (no.40) definitely compromised any hope to cover the plant operating costs, for the repair could not restore hot irradiated fuel storage capacity and compelled the operator thereafter to wait for the decay power to subside before unloading the core, meaning long outages.

Four events brought about long idling periods (nos 40, 61, 63 and the leak of the intermediate heat exchanger bell) and contributed to alienate decisive sectors of opinion, including in the nuclear field.

The description of these eight events is attached in the annex 1.

NO	PERIOD	DETECTION	SAFETY FUNCTION	CAUSE
40	2	late	sodium confinement	design
61	5	late	cover gas integrity	design
12	1	late	core cooling	modification
18	2	commissioning test	core cooling	operation
56	5	equipment test	none	construction
63	6	instantaneous	none	construction
88	10	instantaneous	reactor confinement	operation
92	12	late	reactor confinement	operation

The two incidents (nos.40 and 61) which originated in a design failure have been taken into account in the EFR project as follows:

- By the choice of material used for vessels (which implies that a thorough explanation has to be given to incident 41, including its reproduction in laboratory conditions), by anchoring the safety vessel in the vessel pit, itself constructed in sodium-refractory concrete,
- By permanent inspections of the chemical composition of the gas cover.

6. GROUPING OF THE SIGNIFICANT EVENTS

The significant events occurred at Creys-Malville can be grouped in coherent families, each incident being referred to only once according to its most important criterion.

a - Justified implementing of a safeguard system (residual heat evacuation circuits, dome isolation, diesel generator sets)

no.6 : destruction of a feeder circuit breaker to the 6.6 kV switchboard

no.54 : voltage loss a 6.6 kV switchboard

no.63 : collapse of the turbine hall roof, train A

no.81 : loss of external electricity supplies to train B

no.93 : tripping of stage 1 dome isolation

In the first four cases this involved start-up of one or two generator sets.

b - Sodium leak

no.40 : sodium storage drum leak

no.60 : sodium leak on a feedwater loop auxiliary circuit tee junction

no.65 : sodium leak on the plugging indicator of a residual heat evacuation circuit

This type of incident is specific to fast reactors and justifies them being placed in a separate category. It will be observed that the risk involved in sodium fires motivated the regulating authority to demand, specifically after the Almeria accident, that a sprayed sodium fire resulting from a complete and sudden rupture of a main secondary pipe be taken account of.

c - Physical phenomena leading to reactor shutdown (emergency or intentional shutdown) :

no.18 : fuel assembly overheating

no.24 : emergency shutdown due to temperature fluctuations

no.55 : emergency shutdown due to reactor coolant pump stop

no.61 : reactor coolant pollution

no.79 : failure of a reactor coolant pump coupling

d - Failure of the reactor protection system (bringing into question its operation when solicited)

no.2 : non-lowering of three complementary shutdown system rods

no.5 : absence of train B cladding break detection channels

no.62 and no.69: no-voltage relay anomalies

The two first events are associated with the reactor start-up tests. However, the two last ones, although they initiated an emergency shut-down thank to the "fail-safe" design of the reactor protection system, present the problem of its reliability.

e - Failure of another safety function (evacuation of residual heat, containment)

no.9, no.26 and no.46: diesel generator sets heating

no.12 : non-inertia shutdown of reactor coolant pump

no.13 : non-coupling of a pair of diesel generator sets

no.15 and no.42: poor adjustment of the dome activity threshold

no.36, no.47 and no.50: loss of battery capacity

no.37 : no-voltage on 6.6 kV standby switchboard

no.43 : break of sodium-air exchanger fan blades

no.44 : voltage loss on emergency-supplied 6.6 kV switchboard no.49 : non-start-up of a diesel generator

no.53 : non-closure of a steam generator water valve

no.57 : anomalies on feedwater loop drainage valve

no.59 : absence of a self-blocking support on a residual heat evacuation circuit

no.67 : inoperability of a sodium-air exchanger

no.68 : inoperability of a diesel generator

no.71 : loss of train B standby raw water circuit

no.73 : inoperability of a diesel generator

no.77 : inoperability of two diesel generators on one train

no.80 : inoperability of sodium leak detection alarms in gas spaces immersed in the primary sodium

no.82 : through crack on the balancing line of secondary loop rupture discs

no.83 : inoperability of sodium leak detection on the storage tanks

no.84 : inoperability of a residual heat evacuation system

no.87 : inoperability of a secondary loop argon sweeping circuit

no.88 : simultaneous opening of several containment barriers

no.92 : liquefiable metal seal fusion on large rotating plug

no.94 : loss of intermediate containment integrity by gas activity control circuit

no.95 : inoperability of diesel set

no.96 : inoperability of chilled water production system

no.97: inoperability of sodium leak detection on 8 sections of a secondary loop auxiliary circuit

no.98 : loss of slab cooling

no.100 : shutdown of reactor coolant pump

These events are significant of potential or effective deterioration of a safety function (evacuation of power in most cases).

f - Events concerning handling (of assemblies or components) :

no.1 : drop of a new fuel assembly into its pit

no.4 : deformation of a new fuel assembly no.25 : break of an observation window on the small cask

no.56 : fall of a temporary lifting crane

The separation of the installation into two parts, reactor and handling, together with the specificities of this type of events, allows this category, which is relatively diverse as to the events it covers, to be defined.

g - Emergency shutdown orders or inadvertent rod lowering (without anomaly on the reactor protection system) :

no.7 : shutdown by the power/reactor coolant flow protection

no.8 and no.11 : emergency shutdown tripped by the train B cladding break detection channel

no.10 : inadvertent manual emergency shutdown

no.14 : emergency shutdown tripped by the seismic channels

no.16 and no.17 : emergency shutdown tripped during works on the reactor protection system

no.19 : lowering of a control rod

no.20 : lowering of 3 rods on the diverse shutdown system by permanent supply cut off

no.21 : lowering of 3 rods of the diverse shutdown system during a fast shutdown

no.22 : emergency shutdown by the neutronic measurement channels after a fast shutdown

no.23 : lowering of 2 control rods

no.27 : emergency shutdown by non-inhibition of the neutronic power low level measurement channels

no.28 : emergency shutdown by the neutronic measurement channels on movement of a physical sensor in the core

no.29, no. 30 and no.32 : lowering of one or two diverse shutdown system rods

no .33 : emergency shutdown by the neutronic measurement channel after a fast shutdown

no.34 : lowering of no.2 main control system control rods

no.39 : emergency shutdown order by the power/reactor coolant flow protection

no.41 : emergency shutdown due to works on the computer monitoring the heating of train B reactor

no.51 and no.52: emergency shutdown due to a fault on a power/reactor coolant flow protection

no.85: emergency shutdown without initiator

Generally, these events translate faults on materials or organisation design (man-machine interface) but do not have physical consequences for installation safety.

h - Inadvertent start-up of a safeguard system :

no.31, no.35, no.38 and 45: dome 2nd stage isolation

no.48 : dome 2nd stage isolation

no.58 : dome 1st stage isolation

no.66 : inadvertent start-up of a residual heat evacuation circuit

no.70 and no.72 : dome 1st stage isolation

no.74 : dome 2nd stage isolation

no.75 : dome 2nd stage isolation

no.76 : dome 1st stage isolation

no.78 : dome 2nd stage isolation

no.99 : excessive pressure drop inside the reactor building

Same comment as for the previous category.

i - Non-compliance with procedures (tests guaranteeing correct operation of systems associated with reactor safety)

no.3 : incorrect estimate of the critical mass

no.86 : non-execution of control rod translation force measurements

no.90 : operation of 2 control rods without recording the translation force

no.91 : non-compliance with the delays for executing periodic tests of fire detection

j - fault in the management of nuclear matter

no.64 : uncontrolled discharge of slightly irradiated steel

no.89 : definitive loss of a sealed 1 μCi radioactive source

7 SIGNIFICANCE OF EVENTS FOR THE FAST REACTOR TECHNOLOGY

Out of 101 events considered (including the intermediate exchanger bell argon leak) :

- **66 can be considered non-specific to fast reactors,** i.e :

. in view of their causes (but not necessarily their effects), they could have occurred on PWR reactors,

. they can be prevented by improving the quality approach in terms of the design or procedures and/or by generalising the recent approaches relative to the design of control and instrumentation systems and the man/machine interface (testability, ergonomics).

In particular, they consist of faults on the electrical supply, diesel generators and batteries, faults on electrical connections, routing errors, in-service works or calibration errors.

One could be tempted to mitigate this favourable observation (2/3 of events are not specific to fast reactors) by the fact that the number of components is twice more and that certain circuits are more complex (primary and secondary gas circuits) as compared to PWRs.

The overall results do not however confirm this reservation which would imply that, for an equivalent design (control and instrumentation, man/machine interface), the probability of failure should be identical for individual components and thus roughly double for all the systems and functions.

This is probably due to the fact that, apart from the control of reactivity and the protection of the steam generators, the accidental transients on fast reactors are much slower than those on the PWRs and that there are therefore fewer systems and actuators requiring both very fast and very reliable start-up.

- **35 can be considered specific to the fast reactors**. Among these a distinction must be made between :

18 SPX specific events which correspond to :

. steps not adopted for EFR (biological protection bell, dome, intermediate exchangers bell, ensuring leaktightness between hot and cold headers),

. procedures improved and validated on SPX (e.g. lowering of clusters before changeover to auxiliary motor) : these procedures must obviously be retained or adapted for future fast reactors.

- **17 "SPX & EFR generic events"** (see annex 2) which correspond to :

. non-quality of the equipment or fabrication:

plug left in an assembly leg,

fan blade break,

primary pump coupling break,

. system (control and instrumentation) or component design improved then validated on SPX :

reactor coolant pump control and instrumentation,

cask observation window design,

control and instrumentation on the fast decompression isolation sequence,

design of plugging indicators preheating,

. points on which studies are required :

measurement of the primary flow and tripping of associated protections (2 events),

reactor measurement sensitivity to cable impact due to very currents (2 emergency shutdowns by the train B cladding break detection channel),

fluctuation of the temperature at the sub-assembly outlets (first fertile row),

. generic technological aspects :

design of a main vessel (storage drum),

conditions of hot sodium/cold sodium mixture,

checking of the primary sodium impurity level,

design of dead legs subject to sodium aerosols

design of the cover gas circuit.

8 THE PROJECT REVIEWS OF THE MAIN SYSTEMS OF THE FBR TECHNOLOGY BASED ON OPERATING FEEDBACK

Since 1995, a systematic approach to check the design of the EFR systems has been carried out in the framework of the Project Reviews of the main systems and components based on operating feedback.

These PR have concerned :

- Control rod drive mechanisms
- Main sodium pumps (primary and secondary)
- Intermediate heat exchangers
- Primary sodium auxiliary systems
- Primary argon auxiliary systems
- Sodium circuits
- Steam generators

Each PR has been organised practically in the same way, as follows :

- Introduction, including design specifications and safety requirements of the systems surveyed.
- A survey of the characteristics of the process fluids (if appropriate).
- A comparison between EFR design and the existing (or having existed) designs : mainly PFR, PX, SPX.
- Recommendations, proposals for EFR (and future) design.
- Conclusions.

In order to prepare each PR, certain specific studies have been carried out :

- Comparison of EFR design to existing designs.

- Operating feedback of the same or similar systems from plants in operation (or having been operated).

- Main characteristics and the main feedback of the process fluids.

Each PR has given rise to a synthesis. Hereafter are indicated only those of the conclusions which derived from the feed-back of Superphenix.

PR on control rod drive mechanisms

DSD mechanisms

These mechanisms are mainly derived from the Superphenix DSD mechanisms, so it has to accommodate the results of in-service inspection and maintenance works on the DSD mechanisms and the DSD absorber rods at Superphenix.

PR on Intermediate Heat Exchangers

The conclusions are mainly that EFR intermediate heat exchangers design globally integrates features from Superphenix intermediate heat exchangers, the design of which already benefits from experience gained at Phenix.

PR on primary sodium auxiliary systems

Concerning the "principle design sheet", there are no particular remarks from the operating feedback : integrated electro-magnetic pumps at Superphenix are reliable; an external circuit makes maintenance operations easy.

Adding to the system an integrated plugging meter, improved and optimised as regard to the Superphenix one (200 l/h instead of 50 l/h), might be considered.

This way the two main functions of oxygen content measurement and cold trapping will be independent.

PR on primary argon auxiliary systems

Operating feedback suggests having the working pressure range for the circuit as low as possible. This is easily attempted with a so-called constant pressure circuit (such as Phenix and Superphenix). There is also a concern to limit the sodium leak hazards from the reactor upper closure by geyser effect ; it needs a low argon cover gas pressure. Moreover, it must be possible to lower the relative pressure of the cover gas circuit near zero, especially during refuelling and maintenance and primary components handling, in order to avoid (or to substantially limit) argon leaks to the outside.

Hence it has been suggested that the EFR argon reactor cover gas circuit design return to a (almost) constant pressure circuit, bringing benefits from operating feedback.

PR on sodium circuits (secondary, auxiliary, decay heat removal)

Phenix and Superphenix secondary circuit designs are simple; filling and draining are easy. Moreover, at Phenix, most of nozzles of small piping have been eliminated from secondary pipe-work during renovation works. It follows that the need to have a high point vessel, and a degassing circuit (as actually in design) has to be considered.

In the current EFR design, the slope of the secondary piping is towards the intermediate heat exchangers. This arrangement, which does not allow complete draining of secondary circuits at intermediate heat exchangers, must be avoided (slope to intermediate heat exchangers for the cold leg and from the intermediate heat exchangers for the hot leg as at Phenix and at Superphenix).

Based on French operating experience and usual operating procedures, it has been suggested that the main sodium pumps of EFR Consistent Design be equipped with pony motors in order to assure sodium forced convection in secondary circuits in any configuration.

As the main pollutant of the secondary circuit is hydrogen, the secondary auxiliary circuits have to be optimised in order to measure hydrogen contents and in order to trap hydrides; it would be useful to have data about tritium diffusion from the primary circuit.

PR on steam generators

From the operating feedback from British, French and Russian steam generators which have equipped sodium-cooled fast reactors, it can be confirmed that it is possible to manufacture and to operate large and reliable 'integral once through' steam generators without major difficulties, and respecting high Quality Assurance rules.

The helical steam generator design can be considered favourably based on the Superphenix operating feedback,

The selection of a helical steam generator as a fallback option for EFR is therefore fully justified .

9. CONCLUSION

The frequency of the significant events which occurred in the Creys-Malville facility (with the Superphenix reactor), equal to 7.8 per year for the entire period considered including the start-up period – i.e. 6.8 per year when only considering the period subsequent to first coupling to the grid (1986), is similar to the one recorded for the PWR reactors which is about 8 per unit and per year.

Around half the significant events had no consequence, whether effective or potential, for reactor safety.

On the other hand, it is observed that in 16% of the events, the installation was in a degraded condition without the operator noticing this. It may be noted that seven of these events occurred after June 1990, date of the air ingress detected late in the primary circuit, i.e. around 1 per year or 18% of the events occurring during this period.

48% of the events can be attributed to an imperfect man-machine interface, to the methods, usual organisation of reactor operation or human failures. These factors represent all the causes of events occurring during the last three years, notably the period of continuous operation of the reactor in 1996.

It is observed that the significant events which involved the electricity supply are in a majority. This observation underlines the importance which has to be attached to the integrity of the electricity supplies.

Lastly, the events associated with the presence of sodium are the second largest category (16%) among those which affect the safety functions and the largest in the length of resulting plant idling and in seriousness (two events classified level 2 on the INES scale). This means that contrary to the opinion commonly held during reactor design, the fast reactor technology was not totally mastered.

If one adopts the point of view of future fast reactors, particularly the EFR project studies, the analysis shows that, out of 101 events :

66 may be considered as "non-LMFR specific" i.e.

from their causes they might as well happen on PWRs;

they can be prevented by a strengthened approach of quality assurance in design or procedures and/or by an extensive use of recent concepts relative to instrumentation and control systems and man-machine interface (testability, ergonomics).

18 events correspond to specific Superphenix features not retained for EFR, or to procedures improved since the event occurred.

17 "generic" events correspond either to designs (of systems, equipment) which may have be improved and validated on Superphenix, or to special technical notions, a few of which still require some development.

A systematic review of the operation of Superphenix allowed to validate several concepts which were then retained for the EFR project in the most characteristic fields of liquid metal circuit technology: control rods, pumps, intermediate exchangers, sodium and argon auxiliaries, sodium circuits and steam generators.

The experience of Superphenix thus furthered the technical quality of the next fast reactor generation.

Annex I

DESCRIPTION OF EVENTS CLASSIFIED LEVELS 1 AND 2 ON THE INES SCALE FOLLOWED BY' DESCRIPTION OF THE ARGON LEAK FROM AN INTERMEDIATE EXCHANGER BELL

EMERGENCY SHUTDOWN TRIPPED BY LOSS OF 225 KV AUXILIARY POWER

No.12 **Date** : 22 November 1985 at 23.00

1 - NATURE OF THE INCIDENT

Emergency shutdown tripped by loss of 225 kV auxiliary power.

2 - SEQUENCE OF EVENTS

Reactor was critical at a temperature of 195°C. The reactor coolant pumps were operating on the main motor on 110 rpm.

Train A was supplied by the 400 kV network, the 225 kV auxiliary line was available.

Train B was supplied by the 225 kV auxiliary line, the supply by the 400 kV network being locked out for maintenance works and elimination of the neutral switch.

At 23.00.58, a defect on the 225 kV auxiliary line caused the two circuit breakers on line LGR A and B to open.

At 23.01.03 the min. voltage threshold on the 6.6 kV LGA C and D and LHA C and D switchboards on train B was reached tripping emergency shutdown AU2 and start-up of the two train B diesel generators.

Train A remained supplied normally by the 400 kV system.

At 23.01.05, reactor coolant pumps RCP C and D went into reverse rotation as a result of the loss of inertia. This loss of inertia is due to opening of the self-exciting circuit breaker 03 JA of the variable speed alternators supplying the motors of these two pumps.

Secondary pumps BCS C and D were driven by the auxiliary motors.

At 23.10, procedures I 14 C and D applicable in the event of loss of non-emergency supplied switchboards LGA C and D came into force.

Closure of the obturators on the two reactor coolant pumps C and D took place at 23.28.

At 01.09, dispatching indicated that the 225 kV line would be inoperable at least for the night. It was therefore decided to re-supply train B as quickly as possible through the 400 kV line (locked out).

The end of lockout of the 400 kV train B took place at 02.00.

The instrumentation and control re-qualification tests for the train B power evacuation system began at 4.00.

Re-energising the 400 kV train B transformer took place at 07.00.

Train B switchboards were re-supplied from the 400 kV network at 08.00 and the train B diesels were stopped.

3 - STEPS TAKEN

Application of incident instruction I 14: loss of non-emergency supplied switchboards.

Re-supplying in the shortest possible time of the train B switchboards by the 400 kV system.

Drainage of the sodium circuits since preheating function not emergency supplied (MAS 0).

Cancellation of the planned tests (reactor critical) pending the explanation and the elimination of anomalies having caused pump reverse operation.

On 23 November 1985, a helicopter search was initiated to detect the line fault. This fault was due to the accumulation of frost on the cables (one phase in contact with a tree).

4 - NO-INERTIA STOPPING OF REACTOR PUMP

4.1 - STATUS OF THE SYSTEM AT THE ORIGIN

Since excitation of the alternators of the pump drive sets, compound type, can be interrupted by circuit breakers RCP* 03 JA and straps had been installed so that these circuit breakers could open as soon as the 6.6 kV supply of RCP* 01 MO tripped out, the system for progressive coast-down of the primary coolant pumps was not operational (common mode defect).

4.2 - FIRST MODIFICATION

Relaying modifications were carried out to overcome the anomalies observed during the tests: non-opening of RCP* 03 JA on loss of voltage, motorisation of the alternator on take-over by the auxiliary motor. These modifications can be summarised as follows:

- installation of a section to take into account "auxiliary motor tripped in", instead of a measurement of the motor charge current, which is not very reliable due to the small difference between the charge and no-load currents,

- to avoid operation of the alternator in motor mode, opening of RCP* 03 JA is controlled by the following conditions:

- RCP* 01 MO main motor stopped,

- and, either alternator speed less than 45 rpm, or auxiliary motor RCP* 02 MO started.

The "main motor not started" and "alternator speed less than 45 rpm" conditions also open RCP* 01 JA (which leads to opening of RCP* 03 JA).

4.3 - INCIDENT

On the incident, RCP* 03 JA opened since the following conditions were met :

- speed less than 45 rpm (threshold was generated from the non-emergency supplied LKB* switchboard, the threshold being reached by a voltage drop on train B),

- RCP* 01 JA open due to voltage drop on LKB*,

- RCP* 01 MO stopped (no 6.6 kV at switchboard LGAC/ D).

4.4 - NEW MODIFICATIONS

The insertion of additional relays could reduce reliability. Moreover, on a definitive voltage loss on switchboard LGA, maintaining the excitation contacts closed cannot damage the set. In the other configurations, the risks of alternator operation in motor mode are eliminated by opening RCP* 01 JA through contacts 02 and 08 XR. It was therefore considered preferable to eliminate relay 16 and 17 XR and the relay contact 20 XR on the start-up channel since the protection of the alternator in terms of the motorisation risks is provided by RCP* 01 JA. Maintaining RCP* 03 JA in a closed position when the pump is at the end of slow down or in reverse rotation does not present any particular problems. On a loss of voltage, the inertia slow down of the pumps is thus maintained as long as the auxiliary motor is not started.

ABNORMAL HEATING OF FUEL SUB-ASSEMBLY COEC 3200

No.18 **Date** : Tuesday 7 January 1986

1 - NATURE OF THE INCIDENT

Since the start of power build up, heating of fuel assembly COEC 3200 situated in the core at position 41/25 was detected clearly above that of the other assemblies and monitored without ever reaching the TRTC alarm threshold.

2 - ANALYSIS

A first examination of the sheets of the inspection carried out at the plant before entering the new sub-assembly in the handling drum indicated that the loss of air pressure of this sub-assembly is in the high range of the criterion : loss of pressure equal to 16 mbar for an acceptability of 12 ± 5 mbar.

An enquiry carried out in parallel with the manufacturer COGEMA showed on 7 January in the evening that a natural rubber protective plug could have been forgotten in the leg of the assembly.

A set of consistent correlations confirmed this hypothesis and showed that this assembly, forming part of the batch of four prototypes whose legs had been re-machined, was the only one involved.

3 - STEPS TAKEN

The reactor was stopped as soon as the results of the COGEMA enquiry were known and the faulty assembly was discharged to the storage drum.

An enquiry into the organisation of the manufacturer's quality was carried out in parallel by the main constructor NOVATOME-NIRA and EDF's manufacturing inspection service (SCF).

Chemical analysis of the plug consisting of 98% natural rubber highlighted the presence of a low content of impurities which, when diluted in the 3500 tonnes of reactor coolant sodium, led to a proportion lower than the limit permitted for nuclear quality sodium.

CEA carried out a series of static sodium tests to study the behaviour of the plug at high temperature.

Sub-assembly COEC 3200 was examined before being stored in cask IL 49 until dispatched to the LSAI laboratory (irradiated assembly monitoring laboratory at Marcoule) in June 1989.

The criteria for checking new fuel sub-assemblies before loading by verifying the loss of air pressure were fine-tuned. An endoscopic examination inside the leg has been added.

4 - CONCLUSIONS

The investigation by SCF demonstrated that the presence of a plug is only possible on assemblies re-machined to modify the self-orientating legs. CEA concludes from the tests that, at 400°C with sodium, pyrolysis of the plug is complete and results in the formation of an "amorphous" coke and gas products. Endoscopic examination carried out on 16 and 17 June 1986 confirmed the hypothesis of a forgotten plug. The analysis carried out on the residue samples confirmed this oversight.

SODIUM LEAK IN THE SPACE BETWEEN STORAGE DRUM VESSELS

I.S. No.40

Date : 3 April 1987

1 - NATURE OF THE INCIDENT

- 8 March 1987: leak detection alarm in the space between the storage drum vessels: this alarm is not confirmed locally.
- 9 March 1987: reactor pit bottom intermittent leak alarm : investigations ongoing.
- 31 March 1987: as a result of the investigations, cold nitrogen sampling in the inter-vessel space did not reveal any traces of sodium. Nevertheless, the balances for the sodium levels of the drum and the storage tank show that there are 20 m^3 of sodium between the vessels.

2 - IMMEDIATE ANALYSIS

The leak was confirmed and the first investigations to locate the leak were started (leak at level 14.6 m ± 1 m).

3 - IMMEDIATE STEPS

Evacuation of the equipment (new fuel, rods, COEC 3200, irradiated dummy assemblies) was begun as soon as possible to enable the drum to be emptied.

An additional barrier was installed relative to the sodium which could leak from the retention tank and the mechanical strength of the drum safety tank in the new conditions created by the leak were confirmed.

A strategy was developed to define the reactor operating conditions during the period of drum inoperability.

Partial temporary operation of the APEC (fuel evacuation workshop) for storage of the " steel " assemblies withdrawn from the storage drum and placed in containers.

Storage drum emptying in conditions allowing pre-location of the leak.

This took place between 27 August 1987 and 9 September 1987 and enabled the leak to be located (infrared thermograph, xenon and helium detection in the inter-vessel space).

4 - CAUSES OF THE LEAK

The leak was caused by a horizontal crack around 60 cm long on the lower angle welding bead which secures a plate. This rectangular metal plate welded on the inner face of the main storage drum vessel contributes to maintaining the drum sodium cooling circuit.

After discharging the assemblies contained in the drum and emptying, in September 1987, the sodium contained in the drum main vessel, samples by cutting the metal from the plate which was at the origin of the leak were taken.

The results obtained at the end of 1989, and the laboratory tests and examinations identified the most probable scenario involving the nature of the drum steel (ferritic 15 D3) and the simultaneous presence of three factors : the existence of start sites (micro-cracking) in zones of high hardness, residual stresses close to the elastic limit of the material, and lastly, the contributions of hydrogen which allowed the brittling phenomenon to occur.

However the cracking phenomenon could not be re-created in laboratory using specimen of the same steel grade in similar conditions with the presence of these three factors.

5 - CONSEQUENCES OF THE INCIDENT

The progress of the incident showed the merits of having a retention vessel around the drum main vessel to collect and confine any sodium leak. It also highlighted that this step could be improved by additional steps concerning leak control, monitoring of the integrity of the retention tank and the actions needed in the event of a leak from the second tank.

Processing of the drum incident thus led to similar principles being applied to reactor vessels and reinforced the procedures to be applied in the case of leaks from the main vessel by supplementary measures. These measures are contained in procedure U4 "Steps taken to limit the consequences of a hypothetical leak from the safety tank as a result of a leak from the reactor main vessel".

After the drum sodium drainage and the first investigations, identical faults to those observed on the plate at the origin of the leak were found on similar plates. The reuse of the initial sodium drum after repair therefore proved to be impossible and it was necessary to define a replacement.

When the drum was removed, it was found out that long (several meters) cracks had also formed in the constituting weld beads of the main vessel.

Lastly, reflection after the incident underlined the need to proceed to re-examine the design and manufacturing file for safety-related components in contact with sodium to confirm the absence of zones which could present the risks of leaks.

6 - ADDITIONAL SAFETY ANALYSES CARRIED OUT AS A RESULT OF THE DRUM INCIDENT

The safety analysis carried out as a result of the drum incident covered the causes of the incident, its sequence and its consequences. Based on this analysis, additional technical measures were considered necessary before restarting the reactor. These main measures are as follows:

- setting up the procedures for interventions needed in the event of a reactor main vessel leak,
- re-examination of the radiographic images taken during the manufacture of safety-related components (reactor vessels, etc.),
- in situ confirmation using the MIR (reactor inspection machine) robot of the good condition of the reactor vessel and execution of the first periodic inspections of this vessel.

6.1 - PROCEDURE IN THE EVENT OF MAIN REACTOR VESSEL LEAK FOLLOWING A SAFETY VESSEL LEAK (PROCEDURE U4)

The reactor leak is an event considered extremely improbable and the reactor main vessel is surrounded by a second vessel, so-called safety vessel, to collect any leaks.

However, although these vessels, which are constructed in stainless steel, are different from those of the initial fuel storage drum executed in carbon steel grade 15 D3, it was considered necessary to draw all the conclusions from the drum incident and the measures that had to be taken on the spot. Indeed, when a leak on the first vessel occurs, it is essential to take a certain number of additional steps to measure and control the leak, preserve the integrity of the second vessel to the maximum and arrange around these two vessels the means for monitoring the tightness of the second vessel and the steps to limit the consequences of a leak from it.

So-called procedure U4 was established with these aims in mind. This enables very different accident scenarios associated with a main vessel leak to be dealt with and in this respect breaks down into actions spread over a period of time.

Thus, it is foreseen that in the event of a reactor main vessel leak:

- the reactor is immediately shut down and cooling begins,
- steps are taken to overcome the risks of a secondary vessel leak by pumping the sodium which flows into the inter-vessel space back to the main vessel: this pumping circuit is called the leak recovery circuit and

it can be installed in around 10 or so days, this time being due to the primary sodium reactivity; the parts of this circuit are stored,

- sodium containment and cooling of the core in the event that the safety vessel itself were to leak is guaranteed. To this end, the reactor pit is made leak-tight at its penetrations to allow argon blanketing by an injection sleeve installed permanently; the bottom of the reactor pit is lined with a layer of alumina (Cristalba) with suitable granulometry which, on the one hand, absorbs any sodium leak and acts as a blanket regarding inflammation risks, and on the other completes the leaktightness of the pit bottom. The level of sodium in the main reactor is maintained above the heating part of the assemblies to ensure their cooling, if necessary by bringing in outside sodium from the SNA tanks or secondary loop (BCS). The residual heat is evacuated inside the vessel by the BPR cooling systems via the intermediate exchanger or by the RUR, and to the outside by the RUS circuits, the water of which is replaced by an organic liquid which does not react with the sodium in the event of a leak. The RRI water circuit for cooling the reactor pit concrete is drained.

These actions are spread over a time and chronologically can be separated into:

- reactor shutdown which takes place immediately,

- first phase reactor cooling to 180°C which takes place with the BPR sodium-air exchangers available on the secondary loops in less than one week,

- the installation of the leak recovery circuit which requires a period of ten days associated with the sodium 24 decay,

- measures for completing leaktightness of the reactor pit in the lower part : the neutron detectors are withdrawn and the accesses are plugged. The bottom of the reactor pit, after opening the access door, is lined with alumina (Cristalba). This access first requires cooling of the sodium and decay of sodium 24, and can only take place after 10 or so days. Installing the Cristalba takes several months. The reactor pit access door is then welded up and the reactor pit is inerted (argon). Injection of the argon which normally takes place in the lower part of the reactor pit can also be injected in the upper part,

- during the entire U4 procedure, steps for cleaning the heating part of the core are taken so as to evacuate the residual heat: addition of sodium from tank SNA or a second loop are possible.

Ultimately, core assemblies are discharged. The discharged sub-assemblies are washed and installed in the APEC pool. The reactor sodium can then be drained.

6.2 - RE-EXAMINATION OF THE FABRICATION FILES

The design and fabrication files were re-examined in the light of the incident encountered on the storage drum. This long task covered in priority the main reactor vessel and its safety vessel.

The new analysis of the radiographic images of the two vessels taken during manufacture highlighted:

- signs already noted during the first examination and wrongly interpreted as satisfying acceptance criteria,

- new indications which had not been noted during the initial inspection.

The eventual noxiousness of these indications were then studied. This consisted in adopting penalising hypotheses for the dimensions of indications and their position. As a first step in the calculation, the propagation of the supposed fault was estimated by taking into account mechanical loads which may arise on it during plant operation. This calculation estimates in particular the consequences of a certain number of shutdowns or events leading to mechanical stresses on the structures studied. Thus, we can determine the maximum and pessimistic extension in length and depth of the defect. A second calculation is then made to assess the effect on this defect of an increasing mechanical load which, depending on the case, might be an earthquake or other accident situation. All these calculations take into account the penalising hypotheses and have shown that the anomalies encountered are not noxious.

6.3 - INSPECTION OF THE MAIN REACTOR VESSEL WELDS

The inspection, which was made on 27 June 1988 to 23 August 1988, was carried out using the MIR robot (inspection module for fast reactors).

This module consists of a carriage which bears on each one of the two vessels through its four wheels.

A television camera reads the marks etched on the safety vessel and guides the robot.

The welds are inspected using focalised ultrasonic transducers, which employ the echographic technique, mounted on the robot. A coupling fluid is installed between the transmitter and the weld to be checked using a small vessel containing this fluid. This vessel is in contact with the wall of the zone to be inspected through a seal. A visual inspection of the weld is also carried out by a CCTV camera.

The inspection programme took into account the indications noted during the examination of the radiographic images at the end of the main vessel fabrication.

The inspection carried out did not show any unacceptable faults on the welds corresponding to the size of the faults taken into account in the " noxiousness " studies as a result of rechecking the radiographic images, nor the " noxious " evolution of the triple point weld condition. The triple point designates the horizontal circular weld corresponding to the position where the weight of the core and the reactor internals are mechanically supported.

As with the water reactors, new inspections will be carried out at regular intervals using the MIR robot, the inspection system of which proved to be accurate and effective during this first inspection operation.

FALL OF THE POLAR CRANE DERRICK

No.56 Date : 2 October 1989 at 17.00

1 - NATURE OF THE INCIDENT

The incident concerns an item of plant temporarily fitted on the reactor building polar crane carriage with a view to dismantling the access gangway to the crane arch. The gangway is a structure fixed on one of the polar crane beams which it was decided to replace by a simple hooped access ladder in 1988.

The incident occurred during the statutory testing of this derrick after erection. The reactor was shut down and the dome was entirely closed. The test load on platform R 805 (no-risk zone) had just been raised 15 cm. The upper orientation bearing on the derrick boom broke causing the latter and its winches to fall.

The consequences on the installations were very limited: striking the dome and the chilled water system DEG without loss of leaktightness, deformation of the cable tray without break or electricity fault.

2 - ANALYSIS

The incident is due to the failure of 8 fixing screws on the upper bearing of the boom mast at the gantry upright. The range of boom orientation adopted on design was 90° between the longitudinal and transversal axes of the crane. In the sequences of gantry dismantling operations, the maximum bearing on the derrick could reach 10 m in the crane longitudinal axis (removal of items in R 805) but should not exceed 4 m in the transversal axis (elements on the crane). At the planning stage, the orientation bearing had therefore been placed in the most penalising position, i.e. on the longitudinal axis. During execution, the bearing rotational axis was installed in the transversal axis. On testing the load in the most penalising conditions with the boom in the longitudinal position, the bearing fixing screws were subject to shear stresses which led to their successive failures. It is to be noted that the statutory test in the fabrication workshop was carried out in the direction of greatest strength due to the lack of space.

3 - STEPS TAKEN

Inspections showed:

- dome : absence of fault according to GDL (EDF group of laboratories) report,
- small west cupola : impacted zone undamaged (inspected by dye penetrant and radiographic examination) and longitudinal weld situated close to the impact undamaged (radiographic examination),
- small cupola gangway : structural steel work non-deformed,
- dome gangway : welding of the support on the dome undamaged (dye penetrant examination),
- chilled water piping : replacement of a 7 m section and inspection,
- cable trays : repair, replacement of cable.

The eventual " noxiousness " of the faults was analysed :

- west cupola : demonstration of the buckling performances under dimensioning loading (-0.2 bar) of the deformed part.

Analysis of the incident having demonstrated that the fundamental cause was a failure of the quality assurance measures set up, an inquiry was initiated (see mail SX90-0174 of 19 January 1990).
Henceforth works implying load transport in the reactor hall, risk studies were required and protective measures taken if needed to avert damages.

INCREASE OF THE PRIMARY SODIUM IMPURITIES LEADING TO AN OVERSHOOT ON THE RANGE AUTHORISED BY THE OPERATING SPECIFICATIONS

I.S. N°61 Date : 20 June 1990

1 - GENERAL INCIDENT CHRONOLOGY

After unit shutdown to permute the diluents (Sept. 7 1989 to April 13 1990) and shutdown for works as a result of detection of a leak on the feedwater purification circuit (April 28 1990 to May 31 1990), the reactor was moved into critical mode and reached its nominal power on June 11 1990.

During all this period up to the temperature build-up (June 10 : 20% nominal power), sodium cleanliness was monitored by the operating teams on recordings delivered by the plugging indicators using the usual operation methods : plugging temperature Tb[(1)], general slope of the curve. Monitoring confirmed that there was a good level of cleanliness; in particular, the 110°C [(2)] level lasted around six hours.

From the temperature build-up, the operator observed a plugging temperature rise and therefore an increase in the impurities.

After analysing the recordings, confirmed by the constructor and the sodium chemistry experts, the plant concluded that the reactor was still operating in the range limits authorised by the operating specifications (120°C<Tb<150°C) and that the return to normal conditions could be envisaged in the delay of one month authorised by the specifications. Indeed, the increase in the plugging temperature after shutdown for works with the reactor cover open is a phenomenon which has already been observed. It corresponds to the phase of re-dissolving with the temperature of the impurities formed during the shutdown (the works having allowed air to ingress with associated oxygen and humidity).

It was then observed that the integrated purification filtration cartridges were saturated.

In addition, plugging temperature measurements did not follow the temperature decrease expected by the experts, the operator therefore decided to stop the reactor on July 3.

After reactor fast shutdown, it was maintained isothermal at a temperature of 250 °C, i.e. at least 40 °C above the oxygen saturation temperature in the primary sodium when the rate of impurities was maximum (15 ppm oxygen).

On July 9 1990, an analysis of the cover gas revealed a nitrogen content of 17% and brought into question the previous interpretation. As a result, the plugging measurements led to the conclusion that the reactor had operated outside the operating range before being shutdown.

The application of a programme of investigations to identify the air ingress into the argon of the cover gas resulted in the origin of the pollution being discovered on July 23. This was a circulation pump on an activity measurement channel whose diaphragms were defective. 540 l/h of air entered the reactor argon circuit downstream of this circulation pump.

2 - INCIDENT DIAGNOSIS, POLLUTION LEVEL, PURIFICATION

On completion of the analysis of the plugging indicator recordings over a long period and the analysis of the samples made at different points of the primary argon circuit, the conclusion was reached that around 120 kg of oxygen entered the reactor system.

Moreover, it results from this last analysis that the reactor operated outside the operation specification range for three days before shutdown.

(1) The plugging temperature is obtained on the cooling gradient of the plugging indicator pellet. This gradient is initiated when the flow in the pellet is stable at its higher value. The plugging temperature matches the beginning of flow decrease.

(2) The low level is initiated when the pellet temperature reached 110°C in a cooling cycle and ends when the flow in the pellet reaches 55% of the initial flow. The low level time is therefore an indicator of the clogging speed on the pellet in a measurement cycle and therefore the sodium purity.

2.1 - ELIMINATION OF THE REACTOR COVER NITROGEN

The nitrogen content of the primary argon circuit (17% on maximum pollution) fell as soon as the argon circuit blowers were stopped on July 13 to reach 6% when the air ingress was eliminated on July 23.

This rate then stabilised at 4% during August. The argon was maintained static.

During September, a series of inflation - deflation operations on the reactor cover brought the rate to 0.4%.

Thereafter, the evacuation of the residual nitrogen brought it down to <0.01% at the end of August 1991. This value is below the maximum value permitted by the operating specifications updated after the incident (0.3%).

2.2 - SODIUM PURIFICATION

To reach the level of cleanliness required in all the reactor normal operating conditions (primary sodium oxide rate <3 ppm, operation with an oxygen rate between 3 and 5 ppm during one month at the end of a handling period), a purification programme was implemented.

This programme was run with a view to constantly controlling the changing parameters and maintaining these in a more favourable range than on the shutdown of July 3 1990. Therefore, sudden re-dissolving or carrying along of the sodium oxides whose presence was presumed to be at the surface of the sodium had to be avoided.

It also enabled us to acquire more thorough knowledge of pollution phenomena and behaviour of plugging indicators.

The programme therefore proceeded in several campaigns, each one of them being run based on the results obtained during the previous campaign :

1stcampaign : July - October 1990. Purification at 250°C.

This campaign began by changing the cartridges on the two clogged purification units during the shutdown of July 3 and ended by installing new cartridges. It enabled 40 kg of oxygen to be trapped in this set of cartridges.

2nd campaign : October 1990 - April 1991.

This consisted of the following:

- initial condition : reactor at 250°C, reactor pumps at 75 rpm,
- reactor sodium temperature increase to 350°C,
- reactor coolant pump speed increase to 250 rpm,
- temperature of the reactor increased to 400°C. Disturbances on the plugging measurement led to progressively lowering this temperature until 300°C. As soon as the 375°C stable level is reached, these disturbances are substantially mitigated,
- 300°C level,
- once again, the reactor sodium temperature increased to 400°C. The values measured by the plugging indicator authorised a fall to 180°C without any risk of clogging,
- final condition : reactor block at 180°C, reactor coolant pumps at 75 rpm.

On completion of this stage, the oxygen rate in the reactor sodium was brought to a value close to the 1 ppm considered satisfactory.

3rd campaign : June-September 1991

This stage was preceded by washing of the separation column situated on the reactor argon circuit in order to dissolve any sodium oxides in deposits trapped at the reactor outlet of the argon circuit.

The campaign was undertaken in the following stages:

- Initial conditions, reactor block at 180°C, primary pumps at 75 rpm,
- Reactor sodium temperature increased to 395°C. Interference on the plugging indicators appeared similar to those encountered during the second stage,
- Reactor pump speed increased to 433 rpm,
- Reactor temperature increased to 420°C. No evolution of the plugging parameters confirmed that there is no oxygen pollution input due to this isothermal operation at high temperature in nominal reactor pump rotation conditions,
- Return to 395 °C, reactor coolant pumps operating at 110 rpm. Sampling of reactor sodium for analysis. The purification units indicated signs of clogging at the sodium inlet/outlet moderating heat exchanger. This clogging, which translates into a fall in the thermal exchange coefficient, is attributed to the presence of impurities other than sodium oxide. These impurities, which are in solution in the reactor sodium during isothermal operation at a temperature above 375°C, are deposited on the cold walls of the exchanger tubes,
- Return to 180°C, reactor pumps at 110 rpm.

During all the reactor sodium purification phase, thermal hydraulic monitoring of the reactor took place in order to ensure that there is no clogging phenomenon by carrying along any surface creams or impurities in narrower sections of the sub-assemblies.

2.3 - ALTERNATIVE PURIFICATION STRATEGY BY THE EXCEPTIONAL PURIFICATION CIRCUIT

As a protective measure, the auxiliary reactor sodium purification circuit of the storage drum was re-filled in order to maintain the exceptional reactor sodium purification available through the two circuit cold traps.

Since integrated purification had proved to be sufficient to eliminate the reactor sodium pollution, exceptional purification was not undertaken.

3 - EVALUATION OF THE DIRECT OR POTENTIAL CONSEQUENCES AND ASSOCIATED ACTIONS FOR REINSTATING THE INSTALLATION

In parallel with purification, reinstatement works, non-noxiousness studies and investigations were performed.

3.1 - HYDRAULIC RELIEF VALVE PROTECTING AGAINST PRESSURISATION - DE-PRESSURISATION IN THE PRIMARY ARGON CIRCUIT (RAA0 01 ZH)

This saw its calibration liquid (NaK : sodium-potassium mixture) oxidised by the air which entered the circuit and this prohibited maintaining the argon in circulation. This relief valve has been replaced.

3.2 - INSPECTION PROGRAMME - RE-QUALIFICATION OF THE REACTOR COVER AND THE PRIMARY ARGON CIRCUIT ON WHICH THE OXIDE SPRAY COULD HAVE DEPOSITED

These inspections concluded in the absence of deposits prejudicial to the long term operation of the equipment which were in contact with the sodium aerosols during the pollution period.

3.3 - STUDY OF THE CONSÉQUENCES ASSOCIATED WITH CORROSION AND NITRIDING

A study demonstrated that the effects of the corrosion are minor and limited to the equivalent of 70 days' operation at nominal power in normal conditions of sodium cleanliness. The effects associated with nitriding are non-significant.

The conclusions of this study were validated by a formal notice issued by the CEA/EDF Materials Working Group.

4 - FUNDAMENTAL CAUSES - INFORMATION GAINED

The fundamental causes of the incident were identified as:

- incomplete analysis during primary argon circuit design of the risks of air ingress and more generally non-leaktightness, and their prevention ; this occurring notably at the activity measurements circulation pumps,
- absence of primary argon on-line monitoring devices in spite of such a system having been installed in the Phenix, PFR, KNK and SNR 300 plants,
- incomplete referencing of equipment and preventive maintenance programmes,
- incorrect interpretation of the plugging indicator recordings,
- inappropriate operating specifications both in terms of their clarity and their applicability.

As a result, further actions with a view to drawing all the information from the incident resulting in measures implemented before plant start-up were undertaken before the unit was restarted .

4.1 - INSTALLATION OF MONITORING EQUIPMENT (CHROMATOGRAPH) ON THE PRIMARY ARGON CIRCUIT

4.2 - DESIGN IMPROVEMENT OF THE SYSTEM FOR SAMPLING THE GAS ON THE ACTIVITY MEASUREMENT CHANNELS

A system for detecting diaphragm failure has been installed on the circulation pumps.

4.3 - REVIEW OF THE PHENOMENA ASSOCIATED WITH POLLUTION

A series of studies and research was carried out to determine the nature, mechanical and thermodynamic behaviour in reactor sodium of the impurities, their noxiousness in terms of the risks of clogging and corrosion of steels, their influence on the plugging curves and the effectiveness of the trapping function with regard to each of the impurities.

4.4 - REDRAFTING OF THE REACTOR CLEANLINESS MONITORING PROCEDURES

New criteria for monitoring the purity of the primary sodium have been defined since the parameters used until now to detect any changes (loop temperature Tb) were considered insufficient.

Henceforth, the operator will base its operations on the unplugging temperature measurement [1] associated with the sodium oxide (low unplugging temperature) and on the duration of the low level which is an effective and simple method of qualitively monitoring the sodium purity and its evolution over time.

The unplugging temperature is close to the saturation temperature for the impurity present in the sodium. Nevertheless, it confers a boundary which is higher than the saturation temperature based on thermodynamic data and the cinetics of the dissolution in the pellet.

4.5 - PROJECT REVIEW OF THE PRIMARY ARGON SYSTEM

The principle of this project review was to identify the causes of the anomalies and the dysfunctions which could affect the " intermediate containment barrier " and " the primary sodium inert gas cover " functions of the primary argon systems. They led to additional studies, corrective actions, counter-measures and modifications being defined and progressively implemented.

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[1] The unclogging temperature is obtained on the heating gradient of the clogging indicator pellet. This gradient is reached when the flow in the pellet reaches 55% of the initial flow and terminates when the high temperature level has been reached.

5 - OVERALL ACTIONS

In order to make an analysis which goes beyond the specific measures directly associated with the incident, deeper reflection has been undertaken in the following aspects :

5.1 - REVISION OF THE GENERAL OPERATING RULES

Chapters 3 and 9 of the GOR have been read in detail in order to check that each specification is clear and precise and that it results in an operating instruction which is consistent with it and that the physical means to check compliance are appropriate.

5.2 - RE-EXAMINATION OF THE PREVENTIVE MAINTENANCE PROGRAMMES

The plant has undertaken work on two main lines to ensure exhaustiveness of the preventive maintenance operations undertaken on safety-related equipment :

1st line : full listing through a campaign to identify all the equipment.

2nd line : by a team of equipment expert engineers, preparation of a zero point list of the preventive maintenance procedure and then a drafting schedule.

5.3 - MAINTAINING COMPETENCES AND EXPERT CAPACITIES AVAILABLE TO THE PLANT

Actions have been undertaken in order to ensure permanent management of the competences which have to surround the plant based on a 3-stage approach.

a) list of competences available in all the organisations involved (Novatome-Nira, CEA, EDF) ;

b) preparation of " management contracts " formalising the undertakings of each organisation ;

c) regular preparation of a statement of available competences.

5.4 - EXPERIENCE FEEDBACK, ANALYSIS OF THE PAST, 2ND LEVEL ANALYSIS

An organisation has been set out with means for guaranteeing satisfactory processing of experience feedback from the plant, other French and Overseas fast neutron reactors and the PWR plants.

Group for analysing potential operating problems

Its aim is, through the reading of tests, transfers, modifications and incident documents, to check that all the information has been drawn from the plant start-up period.

Group R

The role of Group R is to organise experience feedback from French and foreign fast neutron reactors together with PWR plants.

6 - CONCLUSION

The primary sodium pollution incident, which is particularly significant of the prototype nature of the fast neutron reactor at Creys-Malville, led to a long shutdown which justified its classification as a level 2 seriousness incident, although it did not bring into question the safety of the installations.

Shutdown for sodium purification was used to make a thorough analysis of the direct and indirect causes of the events and to take corresponding measures : re-examination of the plant maintenance and operating procedures, potential review of expertise available and needed around the operator.

COLLAPSE OF THE TRAIN A TURBINE HALL ROOF UNDER THE WEIGHT OF SNOW

I.S.No.63 Date: 13 December 1990 at 11.00

1- NATURE OF THE INCIDENT

The weight of accumulated snow caused train A turbine hall to collapse, leading to a loss of voltage on this train.

Since the 225 kV line was off-line at the time of the incident, train A switchboards were without voltage.

The loss of voltage led to start-up of the train A standby diesel generators (LHPE and LHPF).

Diesel LHPE did not automatically couple and required local intervention.

2- ANALYSIS

The incident was caused by considerable snowfall which led to the collapse of the train A turbine hall roof. Part of the cladding fell on the high-voltage equipment of train A causing a zero-phase sequence on the main transformer and therefore opening of the line circuit breaker.

The 225 kV was off-line and the loss of voltage led to the two diesels LHPE and LHPF starting without coupling diesel LHPE due to incorrect operation of the voltage regulator (fuse break contacts) and despite changeover to the standby regulator.

Manual coupling of diesel LHPE led to re-powering of the reactor coolant pump E although this was in reverse rotation as a result of failure of a rotation sensor on this pump.

The standby unloading cask (MHU) that had been stored in the turbine hall was damaged.

The reactor building and steam generator access hatch was blocked in a closed position resulting in inoperability of local mode devices.

3- STEPS TAKEN

- Reconnection of the auxiliary 225 kV line and repair of LHPE diesel generator.
- Installation of a 6.6 kV jumper cable between the non-emergency supplied 6.6 kV switchboards of train A and train B.
- Addition of special steps for clearing snow in the " cold weather " instruction.
- Repair of the turbine hall.
- Dimensional analysis of the roofs of the other buildings.
- Repair of the MHU unloading cask.
- Checks of the condition of reactor pump E after re-powering when in reverse rotation
- Decision to apply a loss of electrical source instruction throughout without seeking to use a recovered meantime source.

SIMULTANEOUS OPENING OF SEVERAL CONTAINMENT BARRIERS

I.S. No.88 Date : 3 May 1995

NATURE OF THE INCIDENT

On a servicing operation on reactor coolant system argon blowers (RAA0), early dismantling of a terminal box on blower 02 CO led to opening of the intermediate containment barrier (2nd barrier). The third barrier is by-passed by this circuit.

In parallel, the truck area (4th barrier) was open, leading to the loss of the feedwater containment.

This situation did not meet the operating specifications in shutdown status which require availability of at least one of the containment barriers.

ANALYSIS

The origin of the event is found in the failures which occurred in organisation of multi-competence maintenance and due to human elements.

Analysis of the event highlighted faults in preparing, scheduling, preparing withdrawal from operation and execution of the work.

The specificities of the equipment (complexity of the 2nd barrier and difficulty to identify certain of the materials which constitute this barrier) constituted an aggravating factor.

STEPS TAKEN

- Henceforth, technicians will indicate " Safety, caution : containment barrier " in operation documents and requests for maintenance regimes when carrying out work affecting the containment barrier.

- The requirements of the quality organisation and safety culture have been recalled through the presentation of a nuclear safety memo adapted to the LMFRs, notably communicated to electricity and mechanical engineering staff. A working group took into account the information gained from this incident when drafting the risks analysis guide adapted to Creys-Malville.

- Lastly, when repairs require several specialist contractors, co-ordination will be henceforth the responsibility of a single person.

FUSION OF THE LARGE ROTATING PLUG LIQUEFIABLE METAL SEAL

REACTOR CRITICAL

I.S. No.92 Date : 12 May 1996

NATURE OF THE INCIDENT

During neutronic tests at nominal power < 3 %, the metal to metal supports of the rotating plugs were heated 48 hours before reactor shutdown. This action, which is compatible with the current tests, also led to unintentional heating of the liquefiable metal seal (JML), the liquid status of which is incompatible with reactor criticality.

ANALYSIS

This incident caused an error in an operation sheet of procedure G 8 (Operation to Handling changeover) on a newer version published in 1991. It was not detected by the quality assurance system used to draft the instructions.

The line of defence concerning inadvertent re-supply to the JML power cabinets (administrative lockout G21) did not avoid the event as a result of the authorisation to remove padlocking issued by the shift engineer.

STEPS TAKEN

These were of two types:

- Operating instructions concerning the liquefiable metal seal :

Clarification of the temperature monitoring methods for the liquefiable metal seal indicated in instructions G1 and G1-bis (conditions at start and end of monitoring, temperature at which the liquid metal seal are considered melted).

- Criticality instructions:

An inspection of the liquefiable metal seal condition added to the checks before criticality.

ARGON LEAK FROM THE BELL OF EXCHANGER RCPE-02-EX

NATURE OF THE EVENT

From the outset, the argon supply of intermediate exchangers has justified careful monitoring by the operator. A few re-inflations per month were observed on a few intermediate exchangers including notably IHX RCPE 02 EX.

On start-up in 1994, following the long BCS shutdown, a significant increase in the frequency of bell re-inflation on IHX-RCPE 02 EX was observed (up to around one re-inflation per day). As a result of the fast shutdown of 17/12/94, the leak rate from the bell increased significantly to reach a value close to the threshold set by the operator (160 mbar/day). This justified shutdown of the reactor.

ANALYSIS

The investigations were carried out during reactor operation and then on hot shutdown and cold shutdown. The leak was located and characterised. In view of this characterisation, the absence of a generic aspect was checked for the other intermediate exchangers.

STEPS TAKEN

Safety analysis

The analysis of the safety of the event led to the following conclusions :

- leak evolution could be controlled with great reliability using operator instrumentation (pressure monitoring) and by monitoring the re-inflation frequency.

- the new inflation procedure was the subject of a safety analysis which highlighted a sufficient number of lines of defence in terms of the identified risks.

- in the event of an increase in the bell leak rate, a reactor shutdown criterion is applied. This criterion, which fixes the maximum frequency of bell re-inflation at eight per day, enabled the acceptable primary sodium gassing ($8.8.10^{-4}$ Nm^3 gas /m^3 primary sodium on average) to be complied with, including large margins. This latter rate integrates the results of the tests associated with the risk of gas accumulation in the core support diagrid and transferthrough the core.

- aggravation of the bell argon leak did not have unacceptable consequences for reactor safety. The studies undertaken as part of the analysis of the transfer of gas into the reactor remain within the umbrella case limits. In particular, the insertion of reactivity induced by sudden de-pressurising of the bell as a result of the pipe break at the level of the bell and the passage of gas into the reactor remain acceptable (less than 1$).

- as one of the lines of defence, in the event of fast build up of an intermediate exchanger leak rate, the fast shutdown alarm on bell deflation (the reliability of which has been improved) enables the reactor to be quickly brought to a sub-critical condition.

Repairs

In accordance to a repair safety file, the leak was plugged by installing a metal sleeve applied and held in position by permanent deformation (expansion by passing hydraulic pressure) using two crimpings on either side of the leak on the inner side of the pipe (pipe co-expanded with the sleeve). The equipment used to install the sleeve were then left in the piping, and is subject to specific periodic monitoring. This technique enabled the original use of this piping to be restored. This return to conformity (leak plugged) has proved to be of an acceptable quality both at power (only one inflation weekly has become necessary) and on shutdown.

Annex II

EVENTS WITH BEARING ON EFR

28.09.85	7	reactor trip	hydraulic coupler and primary sodium flowrate measurement	design of the primary flowrate measurement; procedure (rod insertion before changeover to pony motor)
08.10.85	8	emergency shutdown by clad break detection	core measurement conveyance	instrumentation and control (very weak currents)
22.10.85	11	emergency shutdown by clad break detection	core measurement conveyance	instrumentation and control (very weak currents)
22.11.85	12	loss of power induced reactor trip followed by coast-down of primary pumps without inertia	primary pumps inertia device	relaying design
07.01.86	18	abnormal heating of subassembly	plug left in sub-assembly leg	quality control
06.04.86	24	reactor trip due to fertile sub-assembly overheating	temperature fluctuation at sub-assembly outlet	thermocouples positioned closer to sub-assemblies; hot header thermalhydraulics
08.05.86	25	bursting of handling cask observation window	thermal effect on seal tightening	design
02.04.87	39	DVS rods fall-down due to trip signal by flowrate measurement when changing over one primary pump to pony motor	primary sodium flowrate measurement	design of the primary flowrate measurement
08.03.87	40	leakage of the fuel storage drum main	steel grade	steel grade; safety vessel anchored in reactor pit;

		vessel		refractory concrete
21.05.87	43	rupture of one DHR fan blades	fan (fatigue rupture)	quality control
11.01 98 12.01.89	53	anomaly in the fast isolation-decompression sequence of a steam generator	instrumentation and control	instrumentation and control
28.04.90	60	sodium leakage on the F secondary loop auxiliary system	tee-connection (thermal fatigue)	analysis of hot/cool sodium mixing conditions
20.06.90	61	rise in primary sodium impurities level leading to exceed the limits authorised by operational technical specifications	rupture of membrane in gas radiometer; cover gas chemical monitoring	addition of catharometer
12;05;9 1	65	sodium leakage on C-train diverse DHR plugging indicator	plugging indicator (pre-heating)	pre-heating design
15.09.93	79	rupture of E-primary pump coupling	gear coupling	gear design (allowing for shaft tilting), cog material (nitride steel)
19.04.94	82	discovery of through wall crack on rupture disks balancing line of C-train DHR loop	balancing line connections	elimination of connections on dead leg
03.05.95	88	simultaneous opening of the cover gas circuit and the reactor trucking gate	argon blower	maintenance-designed equipment and plant