COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

PRINCIPAL WORKING GROUP Nº 1

GENERIC STUDY ON REACTIVITY RELATED EVENTS
REPORTED THROUGH THE INCIDENT REPORTING SYSTEM (IRS)

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of the Nuclear Energy Agency
of the Organisation for Economic Cooperation and Development

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COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA), is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop and coordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international cooperation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organizations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the coordination of work in different Member Countries including the establishment of cooperative research projects and results to participating organizations. Full use is also made of traditional methods of cooperation, such as information exchanges, establishment of working groups, and organization of conferences and specialist meetings.

The greater part of the CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes cooperative mechanisms with NEA's Committee of Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regards to safety. It also cooperates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.


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Executive Summary

The Principal Working Group N° 1 of the Nuclear Energy Agency (NEA) of the OECD initiated in 1991 a generic study on reactivity related incidents. This study, coordinated by the NEA secretariat, was then conducted by a task force consisting of representatives from Belgium, France (Group Chairman), Germany and Spain.

The report covers mainly the aspects mentioned below:

- lessons learned from reactivity events reported through the IRS
- identification of operational principles that operators and/or utilities should follow.

About 110 IRS reports were reviewed for this study. Although the focus of the study was on PWR events, other reactor types were included whenever the lessons learned were deemed applicable. Some pre-IRS events were also taken into consideration.

The study addresses four categories of reactivity related events with each participating country being in charge of one of the topics cited:

- boron dilution
- rod mispositioning and premature criticality
- anticipated transients without scram (ATWS) incidents including stuck rods
- neutron flux monitoring problems including start-up neutron sources.
Results

Most boron dilution events reported to the IRS are initiated by human errors (improper action or inadequate utilisation of existing procedures) or by lack of quality control in the process involved (maintenance, normal operating performance, fuel pool cleaning, steam generator repair, etc.). The major corrective actions taken by utilities is an improvement or a development of specific procedures including sufficient checks to limit as much as possible the recurrence of uncontrolled dilution. The installation of an on-line boron meter and its operability are also actions often taken by the utilities. Additional or more frequent checks of the water level in the steam generators during filling and maintenance works are also mentioned. In case of dilution caused by fuel pool cleaning or steam generator decontamination, it should be investigated whether borated water can be used instead of pure water.

Control rod mispositioning events were caused mainly by component failures or by operator/maintainer errors including procedural errors. The main actions taken include improvements in equipment design, procedures and inspections as well as implementation of additional interlocks and personnel training. The importance of operator errors may be reduced by sufficient electrical interlocks and increasing automation. Design deficiencies do not contribute significantly to control rod mispositioning events. The strict separation between reactor control and reactor protection systems has limited the consequences of most selected events to scrams.

Two main root causes except early design deficiencies have caused the partial or total unavailability of the scram system. These are human errors during maintenance programmes particularly on all components of the scram system. The incidents selected indicate especially deficiencies in the maintenance programmes on reactor trip breakers and problems during the dismantling and reassembling of control rods and/or upper internal structures.

No incident reported to the IRS has been found where a reactivity event has occurred in coincidence with a failure or malfunction of neutron flux monitoring systems. However, several incidents have been reported in which the unavailability of such systems has initiated a reactivity transient. All reported events occurred during fuel loading and plant start-up. Procedural errors contributed to all selected events. Additionally, maintenance deficiencies and operator errors were identified as causes.
Conclusion

Rod mispositioning events and unavailability of the scram system mainly occur during power operation. Inadvertent boron dilution and neutron flux monitoring problems are typical shutdown and start-up events. Most of the incidents are caused by human errors or procedural deficiencies. This may be related to the fact that the level of automation in the relevant operating sequences is comparatively low. The risk involved may be limited if the scram function is still available and effective. Otherwise, serious incidents can result from inadvertent reactivity transients.
1. INTRODUCTION

The Principal Working Group N° 1 of the Nuclear Energy Agency (NEA) of the OECD initiated in 1991 a generic study on reactivity related incidents. This study, coordinated by the NEA Secretariat, was then conducted by a task force consisting of representatives from Belgium (Miss Hollasky and Mr. D. Degueulder - AVN), France (Mr. A. Gouffon, Group Chairman and Mr. N. Tricot - IPSN), Spain (Mr. J. Condé - CSN) and Germany (Mr. Maqua - GRS).

The study was intended to satisfy the following objectives:

a) streamline or put in proper perspective the lessons learned from reactivity events reported through the IRS;
b) identify the problems that ceased to be reported through the IRS and investigate the reasons;
c) identify operational principles that operators and/or utilities should follow; and
d) follow-up on previously identified corrective actions and determine the reason(s) for event recurrence.

About 110 IRS reports were reviewed for this study. Although the focus of the study was on PWR events, other reactor types were included whenever the lessons learned were deemed applicable. Some pre-IRS events were also taken into consideration.

The following study addresses four categories of reactivity related events with each participating country being in charge of one of the topics cited:

- Boron dilution Belgium
- Rod mispositioning and premature criticality Germany
- Anticipated Transients Without Scram (ATWS) incidents France
  including stuck rods
- Neutron flux monitoring problems including start-up neutron sources Spain

Since the primary objective of the IRS is the exchange of lessons learned among Member countries, rather than prompt an exhaustive event reporting, there was no attempt in this report to compare the results obtained by each country and to blame or credit any organisation. The number of incidents presented and their respective safety significance should therefore be considered accordingly.

In Annex 1, the lists of related events established for each category are presented. A short description of the quick refuelling method used in Belgium is given in Annex 2. Finally, protection against spurious dilution during natural circulation, which was initiated by the PSA studies in France, is described in Annex 3.
2. BORON DILUTION EVENTS

2.1. Risks Involved

Uncontrolled or unplanned boron dilution causes an increase in reactivity in the core which can lead to an uncontrolled increase in the neutron flux with potential departure from nucleate boiling (DNB) and fuel subsequent degradation.

Safety analysis reports (SARs) normally consider dilution situations for refuelling, cold shutdown, hot shutdown and power operation. These reports usually demonstrate that for the cases considered therein there are no associated DNB risks since enough time exists for operator actions to ensure a final safe and stable sub-critical condition.

However, operating experience shows that some events represent scenarios that are inconsistent most with the SAR assumptions.

Dilution events at power constitute a lesser concern due to the availability of the protection systems (i.e. control rods, emergency reactor trip signals on dF/dt, ...). On the contrary, in shutdown modes, the effects of human errors and lapses in design or maintenance quality assurance are more pronounced due to the inhibition of the reactor protection systems in such modes. It should be noted though, the reactivity margin required by the technical specifications in these latter modes is normally greater, and the time to reach criticality is thus greater than in the high power mode.

During the design of the PWRs, the fuel rod dimensions and the lattice pitch are chosen so as to achieve an under-moderated configuration, which ensures negative void and temperature coefficients. Roughly, the curve of the effective multiplication factor ($k_{ef}$) as a function of the moderator-to-fuel ratio $R$ (which is equal to the ratio of the moderator weight to the fuel weight) is shown in Figure 1 below.

![Figure 1](image-url)

Figure 1
The working point (A) of the first curve (Cb=0 ppm) ensures a negative temperature coefficient; if the temperature increases, the working point moves to the left on the curve.

If boron is added, the curve moves down and towards the left due to the poison effect of the boron (decrease of $k_{\infty}$ and displacement of the optimum to the left; indeed due to the presence of the boron, as the absorption of the thermal neutrons increases, the ratio $R$ is effectively reduced). The working point (A') of the second curve (Cb=1000 ppm) is always in the stable part of the curve while the third one (A'', Cb=2500 ppm) being in the right part of the curve corresponds to an over-moderated configuration.

From the stability point of view, the less favourable situation occurs when the boron concentration and the moderator density are maximum (i.e. in the beginning of the cycle with no Xenon, at zero power and with all rods out). Consequently, in the specific case of quick refuelling (see Annex 2 for the quick refuelling description), the absence of control rods and a higher critical boron concentration (boron concentration requirement of 2500 ppm in place of 2000 ppm in normal refuelling) lead to an over-moderated configuration which could induce a neutron burst phenomenon in case of a sudden decrease of the moderator density (i.e. bad primary system venting or vapour production). Therefore, the quick refuelling imposes additional measures to control the requirements pertaining to excessive boron coupled with the absence of control rods.

2.2. Initiating Events Leading to Boron Dilution

Introduction

From the review of all the events listed in Annex 1, four classes of boron dilution are obtained:

- boron dilution during demineralizer maintenance (Table 1),
- boron dilution through primary/secondary interface (Table 2),
- boron dilution from fuel pools (Table 3),
- other boron dilution incidents not included into the previous classes (Table 4).

These four classes, as well as a specific events which illustrate each, are described in the following paragraphs.

Putting Demineralizers into Service

The Chemical and Volume Control System contains, in the letdown line, mixed bed demineralizers which remove soluble impurities from the primary system. Purified borated water is then returned to the primary system via the charging line to maintain the primary inventory.

The resins contained in the demineralizers have to be saturated with boron prior to the demineralizer operation. Not performing this saturation results in a boron dilution of the primary system, until the boron concentrations in the demineralizer and in the primary system are equal.
During routine shutdown operations at Davis Besse (USA) on August 15, 1981, personnel notified that reactor cooling system (RCS) boron concentration had decreased from 1044 ppm to 970 ppm causing an apparent reactivity insertion of > 0.9 % Dv/k. The apparent cause was the placing in service of a purification demineralizer which was previously saturated at 432 ppm boron. Boric acid was added to bring the RCS to 1056 ppm. The shutdown procedure has since been modified to ensure that boron dilution by an unsaturated demineralizer does not occur.

The worst case could be defined as follows:

- RCS at mid-loop,
- initial boron concentration lower than required and near to the critical boron concentration,
- putting into service of an unsaturated demineralizer during the mid-loop operation.

**Dilution Through Primary/Secondary Interface**

**Boron Dilution During Maintenance Operations on Steam Generator (SG)**

At Beznau 2 on August 12, 1986, a filling of the steam generator secondary side was stopped after observation of a rapid primary coolant level rise. Inspection revealed water spilling out the manway of the SG and through 3 tubes. The spilling continued until the tubes were plugged with rubber plugs. These 3 tubes had been cut and pulled out during the preceding shutdown and, at the time of the incident, the hot side tube ends were plugged but not the cold side ends. Boron concentration dropped from 4000 ppm to about 3800 ppm while the safety limit was 1800 ppm. The root cause was a lack of co-ordination and communication between different work teams. The incident highlights the risk induced by performing various tasks simultaneously without strict quality control.

Two dilution categories can be derived from international operating experience. The first one deals with the non-plugging of a tube or the erroneous plugging of a healthy tube leaving the damaged tube as it. The second one deals with unplugged tube inadvertently cut during maintenance. In all cases, human error initiates the boron dilution resulting from the secondary SG filling.

**Boron Dilution During SG Isolation**

Here, the boron dilution is initiated by a human error or by a material failure. Actually, it results from a loss of water through nozzle blanks due to their failure, mispositioning or absence.

At Paluel 1 on February 16, 1989, a decontamination of the water boxes of the steam generators 2 and 3 was started while the nozzle blanks, blocking the inlets and outlets of these boxes from the primary system, were not in place. The demineralized water flow spraying in the boxes was greater than the maximum drainage flow. Part of the decontamination flow passed into the RCS causing boron concentration to drop to 70 ppm. In response to the alarms primary level and boron concentration, the operators stopped the decontamination, borated the RCS and returned the RCS level to its initial value.
Boron Dilution Caused by InadvertentLeaks

In this category, the boron dilution is initiated by an inadvertent tube leak or by other uncontrolled water sources which must be detected as soon as possible.

At San Onofre 1 on September 1, 1980, work was underway to remove and inspect a steam generator tube. The secondary side of the SG had been previously drained and RCS level was at mid-loop. Upon the tube removal, water streamed from the secondary side of the SG from the opening, impinging on the channel head manway edge; some of the water was observed to swirl into the channel head and drain into the RCS. The source of the water in the secondary side of the SG was determined to be leakage from the feedwater and condensate system past a block valve downstream of the feedwater regulating valve. The boron analysis indicated that a maximum dilution of 35 ppm occurred during this event. Actions taken include the installation of new inflatable seals in the steam generator nozzles and the draining of the feedwater line to the steam generators.

Dilution From Fuel Pools

From the international experience feedback, two types of dilution from fuel pools can happen. First, a leaking valve or a procedural error creates a liaison between the fuel pool and the primary system. Second, the use of a pool walls decontamination device spraying with non-borated water can induce a dilution of the primary system. Three similar events (two in France and one in USA) occurred between 1984 and 1988.

At Gravelines 1 on July 12, 1984, an unexpected boron dilution of the primary coolant from 2110 ppm to 1770 ppm occurred following refuelling when the reactor pool draining and the pool walls decontamination were performed simultaneously. Indeed, the decontamination, previously a manual operation, induced significant personnel exposure doses. To reduce it, spray ramps and a water lance each fed by demineralized water have been installed at the upper part of the pool. Due to poor co-ordination, the demineralized water spray system was left delivering a constant supply of water for about 10 1/2 hours. No attention was paid to water level in the pool and the boron meter was inoperative (due to calibration). Consequently, the alarm on low boron concentration was also inoperative due to the unavailability of the boron meter. The results of periodic samples showed the boron concentration had dropped to 1770 ppm.

This incident prompted the utility to:

- order the demineralized water supply valve to be locked out when the reactor pool is connected to the RCS;
- ensure the operability of the boron meter or the checking of the boron concentration if the boron meter is unavailable; and
- specify the conditions of the spray ramps use (for example, use demineralized water only in the final rinsing operation).

Others

At Doel 1 on November 9, 1988, the late shift operator started a dilution of the RCS from 2425 ppm to 2000 ppm. He calculated that 25000 litres of demineralized water had to be added. The dilution was started in the manual mode with a flow of about 125 litres per hour. To ensure a margin, he set the demineralized water batch counter to 21000. However, this number corresponds to a water amount of 210000 litres, as the adjusted value has to be
multiplied by a factor of 10, (the counter scale being in decalitres). Later, the alarm high neutron flux at shutdown was generated. The dilution was stopped and a boron analysis was requested. The result of the sampling showed a boron concentration of 1507 ppm. As this value differed considerably from what was expected, a new analysis was made which confirmed the first one. Finally, a boration was started.

The actions taken were:

- the demineralized water batch counter reading was changed to litres instead of decalitres,
- an on-line automatic boron analyser was installed.

2.3. Lessons Learned - Corrective Actions

Considering the four categories developed in section 2.2 above, it can be stated that most boron dilution events are initiated by human errors (improper actions or inadequate utilisation of existing procedures) or by lack of quality control in the process involved (maintenance, normal operating operations, fuel pools cleaning, SG repair, ...). The major corrective action taken by utilities is an improvement or a development of specific procedures to preclude the recurrence of similar events. These new procedures have to include enough checks to limit as much as possible the occurrence of uncontrolled dilutions. The installation of an on-line boron meter and its operability are also actions often taken by the utilities.

For the specific case of fuel pools cleaning or SG decontamination, it should be investigated whether borated water can be used instead of pure water. If so, the boron concentration of the cleaning water should be at least equal to the primary system boron concentration.

An error in the SG maintenance and repair operations (i.e. one side unplugged tube or tube unknowingly or inadvertently cut) can be easily detected by a fluorescence test. Moreover, this test allows to ensure the quality of the plug welding.

During SG filling, SG decontamination (hydrolysing) or SG isolation with nozzle dams, a periodic SG level check (every 15 minutes for example) could avoid an inadvertent dilution or, detect it quickly.

During operation involving mixed bed demineralizers, the letdown line could be directed to the liquid waste system until a chemical analysis confirms boron saturation. Administrative controls can also be imposed to better supervise the demineralizer maintenance (e.g. use of borated water in shutdown modes, avoidance of maintenance during mid-loop operation).

2.4. Conclusions

For each above mentioned class, the root cause, the initiating event and the possible corrective action(s) are summarized in the following table.

The fourth class includes the boron dilution initiated by an improper alignment of the make-up system, by a defective measurement of boron concentration or by an improper alignment of circuits during tests or maintenance operations (not covered in the above paragraphs).
<table>
<thead>
<tr>
<th>Class</th>
<th>Initiating Event</th>
<th>Root Cause</th>
<th>Corrective Actions</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.</td>
<td>demineralizer maintenance</td>
<td>- human error</td>
<td>- review of procedures</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- material failure (boron</td>
<td>- personnel training</td>
</tr>
<tr>
<td></td>
<td></td>
<td>measurement)</td>
<td>- letdown line redirection</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- administrative controls</td>
</tr>
<tr>
<td>2.1</td>
<td>SG maintenance</td>
<td>- human error</td>
<td>- review procedures</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- personnel training</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- leak test</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- periodic level check</td>
</tr>
<tr>
<td>2.2</td>
<td>SG isolation</td>
<td>- human error</td>
<td>- review procedures</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- material failure (nozzle dams)</td>
<td>- personnel training</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- periodic level check</td>
</tr>
<tr>
<td>2.3</td>
<td>inadvertent leaks</td>
<td>- material failure (tube leak,...)</td>
<td>- improve maintenance</td>
</tr>
<tr>
<td>3.</td>
<td>fuel pools</td>
<td>- human error</td>
<td>- use of borated water</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- lack of QA on design</td>
<td>- review procedures</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- personnel training</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- on-line boron meter</td>
</tr>
<tr>
<td>4.</td>
<td>other</td>
<td>- material failure</td>
<td>- review procedures</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- human error</td>
<td>- personnel training</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- on-line boron meter</td>
</tr>
</tbody>
</table>

The following actions and enhancements are recommended to avoid as much as possible an uncontrolled or unexpected boron dilution especially in shutdown modes:

- enhanced awareness of nuclear safety and specific personnel training with respect to inadvertent boron dilution,
- upgrading and/or global review of maintenance procedures,
- upgrading and/or global review of procedures for demineralizer maintenance,
- upgrading and/or global review of procedures for nozzle dams positioning,
- periodic SG level check during SG maintenance, repair, filling or isolation,
- analysis of the feasibility and the benefit of SG or fuel pools decontamination using borated water,
- analysis of the feasibility of letdown line redirection to the liquid waste system during demineralizer maintenance,
- systematic use of a fluorescence test or other leak tests after SG repair and maintenance before its filling with pure water,
- installation of an on-line boron meter and insurance of its operability; introduction of a technical specification on the operability of the on-line boron meter as a function of modes, especially in mode 6,
- investigation of the opportunity to use borated water in all operations in shutdown modes, each exception to this rule having to be defined in the Technical Specifications.

Note: It would be helpful to systematically add in the relevant IRS reports the initial and final boron concentrations as well as the corresponding critical boron concentration.
3. ROD MISPOSITIONING

3.1. Risks Involved

Reactivity transients can result in an increased neutron flux and in the worst case in prompt criticality of the reactor core. Control rod mispositioning is one of the main causes of reactivity transients.

Since the beginning of commercial use of nuclear power, reactivity incidents have been the focus of interest. Several reactivity accidents involving control rod manipulations occurred at military and research reactors, the most famous one being the accident which occurred at St. Laurent Unit 1 (SL-1) reactor accident, where the central control rod was ejected during manual manipulations.

Commercial reactors have many design features which prevent reactivity accidents. Some general design features which should avoid reactivity incidents will be mentioned here.

If a control rod is withdrawn during power operation in a PWR the subsequent increase of reactivity will depend on the rod's final position, its location within the core, and its rate of withdrawal. This position should be near complete withdrawal for shutdown control rod assemblies.

Other reactivity transients are smoother due to the limitation of control rod speed. The maximum speed of rod insertion is only used for scrams, i.e. for reactor shutdown, which should end reactivity transients. However, it takes several minutes to withdraw a control rod from its shutdown position to its fully withdrawn position. The rate of reactivity increase in this case will be quite small and reactor control and protection devices are designed to react adequately, should unforeseen circumstances occur.

In addition, there are limitations and interlocks overriding rod movement in case of the control system not being capable to cope with a reactivity transient. These systems assure e.g. fuel cladding integrity (departure from nucleate boiling within PWRs), adequate neutron flux shape, and sufficient margin for safe shutdown. If these systems should prove unable to keep the reactor within the licensed operation boundaries, the reactor protection system will shutdown the reactor automatically. This will be assured by redundant and divers activation criteria.

The reactor protection system will scram the reactor if inter alia, neutron flux reaches an upper limit which is adjusted according to the power output. The abnormal rate of increase in power will cause a scram, too. In some reactor designs even an abnormal rate of decrease in power will scram the reactor as a protective measure against rod drop incidents.

Another common design feature is the strict separation between reactor control and reactor protection systems. This influences significantly the design of the control rod drive mechanisms; which have independent equipment for control rod movement and reactor scram. The scram function is designed in accordance with the "fail safe" principle, i.e. if a loss of electricity (black out) occurs, the rods will drop.
3.2 Events Involving Rod Mispositioning

38 incidents reported to the IRS were identified as incidents involving rod mispositioning. One further incident which happened in 1974 in Germany was selected because it resulted in an unexpected criticality of the reactor.

Incidents reported to the IRS involving control rod mispositioning include no events with significant violations of safety or impact on the public. A great number of the selected incidents had only a very limited impact on plant safety, due to the actuation of the reactor protection system. However, there have been some incidents where the reactor protection system could not initiate a scram for the control rods, since they were already in the core (e.g. during shutdown).

Five classes of control rod mispositioning have been identified by scanning the IRS incidents.

3.2.1 Incidents Involving Rod Mispositioning

Erroneous control rod withdrawal will result in a power increase which could lead to criticality. A significant number of erroneous rod withdrawal incidents occurred during the shutdown or start-up modes. In most of the incidents selected, these transients were stopped by scrams. But, there are incidents during shutdown where the scram function is deactivated. One example is the withdrawal of control rods while lifting the upper reactor internals, if the control rods have not been decoupled.

One incident that occurred during shutdown involving control rod withdrawal will be described in detail. It will highlight the potential risk of control rod withdrawal and shows the significance of using diverse reactivity compensation methods.

"Withdrawal of Control Rods During Lifting of Upper Internal Structures" (IRS N°: 57.03)

The PWR was shutdown for refuelling. The handling program sequence requires that, after the reactor vessel head has been raised, the control rods must be uncoupled. Then the upper internal structures can be raised.

The uncoupling operation consists of pulling out the release button a short distance by means of a special tool. The drive rod is then placed on the head of the control rod before the next one is uncoupled. Uncoupling is checked by weight in three stages:

- weight of the control rod plus weight of drive rod,
- weight after the button is pulled to check that the coupling sleeve has indeed disconnected (weight should be that of the rod drive only),
- weight after the removal of drive rod from head and return of button (weight should be same as at the second stage).

The operator felt familiar with the uncoupling procedure and used only a summary checklist. Therefore he did not follow the detailed instructions in the procedure itself. He left out a significant check in the second stage which was not mentioned in the checklist.
This failure resulted in an incorrect uncoupling of all 48 control rods. During lifting of the upper internals, the rods were withdrawn about 3.5 m out of the core. But, they were still inserted about 30 cm. The loss of reactivity was about 6%, the boron concentration in the primary circuit was close to 2100 ppm. After the withdrawal of the control rods, boric acid was added to meet the required negative reactivity margin for cold shutdown.

3.2.2 Erroneous Non-Insertion of a Control Rod

Four control rod non-insertion incidents were reported to the IRS, three of which occurred during full power operation. The fourth one occurred during start-up at 50% full power.

Non-insertion of control rods can be described as the omission of insertion of reactivity. The potential risk includes unintentional criticality during start-up, fuel element failures, and, of course, ATWS accidents. Within this chapter ATWS accidents will not be discussed.

Two incidents were caused by defective electronic circuits which prevented control rod insertion, while in the other incidents the rod control "device" was switched on manual mode. Three of the incidents were stopped by a scram. In one event, although the reactor exceeded the authorised limits of the operating boundary its reactor protection system was not actuated since its limits were not exceeded; the reactor therefore remained critical and the necessary power decrease was not initiated.

Within BWRs there exists the potential risk of unintentional criticality during start-up when performing criticality tests. During a BWR's criticality tests the scram function must be available. These tests are not performed in PWRs. But even at PWRs, inadvertent criticality due to rod mispositioning during start-up has been reported to the IRS. An incident discussed in section 3.2.5 addresses a case whereby inadvertent criticality occurred at a PWR due to rod mispositioning during start-up.

"Inadvertent Criticality During Control Rod Tests" (BWR, 1974, before starting of IRS)

The event happened during commissioning of the plant. After changes in the control rod drive system the test procedure called for the control rods to be fully withdrawn and sequentially, the purpose being to check the annunciation of the control rods limit switches. The results of the test were to be marked on a core schematic diagram. The test was to be stopped if anything unexpected happened.

After some control rods had been moved, a fire in the construction area of the plant was announced. The shift operator performing the test was a member of the fire brigade who then left the control room at once. At this moment control rod D10 should have been inserted. The test was continued by the operator who supervised the shift operator and who was the only person left in the control room. He withdrew the next control rod D9 without having inserted control rod D10. The withdrawal of D9 was not annunciated on the control rod position indication due to an incorrect position indication pre-setting by the supervisor (H9 instead of D9). He reported the missing annunciation to the test manager. During the phone call the reactor became critical.
The criticality was terminated by a reactor scram due to high neutron flux in the intermediate range. The inadvertent criticality was caused by a human error. As a corrective measure it was determined that also during the cold shutdown state at least two operators must stay in the control room.

3.2.3 Erroneous Insertion of Control Rods

Although erroneous control rod insertions during power operation result in power transients and possibly, abnormal flux shapes, they should not, in principle, adversely affect safety (see NRC Information Notice 92-39) except where the potential exists for violating the safety limits of fuel rods.

The insertion of control rods should be safety related for decreasing reactivity. But, the inadvertent insertion of control rods during power operation could have effects on fuel rod integrity. During start-up especially during de-boration of the primary circuit in a PWR fully inserted control rods should be avoided to keep an adequate shutdown margin.

Single rod insertion is often reported to IRS. The main cause of these incidents have been failures in electronic equipment. Single rod insertion has only a low impact on nuclear safety if reactor control and protection system act properly. However, in three incidents, the cause of rod insertion is worth being mentioned in detail. Moisture affected in two of these incidents (one due to spurious actuation of the fire suppression system) the control rod drive mechanisms. These incidents were initiated by common-cause failures. In the third incident, all control rods were dropped without a scram signal due to a loss of control voltage; the resulting sub-cooling transient was stopped half a minute later by a manual turbine trip (subsequent scram signals, e.g. by low pressurizer level would have actuated the turbine trip a little bit later).

"Inadvertent Insertion of Five Control Rods" (IRS-No. 661)

While operating at full power, testing of the neutron flux monitoring system was being performed. Therefore, the reactor neutron flux controller had to be switched from automatic to manual mode of operation. When testing the reactor protection system channel A, five control rods were automatically inserted into the core.

Reactor power decreased about 10%. The operators manually withdrew the control rods to their original position, taking into consideration the local power distribution and thermal operating limits.

The cause of the incident was a wiring defect which affected all rods in a rod group which was preselected for the "selected rod insertion" function.

The incident had no effect since the reactor neutron flux controller was switched on manual mode. However, with the controller switched to automatic mode, control rods would have been withdrawn to maintain the reactor power output level. Therefore, a potential violation of the fuel safety cannot be excluded.

As a corrective action, the requirement that the reactor must be under "manual" control has been added to the testing procedures for the reactor protection system.
3.2.4 Erroneous Non- Withdrawal of Control Rods

Incidents involving non-withdrawal of control rods during start up may be of special interest. Only three incidents in this category have been reported to the IRS. One incident was caused by failure of meter relays for monitoring the control rod position. The second failure resulted from the impact of a small leak of the reactor vessel level indicator piping which caused several control rod drive coils to malfunction. This incident, like the moisture induced incidents described above, is a common-cause failure. The third incident was a coupling failure of control rod in BWR. The control rod stuck in the core and could not be moved out. This incident has the potential risk of fast increasing reactivity if the rod would drop inadvertently out of the core. Two examples, one for PWR and one for BWR, will be discussed in detail.

In a PWR, especially during start-up, control rods not withdrawn have to be avoided while diluting the primary circuit; this ensures that, once criticality is reached, an adequate means of reactivity control exists. Incident scenarios like these were not reported to the IRS, but IRS-No. 732 and 1000 may describe precursors. The investigation shows that most of the incidents involving erroneous insertion of control rods occur during power operation.

A special BWR scenario addresses a control rod stuck in the core (e.g. due to uncoupling). This rod could fall out of the core during start-up or power operation and increase reactivity significantly depending on its position in core. Coupling failures have happened in several countries; one has been reported to the IRS. In PWRs, this incident scenario has a much lower significance, because an uncoupled control rod could only drop into the core.

3.2.5 Other Rod Mispositioning Incidents

Three incidents were reported to IRS that did not fit into the first four categories. The causes of the incidents involve operator errors in control rod movement and inaccurate calculation of adjuster rod positions and critical control rod position, respectively.

One incident affected both shutdown systems due to incorrect computer-simulated calibration factors for in-core flux detectors. Within the second incident the calculation of the critical control rod position was faulty and the criticality was reached much earlier than expected. These incidents have the potential risk that not only reactor control systems but also the reactor protection system is affected. If reactor protection limits are violated e.g. due to faulty calculations, divers activation criteria must be able to scram the reactor.

The third incident involved operator errors during manual adjustment of the control rod pattern. Withdrawal and non-withdrawal of control rods occurred simultaneously. The incident resulted in a neutron flux oscillation which was terminated by scram.

Neutron flux oscillation incidents are not part of this investigation. The detailed causes of these oscillations, reported mainly from BWRs and PHWRs, are still under evaluation. ATWS incidents will be discussed in another chapter of this report.

3.2.6 Other Incidents

Another incident including withdrawal of some control rods by lifting the upper internals is reported in IRS-No. 6132 "Rod cluster control assemblies latched to upper internal during
lifting. Similar incidents involving fewer control rods which have not been reported to the IRS occurred in France, the United States and Germany. It must be mentioned that a "quick refuelling", i.e. withdrawal of all control rods while lifting the upper internals, is licensed for some reactors, e.g. Doel-4 in Belgium (see CSNI Report No. 168 "Generic Study on Incidents Involving Loss of Heat Removal Function").

In all incidents reported, the boron concentration was adequate to avoid criticality. The two required prerequisites to suffer severe consequences are the unexpected withdrawal of the control rods as described above and an insufficient boron concentration. The possible risk of this incident sequence is a prompt criticality of the cold reactor core while the primary circuit is open. It should be mentioned that a scram is impossible during this period, therefore an adequate boron concentration is the only measure to keep subcriticality. As a lesson learned, it should be a requirement for cold shutdown that the boron concentration in the primary circuit alone must ensure subcriticality if the reactor vessel is opened.

Within BWRs, the rod drop accident is of special interest. Due to the fact that a control rod stuck in the core can drop by gravity, special measures have been taken to prevent this incident sequence. Control rod drives have been designed to avoid the rod drop mechanically. If this mechanism fails, the control rod could drop out of the core and cause a significant increase of reactivity. This incident sequence was not reported to IRS, however IRS 907 "Control rod drive coupling failure", describes a stuck rod in the core, which could be a precursor of a rod drop accident.

3.3. Causes of the Incidents

Two main causes of the selected incidents could be identified. Component failures and human failures contributed each to about 50% of the incidents. Other incident causes like design and manufacturing errors affected only some incidents, but were not reported as main causes.

Component Failures

Component failures can occur in all reactor types. The potential risk of component failures depends on reactor type, mode of operation, and if a common-cause failure has happened.

In most of the reported incidents involving inadvertent insertion of control rods, the main causes were attributed to component failures related to failures in electrical equipment e.g. electronic cords and control rod drives. Nevertheless, the risk caused by these failures (IRS-No. 129, 176, 387, 581, 603, 661, 967, 6023, 6134, 6137, 6221, and 6232) was rather small.

Failures in electrical or electronic equipment dominate the other incident categories, too. However, incidents of the other categories related to component failures had some further failure modes:

- reduced voltage on the control rod drives,
- faulty control rod drive switches,
- broken lock pin in control rod drive mechanism.
The broken lock pin was the only mechanical failure (except for coupling failures) reported in the selected incidents.

**Human Failures**

Human actions caused the other 50% of the selected incidents. These failures can be divided into procedural failures, operator/maintainer errors and software failures. These failure modes often occur when human actions are required. Therefore, these failure modes often coincide with start-up or shutdown operations. Human failures are the main contributors to incidents involving withdrawal of control rods.

About half of the incidents involving withdrawn control rods occurred due to procedural failures (IRS-No. 57, 220, 571, 732, 6132). Failure modes reported were:

- operator chose wrong procedure,
- faulty calculation of xenon concentration,
- insufficient de-coupling of control rods before lifting upper internals.

Besides procedural deficiencies, operator or maintainer errors contributed to reactivity incidents, too. These included confusion of control rods, incorrect phasing of control rod drives and variable frequency inverters, respectively, and a wrong set-point of a safety valve, which caused a pressure transient.

Some incidents occurred due to failures in control devices. One of them was caused by a software failure. The failure mode is highlighted because of the increasing importance of software driven control devices.

**3.4. Lessons Learned**

Two main reasons led to the reported reactivity incidents:

- component failures, and
- operator/maintainer errors including procedural errors.

These failure modes cover the causes of nearly all the incidents selected for this evaluation. Design or construction anomalies, do not contribute significantly to control rod mispositioning incidents. This may be caused by the fact that since the beginning of commercial use of nuclear power, reactivity incidents were well known and proper design of shutdown equipment (e.g. control rods) was of high importance.

The lessons learned from the selected incidents can be described as follows:

- improvement in equipment design,
- improvement in procedures,
- improvement in inspections,
- implementation of interlocks,
- personnel training.

Actions taken are covering many aspects due to the various causes for control rod mispositioning. General recommendations how to avoid control rod mispositioning cannot be given here.
However, some points should be highlighted. The importance of operator errors can be reduced by sufficient electrical interlocks and increasing automation. To avoid additional failures, highly reliable instrumentation and control systems as well as components are necessary. Within the design and construction phases of new power plants, appropriate measures can be implemented. Backfitting of interlocks in older reactors may cause significant changes in reactor control systems and might be difficult.

3.5. Trends and Operational Principles

Rod mispositioning incidents have been reported to IRS since its inception. In the last few years, fewer mispositioning incidents were reported, but this fact may not be an indication of the rate of rod mispositioning. This illustrates the difficulties of trend analysis on IRS incident reports.

A trend analysis of German operating experience shows that the rate of rod drops in PWRs is nearly constant over the last 10 years. A similar evaluation of rod drops reported according to the German reporting criteria indicates a decreasing rate. The cause of this are improvements in reactor control systems which mitigate the consequences in most events to power reductions (rod drops without actuation of a reactor scram usually do not need to be reported according to the German reporting criteria.)

Therefore, the lower number of reports to IRS in the last years could simply be an indication that no serious new problems have arisen concerning rod mispositioning.

The strict separation between reactor control and protection systems has limited the consequences of most selected incidents to scrams. But there are incidents where a scram is not possible especially during the shutdown and start-up modes of operation. It is recommended that checks of boron concentration in PWRs or checks of neutron flux in BWRs be made as reliable as possible.

One example of a measure to avoid start-up with fully inserted control rods is an electrical interlock which prevents injection of demineralized water into the primary circuit if the control rods are fully inserted. These interlocks may be derived from the control rod insertion limits that limit the movement of control rods within allowed ranges, as a function of reactor power.

4. PARTIAL OR TOTAL UNAVAILABILITY OF THE SCRAM SYSTEM

4.1 Risks Involved

We learned from the NRX accident (1952) that a reactor should always have a fast shutdown capacity available and that this capacity should be independent of any control system. Nuclear power plants have thus been designed, since then, with control systems to maintain system parameters within pre-set limits, and with redundant protection systems to protect the plant in the event parameters exceeded the normal limits. These protection systems are designed to maintain acceptable system conditions following any anticipated transient and, in addition to automatically shutting down the reactor should be able to guarantee a minimum margin of negative reactivity. The negative reactivity margin, provided by control rods in light water reactors, represents the sub-criticality level which, at a given time, would be reached after the rod insertion (usually assuming that one is stuck out of the
core), with allowance made for the contingent addition of reactivity due to reduction in the power of the core. To calculate the negative reactivity margin, a conservative balance is established between the reactivity introduced by the power change and the negative reactivity introduced by the rod clusters. In some PWR plants, calculations done with realistic assumptions, show that even with a several stuck rods, the consequences of a cooldown accident are still limited.

As early as 1969, discussions on the impact of partial or total unavailability of the protection system or the control rods began in the US. These types of systems have been designed to be redundant and capable of performing their safety function, even with the occurrence of single failures. But the possibility of a common mode failure that could reduce the reliability of protection systems in such a way that the system might not function properly in the event of an anticipated transient, has been recognised. The study of the Anticipated Transient Without Scram (ATWS) lead most of the countries to implement modifications on nuclear power plants in order to increase the reliability of trip actuation and/or to mitigate the consequences of the transient in case of total unavailability of the scram.

Thus, for most of the plants, studies show that the consequences of either single failures on protection system, a few stuck rods or a common mode failure in case of some expected transient are still acceptable. But control rods insertion capability should be of high reliability. Any occurrence that reduces this capability should be considered safety significant, particularly occurrences involving more than one control rod and, more generally, common mode failures.

From the review of all the events listed in Annex 1, three classes of incidents related to the scram system have been defined: control rod failures, scram breaker failures and other incidents not included in the previous classes. These classes are described in the following paragraphs as well as specific events which illustrate each particular class. The list of events are presented in tables 10, 11 and 12.

4.2 Control Rod Assembly Failures

The first class of events reported in the data bank could be attributed to early design deficiencies, no obvious generic lessons can be drawn for other designs. These events are:

- Browns Ferry unit 3 (BWR) where 75 rods out of 185 failed to fully insert during a shutdown routine.
- Calder Hall - D (GCR) - one rod out of 48 failed to drop fully during pre-start-up tests.
- Fort St. Vrain (HGTR) which has experienced different reactor trips with partial insertion of some rods.

The second category deals in improper re-assembling after modification or repair. In the three incidents reported the anomalies where detected by rod drop tests during start-up of the plant.

- Point Beach (PWR - 1985): Two stuck rods due to loose parts resulting from inadequate installation procedure. Loose parts resulted from a control rod guide tube modification conducted during the refuelling outage. Several guide tubes had to be repaired.
- Blayais (PWR - 1988): During start-up tests, following an annual refuelling
outage, two control rods failed to drop fully. Two control rod guides, which had been damaged during the replacement of the guide pins, were at the origin of the incident.

Chinon (PWR - 1989): A damaged control rod was discovered, during the investigation initiated after discovering a measured rod drop time out of the allowable limits. During the refuelling outage, guide tubes with new pins were installed and one of them, which was degraded, was at the origin of the anomaly.

These latter three incidents highlight that dismantling and reassembling control rods and/or upper internal structures (for subsequent replacements of the upper internal pins for example) should never be considered innocuous, particularly if all control rods are concerned. Beyond the usual quality control procedure, cautious visual inspection should be carried out and as a third level of defence, rod drop tests must be conducted as soon as possible after the closure of the vessel head, in order to re-qualify the reactor shutdown system. Rod drop times can be measured in cold conditions or/and in hot conditions and with or without reactor coolant pumps (RCP) in service. The more accurate measurement is obtained in cold conditions with RCP out of service.

The third category of incidents is related to a lack of control and maintenance programme of the control rods.

- Susquehanna (BWR - 1984): Four control rods failed to insert and nine of them hesitated before scrambling. The origin of the problem was the Scram pilot solenoid valves (SPSV) failure due to the disk sticking to the seat. The investigation showed that:

  - SPSV manufacturer was aware of the potential problem and the disc material had already been changed on spare parts,
  - at least one year earlier, some of the control rods had experienced hesitation, and that during last rod drop tests some measured values were out of the allowed range.

- Forsmark-2 (BWR - 1985): Detachment of a friction plate, originated by intergranular stress corrosion cracking, led to a control rod stuck in the core.

- Chooz - A (PWR - 1986): During the annual shutdown, a TV camera inspection showed signs of friction at the bottom, cracks on the pins or/and broken welds on some control rods. These deteriorations could lead to potential control rod jamming.

- Generic on French PWR's: In 1986, during systematic inspection of the control rods on a 900 MWe plant, significant wear was found on the absorber rod cladding both at the upper part and the lower end of the rods. At that time, a specific testing device in development, was used to make exploratory examinations on two plants. In view of the results, it was decided that rods had to be checked on older plants, and that the development of the control rod inspection device should be accelerated in order to be able, in the future, to check all the rods. At that time, the problem was identified as serious and several interim studies and actions were proposed, but since the kinetics of the phenomenon were assumed to be slow, no urgent actions were required.
In 1988, in Dampierre-1, during a refuelling outage, one rod of a control rod assembly was discovered broken. The lower part of the rod was found in the guide tube of the fuel assembly. Subsequently, all control rod assemblies were inspected by eddy currents and ultra-sonic examination was done on selected ones (27 control rods assemblies were replaced). The requirements for the control of the rods (number of plant per year and acceptance criteria) were made more stringent and special periodic insertion tests began to be mandatory.

When unit 4 of Gravelines-1 was in cold shutdown state on 1st April, 1989, the Utility noticed that a control rod cluster was stuck. After investigation, the operating organisation concluded that the cluster was blocked mechanically in the fuel assembly or in the guide tube, having discounted the control rod drive mechanism and the electrical and electronic components. Inspection of the lower part of the cluster revealed a broken absorber pin located at the edge of the cluster. The lower part of the pin had fallen into the fuel assembly and the twisted upper part explained the blockage of the control rod assembly.

The following strategy was adopted:

1. Use of all stock available in 1989 to replace the most worn clusters in accordance with rejection criteria defined with the aim of avoiding breaks during the cycle following the inspection; the possibility of replacing the least worn clusters was examined on a case-by-case basis, taking available stock into account.

2. Development of a procurement strategy that enables the replacement of all 900 MWe PWR control rod clusters by the end of 1990 (500 RCCA were replaced).

3. Continuation of studies to determine the root cause of the problem and to define increasingly stringent rejection criteria.

To date control rods are systematically inspected every 3 years on 900 MWe French plants and the results are closely examined in order to detect any unexpected wear evolution.

4.3. Failure of Reactor Trip Breaker (RTB)

On February 22, 1983, and again on February 25, 1983, the SALEM Unit 1 reactor control rods failed to insert upon receipt of an automatic trip signal from the reactor protection system. However, the rods did insert and shut down the plant upon receipt of a manually initiated trip signal. These events were of a major safety concern since, as the two reactor trip breakers failed to open, all redundancy was lost to protect the integrity of the reactor core. The breaker failures occurred as a result of several possible contributors, the predominant cause was excessive wear accelerated by lack of lubrication and improper maintenance.

Actions taken included design changes on RTB (use of both under voltage trip attachment and shunt trip attachment in case of automatic trip), increase of the surveillance test requirements and development of a comprehensive maintenance procedure. These
incidents and the actions taken by the US NRC, lead most of the reactor vendors and utilities to review their RTB records and to increase their surveillance tests and maintenance programme, ensuring that:

1. as far as possible, the test should simulate the automatic actuation of safety systems, and

2. after any reactor trip, the sequence of event records be checked for correct actuation of the reactor protection system.

Five out of the six cases of failure of one out of two RTB reported to the IRS, have been attributed to the lack of adequate maintenance on RTB, but, except for the incident on Mac Guire which concerns the manual trip function, all incidents occurred before 1986. The last case in Sequoya unit 2 was due to human error during the manual test of the RTB, which lead to the failure of the undervoltage card. The report states that this error was a potential common mode failure mechanism for the reactor protection system.

4.4 Other Incidents

Three other incidents, concerning the non operability of the scram system, were identified. They all concern the violation of technical specifications during the start-up phase. Since this topic is covered by studies on human factors during outages, these incidents are not analysed.

4.5 Conclusions

The above study shows that, if we exclude a few early design deficiencies, two main root causes could constitute a common mode failure for the scram system: human error during operation of maintenance, repair or modification, and lack of tests or maintenance programmes particularly on control rods, trip breakers and more generally on all components of the scram system.

On the basis of information provided by the IRS data bank, it seems that the tests and the maintenance programmes on reactor trip breakers are much more efficient to-day than they were in the early eighties.

The other lessons concern the dismantling and reassembling of control rods and/or upper internal structure, which should never be considered innocuous, particularly if all control rods are involved.

If not already done, the following actions could be taken:

- review the maintenance programme on scram system components,
- review the re-qualification and periodic test programmes in order, in particular, to insure that the tests conditions simulate correctly the automatic actuation of safety systems,
- identify the operations (maintenance, tests, repair or modification) concerning all the control rods or all the RTB or, more generally, the redundant protective channels and look at different levels of defence (QC, Test...) which could reduce the probability of common mode failure.
5. NEUTRON FLUX MONITORING PROBLEMS

The task of monitoring the neutron flux in a correct way has an evident relation to reactivity incidents. A failure to accomplish this task during an incident may avoid early detection of the incorrect plant status, and may worsen the transient by not allowing for the appropriate control and protection actions to be taken as foreseen in the plant design. On the other hand, a failure of the Neutron Flux Monitoring Systems may produce a reactivity incident through inadequate detection of a reactivity change that would have been normal otherwise.

No incident reported to the IRS has been found where a reactivity event has happened in coincidence with a failure or malfunction of the Neutron Flux Monitoring Systems. However, several events are included in the data base in which the unavailability of such systems has initiated a reactivity transient.

All reported events involving Neutron Flux Monitoring Systems occurred during fuel loading and, the vast majority, during plant start-up (Table 13). These two operational states will be dealt with separately in the following sections.

5.1 Incidents During Fuel Loading

Problems reported during fuel loading refer to the instrumentation not being able to provide adequate information about the neutron flux evolution as the reactor core is loaded. No real reactivity event under these conditions has been reported, but the potential or actual lack of surveillance during fuel loading has been identified in two recent reports, one of them generic.

The first report warns against the possibility of an inadvertent criticality when loading highly-enriched fuel into the core. Although analyses are performed to confirm that the refuelling boron concentration guarantees subcriticality for the final core configuration, these analyses may not be sufficient to assure the same for all the loading sequence steps, in which fuel assemblies are loaded at intermediate positions. The potential for several high-enrichment assemblies being grouped together during loading has been found. Under these conditions, the addition of a single highly reactive assembly adds a significant amount of reactivity to a subcritical configuration, making the inverse count rate method not valid to provide adequate warning of an approach to criticality. Corrective action taken includes thorough revision of fuel loading procedures to identify intermediate positions not covered by the analysis, and either explicitly analyse or avoid them. No further reports involving this subject have been found.

The generic report concerning fuel loading surveillance informs about several conditions detected at different PWR plants that impede correct reactivity monitoring. Neutron flux is usually measured by the source range instrumentation during fuel loading. A minimum counting rate in these channels is guaranteed by loading either secondary neutron sources or irradiated fuel into the core early into the loading sequence. It has been found that some loading procedures allowed for the load of a high number of fuel assemblies in the core before the minimum counting rate needed or effective surveillance was obtained. In other cases, the necessary counting rate was obtained by loading an irradiated bundle too close to the detectors, in which case most of the counts obtained come from the irradiated fuel, rendering core surveillance ineffective. Administrative measures have been taken in order to
make sure that at least one source range channel is continuously monitoring the core of fuel assemblies loaded, with a minimum counting rate of neutrons coming from this core.

In both cases, reported correct neutron flux surveillance during fuel loading is challenged by procedural inadequacies. The same fact, among others, is also found in the incidents during start-up reported.

5.2 Incidents During Plant Start-Up

Several incidents related to the Neutron Flux Monitoring System during plant start-up can be found in the IRS. At his stage of the plant operation, large changes in the core flux are expected, and the correct behaviour of the control and protection actions driven by the neutron flux instrumentation is of a definite importance. Three cases have been found in which criticality was obtained earlier than expected, due to an incorrect reactivity balance performed by the operator. Reactor trip by high neutron flux in the intermediate range was reached in all cases, because the operator was not aware that the reactor was critical. In one case the operator did not pay enough attention to the indications provided by the source range monitors, while in the other two the operator was following the source range indications not knowing that the channels were disconnected.

There are several problems that can be analysed in these reports. Incorrect start-up procedures play an important role in the incidents. Procedural inaccuracies were identified as the reason for the wrong reactivity balance performed by the operator in two cases, hence constituting a first cause for the incident as in the fuel loading reports.

Maintenance procedures that allow partial or complete systems to be unavailable during start-up after a plant shutdown, and surveillance procedures that do not detect the wrong status of those systems are also a concern. This is particularly the case in one additional report in which not only core surveillance was disabled during the pre-critical phase of the start-up process, but the complete Reactor Protection System was inoperative due to a non-detected loss of electrical supply to the appropriate buses. It must be said that this concentration of procedure weaknesses is not privative of core monitoring systems, but affect all plant systems in general. A significant number of reports can be found in the IRS in which different systems, including the reactor trip system, were inoperative during plant start-up and the situation was not detected and corrected until later on in the process.

Another important feature common to all reports is the lack of ability of the plant operator to detect the inoperability of the systems. As already indicated, the operator was following the approach to criticality on disconnected channels in two cases, while the complete reactor protection system was unavailable in one case. Such situations can only be detected through alarms and indications in the control room, in a very high number in the latter case. These warnings were somehow disguised among the high number of alarms present during plant start-up, so that they went unnoticed to the operator. It has to be noticed that the majority of these alarms are usually of little importance and can be avoided. A lesson that can be learned from these events is that the actuated alarms in the control room should be kept to a minimum, in other to ease the prompt identification of the new alarms by the operator.

If we add up the case where the operator did not realize the reactor was nearly critical despite a correct instrumentation indication, a certain lack of attention from the operator may also be identified in these reports. It seems that a careful survey of the instrumentation
indications and alarms could have led to the identification of the wrong criticality prediction and of the instrumentation unavailability.

5.3 Recommendations and Conclusions

From these events, the following recommendations can be summarized:

1. Procedural errors are identified in all the events reported. Fuel loading and approach to criticality procedures need a thorough revision to ensure their adequacy in the aspects already described.

2. Maintenance methods and procedures should guarantee that the plant systems worked on are returned to the operational condition after a plant shutdown. The appropriate surveillance procedures should check the systems status.

3. Confidence in the procedures by the operator should not impede the continuous verification of the plant status, particularly during the start-up process. Special attention should be paid to the Neutron Flux instrumentation indications and alarms. In order to allow for an early identification of these warnings, the operator should take the appropriate actions to keep the actuated alarms originating in other systems to a minimum.
REFERENCES

1. NEA IRS DATA BANK


(*) Restricted
Annex 1

List of Incidents

Boron Dilution

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Partial or Total Unavailability of the Scram System

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Neutron Flux Monitoring Problems

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<tr>
<th>Table</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Table 13</td>
<td>Neutron flux monitoring problems</td>
</tr>
<tr>
<td>Plant Name</td>
<td>Event Date</td>
</tr>
<tr>
<td>-------------</td>
<td>------------</td>
</tr>
<tr>
<td>Crystal River 3</td>
<td>77/02/16</td>
</tr>
<tr>
<td>Oconee 2</td>
<td>80/05/07</td>
</tr>
<tr>
<td>Trojan'</td>
<td>80/06/27</td>
</tr>
<tr>
<td>David Besse</td>
<td>81/08/19</td>
</tr>
<tr>
<td>San Onofre 1</td>
<td>82/06/09</td>
</tr>
<tr>
<td>Blayais 3</td>
<td>86/10/00</td>
</tr>
<tr>
<td>Koeberg site</td>
<td>86 &amp; 87</td>
</tr>
<tr>
<td>Dampierre 2</td>
<td>91/07/24</td>
</tr>
<tr>
<td>Nogent 2</td>
<td>91/08/07</td>
</tr>
<tr>
<td>Gravelines 1</td>
<td>91/09/24</td>
</tr>
<tr>
<td>Penly 1</td>
<td>91/09/30</td>
</tr>
<tr>
<td>Thange 1</td>
<td>91/11/22</td>
</tr>
</tbody>
</table>

1 to be connected with events of class 2.1
### Table 2.1: Boron Dilution During Maintenance Operations on Steam Generators

<table>
<thead>
<tr>
<th>Plant Name</th>
<th>Event Date</th>
<th>Description</th>
<th>Corrective Actions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Surry 2</td>
<td>7/6/730</td>
<td>Leakage from 2 tubes cut during the removal of a section of the 7th tube support plate on the secondary side of the SG</td>
<td>NUREGCR-2798, NUREGCR-2798</td>
</tr>
<tr>
<td>St. Lucie 1</td>
<td>7/6/730</td>
<td>Leakage from 2 tubes cut during the removal of a section of the 7th tube support plate on the secondary side of the SG</td>
<td>NUREGCR-2798, NUREGCR-2798</td>
</tr>
<tr>
<td>Point Beach 1</td>
<td>7/11/71</td>
<td>Leakage from 2 tubes cut during the removal of a section of the 7th tube support plate on the secondary side of the SG</td>
<td>LER 82-006, NUREGCR-2798</td>
</tr>
<tr>
<td>Ginnia</td>
<td>8/2/03</td>
<td>Leakage from 2 tubes cut during the removal of a section of the 7th tube support plate on the secondary side of the SG</td>
<td>LER 82-049, NUREGCR-2798</td>
</tr>
<tr>
<td>Calvert Cliffs 2</td>
<td>8/2/03</td>
<td>Leakage from 2 tubes cut during the removal of a section of the 7th tube support plate on the secondary side of the SG</td>
<td>LER 82-049, NUREGCR-2798</td>
</tr>
<tr>
<td>Beznau 2</td>
<td>9/12/03</td>
<td>Leakage from 2 tubes cut during the removal of a section of the 7th tube support plate on the secondary side of the SG</td>
<td>LER 82-049, NUREGCR-2798</td>
</tr>
<tr>
<td>Blybhia</td>
<td>9/30/03</td>
<td>Leakage from 2 tubes cut during the removal of a section of the 7th tube support plate on the secondary side of the SG</td>
<td>LER 82-049, NUREGCR-2798</td>
</tr>
</tbody>
</table>

### Table 2.2: Boron Dilution During Steam Generator Isolation

<table>
<thead>
<tr>
<th>Plant Name</th>
<th>Event Date</th>
<th>Description</th>
<th>Corrective Actions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ginna</td>
<td>7/41/18</td>
<td>When workmen removed the mainway cover and insert from the SG hot leg pressure charged and caused water to gush out the mainway. But the ball valve of water drained into the loops, resulting in a dilution</td>
<td>NUREGCR-2798, LER 82-049, NUREGCR-2798, LER 82-049, NUREGCR-2798</td>
</tr>
<tr>
<td>Palendre</td>
<td>8/6/705</td>
<td>When workmen removed the mainway cover and insert from the SG hot leg pressure charged and caused water to gush out the mainway. But the ball valve of water drained into the loops, resulting in a dilution</td>
<td>NUREGCR-2798, LER 82-049, NUREGCR-2798, LER 82-049, NUREGCR-2798</td>
</tr>
<tr>
<td>San Cristo 1</td>
<td>8/9/216</td>
<td>When workmen removed the mainway cover and insert from the SG hot leg pressure charged and caused water to gush out the mainway. But the ball valve of water drained into the loops, resulting in a dilution</td>
<td>NUREGCR-2798, LER 82-049, NUREGCR-2798, LER 82-049, NUREGCR-2798</td>
</tr>
</tbody>
</table>

**Reference**
- NUREGCR-2798
- LER 82-006
- LER 82-049
- UR 0784.00
- IRS 1111.00
- NRC 1974
- NRC 1980
- RS 1148.00
### Table 2.3: Boron Dilution Caused by Inadvertent Leaks

<table>
<thead>
<tr>
<th>Plant Name</th>
<th>Event Date</th>
<th>Plant Status</th>
<th>Description</th>
<th>Corrective Actions</th>
<th>Reference</th>
</tr>
</thead>
</table>
| San Onofre 1 | 80/09/01   | Refuelling   | Upon tube removal, water streamed from the secondary side of SG. The source of the water was determined to be from the FW system leaking past a block valve downstream of the FW regulator valve. | • revision of the make-up process  
• new inflatable seals  
• draining of the FW lines to the SG | PRE Vol. 2 N° 6 November 1980 |
| Calvert Cliffs 1 | 82/05/14 | Refuelling   | Dilution due to SG tube leak. The tube was perforated during support rim cut. | revision of the support rim cut procedure | LER 82-023 |

### Table 3: Boron Dilution From Fuel Pools

<table>
<thead>
<tr>
<th>Plant Name</th>
<th>Event Date</th>
<th>Plant Status</th>
<th>Description</th>
<th>Corrective Actions</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>San Onofre 2</td>
<td>82/03/24</td>
<td></td>
<td>A leaking RWST valve allowed pure water to fill the piping between the RWST and the charging pumps.</td>
<td></td>
<td>LER 82-008</td>
</tr>
</tbody>
</table>
| San Onofre 2 | 82/03/14   | RHR in operation | Loss of RHR flow in both trains resulted in a dilution from the RWST (opening of the liaison to establish the pump prime). Boron concentration in RWST varied from 612 ppm at the top to about 1900 ppm at the bottom where the boron concentration is usually measured. | installation of a line to recirculate RWST content from top to bottom | IRS 0172.00  
LER 257 |
| Gravelines 1 | 84/07/11   | Refuelling   | Dilution due to the use of a device for decontaminating the pool walls with the aid of sprays fed by demineralized water | ensuring the operability of the boron meter  
redifinition of conditions of use of the spray ramps | IRS 0867.00 |
<p>| Dampierre 4 | 87/05/30   | Refuelling   | The pool spray system is alimented by demineralized water by RWST water. The pure water valves remained opened. | procedures modifications | Bulletin SN 57 |
| Robinson 2   | 88/12/20   | Refuelling   | The reactor upper internals were being sprayed with pure water in the storage area. |                                          | INPO SER 18-89 |</p>
<table>
<thead>
<tr>
<th>Plant Name</th>
<th>Event Date</th>
<th>Plant Status</th>
<th>Description</th>
<th>Corrective Actions</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>David Besse</td>
<td>79/04/23</td>
<td>RCS filling</td>
<td>Deboration during RCS filling due to a combination of mechanical, personnel and procedural errors</td>
<td>• modification of procedures • periodic sampling during the fill</td>
<td>PRE Vol. 1 N° 3 July 1979</td>
</tr>
<tr>
<td>Surry 1</td>
<td>82/04/15</td>
<td>Power</td>
<td>Failure to maintain make-up boron concentration equal to RCS concentration resulted in an over dilution</td>
<td></td>
<td>LER 82-048</td>
</tr>
<tr>
<td>Flamanville</td>
<td>85/06/06</td>
<td>Start-up</td>
<td>Failure in the boron/water feed system not identified by the boron concentration measurement</td>
<td>• conversion of steady state command to pulse command • advisability of checking the boron meter</td>
<td>IRS 0866.00</td>
</tr>
<tr>
<td>Doel 3</td>
<td>85/07/31</td>
<td>Hot shutdown</td>
<td>Unexpected dilution during RCS venting due to bad boron concentration measure</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Bugey 4</td>
<td>86/10/15</td>
<td>Refuelling</td>
<td>Dilution due to an alignment error on the reactor make-up system and to an incorrect alignment of the boron meter</td>
<td>modification of procedures</td>
<td>IRS 0733.00</td>
</tr>
<tr>
<td>Paks 1</td>
<td>88/00/00</td>
<td></td>
<td>Decrease of boron concentration in ECCS tanks due to leakage valve</td>
<td>• boron concentration requirements modified</td>
<td>IRS 6178.00</td>
</tr>
<tr>
<td>Doel 1</td>
<td>88/11/09</td>
<td>Start-up</td>
<td>A much further than expected dilution of the primary coolant occurred, following refuelling, due to an incorrect adjustment of the total amount of demineralized water to be injected into the RCS. In fact, the operator confused litre with decalitre.</td>
<td>• DW counter adapted into litres • on-line automatic boron analyser installed</td>
<td>IRS 1000.00</td>
</tr>
<tr>
<td>Surry 1</td>
<td>89/04/23</td>
<td>Cold shutdown</td>
<td>Slow dilution due to RCP standpipe primary grade water make-up valve in leakage</td>
<td>procedures revised or developed</td>
<td>LER 89-016</td>
</tr>
<tr>
<td>Surry 2</td>
<td>89/10/25</td>
<td>Cold shutdown</td>
<td>Dilution due to improper setting of the blender</td>
<td>requirement for following the procedures</td>
<td>LER 89-015</td>
</tr>
<tr>
<td>Golfech 1</td>
<td>90/01/04</td>
<td>Cold shutdown</td>
<td>Simultaneous filling of RCS and make-up of borated water to RWST resulted in a dilution because water tended to flow towards the RCV system and boron towards the PTR system due to the absence of a mixer and to positions of the water and boron taps relative to the feeders to the PTR and the RCS systems</td>
<td>procedures modified • modification of location of the boron and water connections studied</td>
<td>IRS 1128.00</td>
</tr>
<tr>
<td>Belleville 2</td>
<td>91/07/10</td>
<td>Cold shutdown</td>
<td>Inadvertent water introduction into RCS during a manouvriability test on an isolating valve of a tank. The water was coming from an incomplete draining of the SI tank following its hydrostatic test.</td>
<td>hydrostatic tests with borated water</td>
<td>IRS 1273.00</td>
</tr>
</tbody>
</table>
### Table 5: Withdrawal of Control Rods

<table>
<thead>
<tr>
<th>IRS No</th>
<th>Date</th>
<th>Country</th>
<th>Plant</th>
<th>Type</th>
<th>Status</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>57.3</td>
<td>81/07/21</td>
<td>France</td>
<td>Bugey 2</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Withdrawal of control rods during reloading</td>
</tr>
<tr>
<td>546</td>
<td>84/08/25</td>
<td>UK</td>
<td>Chapelcross 1</td>
<td>GCR</td>
<td>Start-up</td>
<td>Control rod withdrawal fault arising from faulty control rod drive switch</td>
</tr>
<tr>
<td>707</td>
<td>86/03/18</td>
<td>USA</td>
<td>Peach Bottom 3</td>
<td>BWR</td>
<td>Start-up (3% power)</td>
<td>Out of sequence control rod withdrawal due to a series of operator errors</td>
</tr>
<tr>
<td>732</td>
<td>86/05/21</td>
<td>France</td>
<td>Bugey 2</td>
<td>PWR</td>
<td>Hot shutdown</td>
<td>Reactor scram through high flux variation during divergence</td>
</tr>
<tr>
<td>885</td>
<td>87/04/26</td>
<td>Canada</td>
<td>Point Lepreau 1</td>
<td>PHWR</td>
<td>Shutdown</td>
<td>Reversal of adjuster drive phases</td>
</tr>
<tr>
<td>887</td>
<td>87/05/12</td>
<td>Canada</td>
<td>Point Lepreau 1</td>
<td>PHWR</td>
<td>Start-up (0.01% power)</td>
<td>Incorrect phasing of variable frequency inverter</td>
</tr>
<tr>
<td>909</td>
<td>86/09/14</td>
<td>UK</td>
<td>Hinkley Point B2</td>
<td>AGR</td>
<td>Full power</td>
<td>Reactor trip due to withdrawal of regulating control rods</td>
</tr>
<tr>
<td>6132</td>
<td>88/03/26</td>
<td>Korea</td>
<td>KNU 8</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Rod cluster control assemblies latched to upper internal during lifting</td>
</tr>
</tbody>
</table>

### Table 6: Non-Insertion of Control Rods

<table>
<thead>
<tr>
<th>IRS No</th>
<th>Date</th>
<th>Country</th>
<th>Plant</th>
<th>Type</th>
<th>Status</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>295</td>
<td>82/09/30</td>
<td>USA</td>
<td>Zion 1</td>
<td>PWR</td>
<td>Full power</td>
<td>Control rod drive failure and reactor trip</td>
</tr>
<tr>
<td>401.G6</td>
<td>83/12/27</td>
<td>Sweden</td>
<td>Ringhals 3</td>
<td>PWR</td>
<td>Full power</td>
<td>Power grid outage, Dec. 27, 1983</td>
</tr>
<tr>
<td>496.G3</td>
<td>84/09/26</td>
<td>USA</td>
<td>Trojan</td>
<td>PWR</td>
<td>Start-up (50% power)</td>
<td>Stuck open main stream relief valve and reactor trip</td>
</tr>
<tr>
<td>889</td>
<td>87/02/04</td>
<td>France</td>
<td>Blayais 4</td>
<td>PWR</td>
<td>Full power</td>
<td>Exceeding authorized limits concerning the operating boundary and the insertion of regulating rods</td>
</tr>
<tr>
<td>IRS N°</td>
<td>Date</td>
<td>Country</td>
<td>Plant</td>
<td>Type</td>
<td>Status</td>
<td>Title</td>
</tr>
<tr>
<td>--------</td>
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<td>------------</td>
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<td>-----------</td>
<td>-----------------------------------------------------------------------</td>
</tr>
<tr>
<td>110.2</td>
<td>81/06/24</td>
<td>Canada</td>
<td>Bruce A2</td>
<td>PHWR</td>
<td>88% power</td>
<td>Dual computer failure</td>
</tr>
<tr>
<td>129</td>
<td>81/10/02</td>
<td>Japan</td>
<td>Takahama 2</td>
<td>PWR</td>
<td>Full power</td>
<td>Circuit card failure in control circuit of the control rod drive mechanism</td>
</tr>
<tr>
<td>176,G4</td>
<td>81/11/14</td>
<td>USA</td>
<td>Ginna</td>
<td>PWR</td>
<td>Full power</td>
<td>Inadvertent actuation of fire suppression system</td>
</tr>
<tr>
<td>263</td>
<td>82/12/24</td>
<td>Japan</td>
<td>Takahama 2</td>
<td>PWR</td>
<td>Full power</td>
<td>Reactor trip due to inadvertent operation in power supply to the control rod drive mechanism (CRDM)</td>
</tr>
<tr>
<td>374,G1</td>
<td>83/03/11</td>
<td>USA</td>
<td>Quad Cities 1</td>
<td>BWR</td>
<td>Shutdown</td>
<td>Improper control rod manipulations</td>
</tr>
<tr>
<td>374,G2</td>
<td>83/07/14</td>
<td>USA</td>
<td>Hatch 2</td>
<td>BWR</td>
<td>Start-up</td>
<td>Improper control rod manipulations</td>
</tr>
<tr>
<td>387</td>
<td>83/07/16</td>
<td>Finland</td>
<td>Lovisa 2</td>
<td>PWR</td>
<td>Full power</td>
<td>Dropping of all control rods into the reactor core without a trip signal from the reactor protection system</td>
</tr>
<tr>
<td>581</td>
<td>85/03/23</td>
<td>USA</td>
<td>North Anna 2</td>
<td>PWR</td>
<td>Full power</td>
<td>Partial loss of power due to actuation of transformer single differential trip relay</td>
</tr>
<tr>
<td>603</td>
<td>85/06/01</td>
<td>Canada</td>
<td>Bruce A2</td>
<td>PHWR</td>
<td>87% power</td>
<td>Shutoff rod driven into core due to logic unit board fault</td>
</tr>
<tr>
<td>IRS No.</td>
<td>Date</td>
<td>Country</td>
<td>Plant</td>
<td>Type</td>
<td>Status</td>
<td>Title</td>
</tr>
<tr>
<td>--------</td>
<td>----------</td>
<td>-----------</td>
<td>-----------</td>
<td>------</td>
<td>------------</td>
<td>----------------------------------------------------------------------</td>
</tr>
<tr>
<td>661</td>
<td>88/04/06</td>
<td>Switzerland</td>
<td>Leibstadt</td>
<td>BWR</td>
<td>Full power</td>
<td>Inadvertent insertion of 5 control rods</td>
</tr>
<tr>
<td>961</td>
<td>88/09/09</td>
<td>UK</td>
<td>Oldbury A2</td>
<td>GCR</td>
<td>Full power</td>
<td>Trip of reactor 2 due to a bulk control rod falling into the core</td>
</tr>
<tr>
<td>967</td>
<td>88/12/06</td>
<td>Japan</td>
<td>Takahama 3</td>
<td>PWR</td>
<td>Full power</td>
<td>Reactor automatic shutdown following the fall of control rods</td>
</tr>
<tr>
<td>1105</td>
<td>89/12/26</td>
<td>USA</td>
<td>Prairie Island 2</td>
<td>PWR</td>
<td>Full power</td>
<td>Reactor trip with partial loss of off-site power</td>
</tr>
<tr>
<td>1211</td>
<td>91/05/10</td>
<td>Japan</td>
<td>Tokai</td>
<td>GCR</td>
<td>Power operation</td>
<td>Reactor manual shutdown due to trouble of control rod driving mechanism</td>
</tr>
<tr>
<td>6023</td>
<td>84/11/13</td>
<td>Korea</td>
<td>Ko-Ri 1</td>
<td>PWR</td>
<td>Power operation</td>
<td>Reactor trip on control rod system failure</td>
</tr>
<tr>
<td>6134</td>
<td>87/11/05</td>
<td>USSR</td>
<td>Balakovo 1</td>
<td>PWR</td>
<td>Full power</td>
<td>Malfunction of a control rod in the reactor control and protection system and failure of the steam generator level controller</td>
</tr>
<tr>
<td>6137</td>
<td>88/09/21</td>
<td>USSR</td>
<td>Kola 4</td>
<td>PWR</td>
<td>Full power</td>
<td>Spontaneous drop of reactor protection system control rod into the core while the reactor was operated at rated power</td>
</tr>
<tr>
<td>6176</td>
<td>87/09/23</td>
<td>India</td>
<td>Rajasthan 2</td>
<td>PHWR</td>
<td>Power operation</td>
<td>Dropping of a central ease adjuster rod during replacement at Raps-2</td>
</tr>
<tr>
<td>6221</td>
<td>89/10/14</td>
<td>USSR</td>
<td>Rovno 3</td>
<td>PWR</td>
<td>95% power</td>
<td>Shutdown due to spontaneous drop of a control rod</td>
</tr>
<tr>
<td>6232</td>
<td>89/02/02</td>
<td>USSR</td>
<td>Chernobyl 2</td>
<td>LWGR</td>
<td>Full power</td>
<td>Spontaneous insertion of five reactor control and protection system rods into the core</td>
</tr>
</tbody>
</table>
### Table 8: Non-Withdrawal of Control Rods

<table>
<thead>
<tr>
<th>IRS N°</th>
<th>Date</th>
<th>Country</th>
<th>Plant</th>
<th>Type</th>
<th>Status</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>283</td>
<td>82/10/06</td>
<td>USA</td>
<td>Zion 2</td>
<td>PWR</td>
<td>Start-up</td>
<td>Failure of control rod drive coils</td>
</tr>
<tr>
<td>907</td>
<td>88/09/24</td>
<td>Switzerland</td>
<td>Leibstadt</td>
<td>BWR</td>
<td>Full power</td>
<td>Control rod drive coupling failure</td>
</tr>
<tr>
<td>988</td>
<td>88/12/20</td>
<td>Japan</td>
<td>Tokai 1</td>
<td>GCR</td>
<td>Reducing power</td>
<td>Reactor automatic shutdown during power reducing operation</td>
</tr>
</tbody>
</table>

### Table 9: Other Incidents

<table>
<thead>
<tr>
<th>IRS N°</th>
<th>Date</th>
<th>Country</th>
<th>Plant</th>
<th>Type</th>
<th>Status</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>220.2</td>
<td>82/06/30</td>
<td>Italy</td>
<td>Caorso</td>
<td>BWR</td>
<td>Start-up</td>
<td>Scram on high high APRM trip</td>
</tr>
<tr>
<td>571</td>
<td>85/05/28</td>
<td>USA</td>
<td>Summer 1</td>
<td>PWR</td>
<td>Start-up</td>
<td>Premature criticality during start-up</td>
</tr>
<tr>
<td>1080</td>
<td>89/03/03</td>
<td>Canada</td>
<td>Bruce B6</td>
<td>PHWR</td>
<td>Start-up</td>
<td>Incorrect positioning of adjuster rods</td>
</tr>
</tbody>
</table>
Table 10: Partial or Total Unavailability of the Scram - Control Rods

<table>
<thead>
<tr>
<th>Date</th>
<th>Report N°</th>
<th>Unit</th>
<th>Type</th>
<th>Status</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>80/06/28</td>
<td>0011.04</td>
<td>Browns Ferry 3</td>
<td>BWR</td>
<td>At power</td>
<td>Partial failure of the scram system</td>
</tr>
<tr>
<td>83/09/01</td>
<td>0475.00</td>
<td>Chinon A3</td>
<td>GCR</td>
<td>At power</td>
<td>Damaged upper internals-jamming of fine control rods</td>
</tr>
<tr>
<td>84/06/04</td>
<td>0434.00</td>
<td>Calder Hall D</td>
<td>GCR</td>
<td>Start-up</td>
<td>Failure of a control rod to drop fully during test</td>
</tr>
<tr>
<td>84/06/23</td>
<td>0507.00</td>
<td>Fort St. Vrain</td>
<td>HTGR</td>
<td>Shutdown</td>
<td>Degraded reactor shutdown systems</td>
</tr>
<tr>
<td>84/10/06</td>
<td>0506.00</td>
<td>Susquehanna</td>
<td>BWR</td>
<td></td>
<td>Four control rods fail to insert during testing</td>
</tr>
<tr>
<td>85/05/30</td>
<td>0609.00</td>
<td>Point Beach 1</td>
<td>PWR</td>
<td>Start-up</td>
<td>Two stuck control rods due to loose parts resulting from inadequate</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>installation procedures</td>
</tr>
<tr>
<td>85/07/07</td>
<td>0548.00</td>
<td>Forsmark 2</td>
<td>BWR</td>
<td>Start-up</td>
<td>IGSCC in control rod blade anti-friction plates</td>
</tr>
<tr>
<td>85/02</td>
<td>0576.00</td>
<td>Chooz</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Control rod faults</td>
</tr>
<tr>
<td>86/03</td>
<td>0856.90</td>
<td>Generic (France)</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Cladding wear on reactor control rod pencils</td>
</tr>
<tr>
<td>88/09/09</td>
<td>0922.00</td>
<td>Dampierre 1</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Rupture of one rod of a RCC assembly</td>
</tr>
<tr>
<td>88/12/23</td>
<td>0986.00</td>
<td>Blayais 4</td>
<td>PWR</td>
<td>Start-up</td>
<td>Two stuck control rods</td>
</tr>
<tr>
<td>89/04/01</td>
<td>0966.02</td>
<td>Gravelines B4</td>
<td>PWR</td>
<td>At power</td>
<td>Control rod blockage</td>
</tr>
<tr>
<td>89/09/07</td>
<td>1077.00</td>
<td>TVO 1</td>
<td>BWR</td>
<td>Start-up</td>
<td>Metallic impurities in the reactor of TVO 1</td>
</tr>
<tr>
<td>89/07/25</td>
<td>1091.00</td>
<td>Chinon B1</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Control rod drop time exceeding limits</td>
</tr>
</tbody>
</table>
Table 11: Partial or Total Unavailability of the Scram - Reactor Trip Breakers

<table>
<thead>
<tr>
<th>Date</th>
<th>Report N°</th>
<th>Unit</th>
<th>Type</th>
<th>Status</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>82/12/20</td>
<td>281.00</td>
<td>Robinson 2</td>
<td>PWR</td>
<td></td>
<td>Test failure of reactor trip circuit breaker</td>
</tr>
<tr>
<td>83/02/25</td>
<td>232.02</td>
<td>Salem 1</td>
<td>PWR</td>
<td>Start-up</td>
<td>Failure of automatic trip system</td>
</tr>
<tr>
<td>83/02/18</td>
<td>243.00</td>
<td>Beznau 2</td>
<td>PWR</td>
<td>At power</td>
<td>Reactor trip involving failure of a reactor trip breaker</td>
</tr>
<tr>
<td>83/03/03</td>
<td>233.00</td>
<td>San Onofre 3</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Failure of the undervoltage trip function of reactor trip breakers</td>
</tr>
<tr>
<td>83/04/21</td>
<td>250.50</td>
<td>Generic</td>
<td>PWR</td>
<td></td>
<td>Potential deficiencies of DS-416 reactor trip breakers</td>
</tr>
<tr>
<td>85/01/12</td>
<td>568.00</td>
<td>Sequoyah 2</td>
<td>PWR</td>
<td>At power</td>
<td>Reactor trip involving failure of reactor trip breaker</td>
</tr>
<tr>
<td>85/10/29</td>
<td>631.00</td>
<td>DC Cook 2</td>
<td>PWR</td>
<td>At power</td>
<td>Reactor trip with one reactor trip breaker failing to open</td>
</tr>
<tr>
<td>87/07/02</td>
<td>811.00</td>
<td>McGuire 2</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Reactor trip breaker failed to open on manual initiation from the control room</td>
</tr>
</tbody>
</table>

Table 12: Partial or Total Unavailability of the Scram - Other Incidents

<table>
<thead>
<tr>
<th>Date</th>
<th>Report N°</th>
<th>Unit</th>
<th>Type</th>
<th>Status</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>87/07/24</td>
<td>746.02</td>
<td>Oskarshamn 3</td>
<td>BWR</td>
<td>Shutdown</td>
<td>Criticality shutdown margin measurement started without the hydraulic scram system being operable</td>
</tr>
<tr>
<td>90/09/09</td>
<td>1123.00</td>
<td>Leibstadt</td>
<td>BWR</td>
<td>Start-up</td>
<td>Reactor start-up with degraded reactor shutdown system</td>
</tr>
<tr>
<td>87/05/26</td>
<td>1086.00</td>
<td>Cattenom 2</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Unavailability of the reactor protection system after first core loading prior to cold precrical tests</td>
</tr>
</tbody>
</table>
Table 13: Neutron Flux Monitoring Problems

<table>
<thead>
<tr>
<th>Date</th>
<th>Report N°</th>
<th>Unit</th>
<th>Type</th>
<th>Status</th>
<th>Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>83/09/09</td>
<td>474</td>
<td>St. Laurent A</td>
<td>GCR</td>
<td>Start-up</td>
<td>Nuclear instrumentation unavailable</td>
</tr>
<tr>
<td>93/02/28</td>
<td>571</td>
<td>Virgil Summer</td>
<td>PWR</td>
<td>Start-up</td>
<td>Premature criticality during start-up</td>
</tr>
<tr>
<td>87/06/03</td>
<td>732</td>
<td>Bugey 2</td>
<td>PWR</td>
<td>Start-up</td>
<td>Reactor scram through high flux variation during criticality approach</td>
</tr>
<tr>
<td>89/03/15</td>
<td>1051</td>
<td>Calverts Cliff 1&amp;2</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Potential loss of required shutdown margin during refuelling operations</td>
</tr>
<tr>
<td>90/00/00</td>
<td>1193.G0</td>
<td>Generic (France)</td>
<td>PWR</td>
<td>Shutdown</td>
<td>Neutron flux surveillance during fuel loading and unloading</td>
</tr>
</tbody>
</table>
Annex 2

Quick Refuelling in Belgium

A2.1 Introduction

In the seventies, Westinghouse developed a rapid refuelling package which allows the removal of a PWR vessel head and its associated components including upper internals in a single lift. With this design, the number of delicate operations, such as the unlatching of the control rods and its permutation into the fuel assemblies, is reduced.

The primary goal of the quick refuelling is thus an appreciable time saving tool reducing the number of delicate operations.

The radiation level is higher for the quick refuelling, but as the intervention time is shorter, the total doses are approximately the same as the normal refuelling.

A2.2 Quick Refuelling Description

To gain access to the fuel in a normal fuelling process, the following separate operations are required:

- remove the missile shield,
- install the vessel head lifting ring,
- uncouple all cable connections,
- remove the ducts of the control rod drive mechanism (CRDM) cooling system,
- remove the vessel head to expose the upper internals,
- unlatch all CRDM drive rods,
- remove the upper internals to reveal the core.

The rapid refuelling upper package combines all the operations described above into a single-lift operation by the polar crane. The new system was demonstrated successfully for the first time during construction at Doel 4 in May 1983 when the upper package was removed from its position on the reactor vessel to its storage position in the reactor cavity. Although an identical rapid refuelling system was installed at Tihange 3, only Doel 4 uses it.

The lifting process includes four steps as can be seen in Figure 2:

- connection of the lifting ring to the polar crane,
- lifting of the upper package up to a position where the lower part of the package is near the cavity seal,
- lifting the upper package outside the vessel,
- transfer of the upper package to its storage position.

A2.3 Upper Package Description

The upper package (see Figure 3) is a system which combines the vessel head lifting ring, lift columns, reactor vessel missile shield, CRDM forced air cooling system, and electrical and instrumentation cable routing into a efficient one-package reactor vessel head design.
Because the lifting ring is attached to the missile shield during plant operation, the operations of attaching and removing the ring during refuelling are eliminated.

A2.4 Quick Refuelling Savings Estimation

In 1985, an evaluation between the normal and the quick refuelling was made by the Belgian utility. The comparison of the two refuelling methods at Doel 4 shows a time saving of 3 days for the opening and the closing of the reactor and 1 1/2 days for the fuel replacement.

The radiation exposure evaluation made during the first refuelling of Doel 4 indicates that the total dose for the quick refuelling is reduced by about a factor 3 in comparison with the total dose for the normal refuelling of the fourth refuelling of Doel 3. However, the results of analyses have shown that, if the contamination of the reactor vessel head is of the same magnitude, the corresponding doses of the quick and normal refuelling are nearly the same.

A2.5 Influence of Quick Refuelling on Boron Concentration Requirement

The boron concentration has been increased from 2000 ppm to 2500 ppm to guarantee a $K_{\text{ep}}$ of 0.95 with the control rods withdrawn. The higher boron concentration has an impact on the core stability as explained in section 3 of the main report.

To limit the probability of a severe reactivity excursion during quick refuelling, the Doel 4 utility took the following additional measures:

- reinforcement of the sub-criticality controls
- anti-dilution requirements: double isolation of water sources
- availability of the control rod system as soon as possible.

A2.6 Quick Refuelling In Belgium

The quick refuelling is only in operation at Doel 4. Other Belgian units use the normal refuelling process.

Figure 2
Annex 3

Protection Against the Spurious Dilution
During Natural Circulation

A3.1. Introduction

Electricité de France (EdF) has conducted a study on the start-up of one primary pump on a diluted loop. Neutronic studies have shown that this type of event could have important consequences on the fuel and the probability of occurrence of this phenomenon obtained from the PRA study is the most important from the core melt point of view. EdF thus decided to modify its nuclear power plants. Similar modifications are also under way on all units in Belgium.

A3.2 Short Description of the Considered Event

The considered event can be shortly described as follows:

- plant status :

  * reactor coolant system :
    - hot shutdown conditions
    - primary pumps tripped (natural circulation)

  * chemical and volume control system (CVCS) :
    - dilution underway or completed shortly before the trip of thereactor coolant pumps (or volume control tank filled with pure water),

- event scenario :

  Due to the dilution process, pure or nearly pure water is injected into the reactor coolant pump casings. Proper mixing is unlikely because of the low primary flow rate in natural circulation. The start-up of one primary pump may introduce the accumulated pure water inside the core, producing a reactivity excursion with potential local boiling and core damage.

A3.3 Description of the Corrective Actions

In case of a trip all of all primary pumps, during or following a dilution period, the inlet of the chemical and volume control system pumps are to be connected to the RWSTs, with concurrent isolation of the control volume tank connection. The closure of the demineralized water supply to the CVCS can be taken as an additional protective measure.

The actuation signal is the low flow rate in primary loops or the RCP breakers open, in conjunction with a dilution signal on the CVCS make-up controller (or ended for less than
one hour) or boron supply valve closed and demineralized water supply valve open. This signal is blocked by priority key locked impulse switch. It should be noted that the closure of the volume control tank isolation valve is not made if the corresponding RWST valve is not fully open. In order to prevent any spurious dilution while in natural circulation, and taking into account that the proposed actions need some time to implement, immediate administrative instructions in the form of interim procedures should be provided to the operators.