GENERIC STUDY ON THE EVENTS REPORTED THROUGH THE NEA/IRS AND INVOLVING A LOSS OF CONTAINMENT FUNCTIONS.

Prepared by Experts from Principal Working Group No.1 of the NEA Committee on the Safety of Nuclear Installations

JULY 1988

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
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GENERIC STUDY ON THE
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Prepared by a
CSNI/PWG1 Group of Experts

PART 1: ANALYSIS

Paris, July, 1988
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## PART 1: ANALYSIS

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## ANALYSIS

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INTRODUCTION
1. Participants

This report was prepared for Principal Working Group #1 of the OECD/NEA Committee on the Safety of Nuclear Installations by a Study Group consisting of representatives from Canada, Finland, France, Italy and Spain. The members of this Group were:

G Ishack, Canada
H Heimburger, Finland
O Nevander, Finland
C Feltin, France
M Dozias, France
G Grimaldi, Italy
P Barsanti, Italy
M Citta, Italy
G la Rosa, Italy
J Reig, Spain
F Robledo, Spain
2. Background

During its meeting of September 1986, Principal Working Group #1 of the OECD/NEA Committee on the Safety of Nuclear Installations identified the need for a generic study to be performed on the events reported to the NEA-IRS which involved any loss of containment functions; this study would constitute a follow-up to the report produced by Dr Lindauer, issued November 1986, on the "loss of Safety Functions". Consequently a group was formed, to effect the said study, consisting of representatives from Canada, Finland, France, Italy and Spain.

The task was performed in six steps:

(i) Agreement on the scope of the study; i.e. components and subsystems to be included and the criteria for their selection.
(ii) Identification of the events reported to the NEA/IRS which are relevant to the study and which were reported during the period January 1980 - June 1987.
(iii) Categorization of the events into coherent groups.
(iv) Analysis of characteristic events within each group addressing, inter alia, root causes, various design features, reliability of individual components, training and procedures.
(v) Comparison of some of the analyzed events and/or aspects with units that did not experience such problems.
(vi) Extraction of conclusions and/or observations wherever applicable or possible.
3. Scope of the Study

The components/subsystems included are only those which are usually "dormant", i.e. those components whose adequate operation is necessary under unusual, transient or accident conditions. Hence, the analysis addressed only those incidents which had an actual or potential effect on the adequate functioning of the containment structure, containment isolation, containment sprays, containment coolers, and containment airlocks/hatches.

4. Event Identification

Four independent screening processes were performed successively. The first three, done by Italy, Canada, and the WEA, identified 72 incidents that had been reported to the WEA-IRS and which were relevant to the study; the fourth screening, carried out manually by members of the study group, reduced that number to 67.

5. Categorization of the Events

The 67 events were grouped into the following five categories:

Category A: Events involving containment spray problems.
Category B: Events involving isolation valve problems.
Category C: Events involving containment breach or inadvertent and/or uncontrolled release or leakage from containment, including hatch/airlock problems.
Category D: Events involving structural defects or problems in primary containment buildings.
Category E: Events involving containment coolers, or transients/incidents that affected or had the potential of affecting the containment integrity and/or functions.

The following table cites the total number of events classified in each category, the corresponding IRS report numbers, and the country that carried out the analysis.
<table>
<thead>
<tr>
<th>COUNTRY</th>
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<td>684</td>
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1. General Observations and Conclusions

1.1 This study, which addressed primarily those containment problems that were reported through the NEE-IRS, has highlighted numerous problems related to the design and adequate performance of most of the containment sub-systems, e.g. structural components, sprays, isolation valves, airlock door seals, ice condensers, hydrogen recombiners and area coolers. Some of these problems were "cascading" or "cross-linked", e.g. loss of service water to the area coolers potentially leading to adverse effects on equipment and structures (566 - US and 370 - Canada); some were independent, e.g. failure of the isolation valves to close within the required time limit, and failure of the air supply to the airlock door seals (376 - US and 508 - France); and some had the "human factor" as the cross link, e.g. unavailability of both trains of the containment sprays (287 - US), isolation valves failing to close (82.05 - US), open door between drywell and wetwell (360: - Sweden) and partial blockage of ice condenser flow channels (414.09 - Finland). The following questions, among others, thus arise:

a- What is the degree of dependance of "adequate" containment performance on each of these sub-systems; i.e. to what extent is each sub-system credited in the accident analysis?

b- To what extent is each of the following considered or factored in the accident analysis:

(i) feedback from operating experience,
(ii) the human factor as the potential cross link leading to several failures and/or unavailabilities, and
(iii) the simultaneous failure and/or unavailability of more than one sub-system.

1.2. The importance of learning from experience acquired by others, and of making the most of the information circulated through the NEE-IRS, is underlined by the similarities among several of the events which occurred in different countries and which were reported at various points in time. Examples are:

- containment impairments due to problems with airlock door seals (five events in Canada and six in France),
- degradation of containment integrity due to inadequate inspection of anchor tendons (event 562 - US and event 113 - Canada), drywell/suppression chamber problems (288 and 565 - US, 360 - Sweden and 414.09 - Finland), and others.

This furthermore raises the following questions:

a- Why were such problems not reported by others?
b- How can countries/utilities cooperate to solve a common problem?
In this regard, it is worth noting that events 566 and 610 were triggered by failures in rubber expansion joints in cooling water systems; in one of the Canadian plants, failures in service water expansion joints (which did not affect containment) prompted the licensee into replacing all such joints by metal spools.

1.3 There are several cases where:

a- lessons learned at one plant were either not transferred to or were ignored by others (within the same country), or
b- recommendations issued at a certain plant were not followed through to completion, thus leading to the recurrence of the event, or
c- both (a) and (b) seem to exist.

This leads to the question as to whether the human factor encompasses an aspect which is not addressed or studied as frequently as "Operator Errors", namely the follow-up process within a plant or an organization which ensures the existence of adequate feedback.

Another important human factor aspect is demonstrated in event 738 (Section 3.2.3.6) whereby the Architect/Engineer accurately followed deficient design drawings supplied by the WSSS vendor.

1.4 It is not clear, in many of the event reports, if a "parametric study" of all the contributing factors were carried out to determine the most "critical" cause; i.e. in several incidents, the corrective actions consisted of design changes, improvements in training, re-writing procedures and strengthening administrative controls and/or surveillance inspections. Whereas in most cases these actions seem to have rectified the problem at the respective plant, it is not obvious which of them corrected the "weakest" point. Such an exercise would help identify preventive measures to be taken to avoid the occurrence of other events.

1.5 Inadequate testing, maintenance and operating procedures have contributed to about 19% of the events in this study; hence the necessity of emphasizing the importance of such procedures. This situation may possibly suggest that not all the potentially serious scenarios were considered in the design and pre-operational stages.

1.6 One typical recurring operational problem, often connected with the failures to follow procedures, is the restoration of systems and equipment after maintenance or after any manoeuvre requiring a lineup different from that needed for high power operation; hence the necessity of reviewing such events (and others) with maintenance personnel
1.7 In all the 10 events involving containment spray problems whether impairments or inadvertent operations, human factors were contributing causes; in most cases, human errors were the only causes. These errors included: personnel training, writing procedures, following procedures, carelessness, and administrative controls.

1.8 The degradation of the containment pressure suppression capability of the PS-type containment is always a very serious event, that will jeopardize the whole containment structure if a LOCA were to occur. In this study, five incidents have been recognized as belonging to this category. It is reasonable to assume that other incidents of this type have occurred but were not necessarily reported through the WEA/IRS. It would then seem that we are faced with two choices, to avoid the recurrence of such failures:

---

to (re)build the plants in such a way that they will tolerate the loss of the PS-principle (or event better, tolerate human errors at least to some extent), or

---

to put more emphasis (once again) on the training of personnel and on the development of procedures.

The Finnish BWR-units designed by Asea-Atom Sweden (almost identical to the US Mark II- type containments) will be equipped during the annual refuelling outage of 1988 with high capacity large diameter overpressure protection systems. The system includes a 600 mm pipeline with two block valves normally open and a rupture disc designed to rupture before the containment structural pressure capability limit would be exceeded.

1.9 As in the case of inadvertent actuation of fire protective systems (e.g. deluge systems), the inadvertent operation of containment sprays raises the question of environmental qualification of equipment whose availability is required after being exposed to water spray or after being submerged under water (see for example events 176.02 - US and 574 - Canada).

1.10 The single most frequently reported problems seem to be those related to performance of isolation valves; these account for 39% of all the events analyzed in this study, with the human factor again being a prominent contributor. The most significant failures that were reported to have occurred in isolation valves are:

a- stem galling, binding in the valve packing and impure actuator oil (reported in US and Spanish plants);
b- separation between the valve’s main disc and its stem (reported in US and Spanish plants), and

c- "hydraulic lockup" of limitorque valve operators leading to valve failure due to grease viscosity and/or design features; this was furthermore considered as a situation potentially leading to common mode failures of numerous limitorque valves in several safety-related systems.

Furthermore, containment leak tightness seems to have been significantly affected by corrosion of carbon steel pipes at certain penetrations (e.g. event 522 - Italy)
1.11 Incidents of "hydraulic lockup" were reported to have occurred at Limerick, USA on motor operated valves, after the modifications recommended by the manufacturer (Limitorque) had been incorporated. This suggested that provision of an internal relief path (the design change effected) does not provide adequate protection against hydraulic lockup under all circumstances of valve operation.

1.12 This study has highlighted a certain type of incident where the containment integrity was unaffected. Rather, the containment was completely by-passed resulting in uncontrolled external leaks of the reactor coolant. Some of the causes of these occurrences were operator/maintainer errors (36 and 69 - France and 688 - US) in manipulating and/or lining up valves, and failure of the scram discharge volume drain valves to close fully (375 and 572 - US); the latter two events furthermore highlighted several design and procedural deficiencies.

Two additional points are worth emphasizing here:

a- Event 375 was initiated by a valve disc disengaging from its stem - a failure similar to that described in 1.10 -a above.

b- Event 572 was triggered by actuation of the fire deluge system, hence comment 1.9 also applies.

1.13 In the systematic "human performance assessment" approach adopted by Finland, in the analysis of eight of the events included in this study, some generic remarks were made (see also Appendix I), excerpts of which are:

a) In the Finnish BWRs, during the annual 1987 training course, 2-3 days were devoted to work order practises and procedures.

b) Many of the problems identified (valves forgotten in wrong positions, inadequate personnel training, and poorly written procedures) are associated with a lack of personnel motivation.

c) Procedures and instructions should be updated periodically to ensure that consistency and homogeneity is eventually reached, thus making their use easier under abnormal conditions.

d) Independent inspections should be added to the internal routines of a given plant.

e) Technical Specifications should require that work on redundant trains or components be done one at a time.
1.14 Many case studies have been undertaken to evaluate the performance of the containment systems as a whole or to assess the performance of some of the containment sub-systems. Two examples are:


1.15 This generic study could constitute a useful input to work currently being done by specialist groups studying various aspects of containment design, performance and integrity under accident conditions, and to experiments such as those being done by Sandia Laboratories on scaled-down models of primary containments. Of particular interest in this regard are perhaps those incidents in which:

a) containment pressure and/or temperature transients occurred (6.03 - Japan and 566, 610 - US);
b) containment area coolers were unavailable (370 - Canada);
c) hydrogen recombiners were inoperable, either due to
   (i) a short circuit caused by actuation of the fire deluge system (176.02 - US), see also comment 1.9, or
   (ii) incorrect wiring (479 - Switzerland);
d) latent problems were identified in structural components (e.g. deterioration in anchors and wire tendons, 562 - US; deterioration in vacuum building roof rubber seal and tensioned tendons, 113 - Canada and corrosion and missing structural welds in the torus, 455 - US); and

e) significant degradation in leak tightness was identified in isolation valves (e.g. 336 - US) and penetrations (e.g. 522 - Italy)
2. Category A: Events Involving Containment Spray Problems

2.1 Incident Summaries

See Appendix A

2.2 Comments

2.2.1 10 events, constituting about 15% of the total, were classified in this category.

2.2.2 In all but four of the events (27, 75, 437.05, and 574) the containment spray system was impaired. The impairment in four cases was total (287, 448, 416.01 and 416.02). The longest total spray system impairment reported (287) was 1.5 years.

2.2.3 In all the events but one (27), procedural inadequacies were contributing causes. Additionally, poor administrative controls and/or organizational QA were identified as primary causes in 292, 684 and 574, and inadequate training was highlighted in 448 and 292. An error in unit occurred in event 437.05.

2.2.4 In four incidents (27, 75, 437.05 and 574) the containment spray system was inadvertently actuated. In addition to adverse effects on equipment (instrumentation, motors, electrical junction boxes, etc.) such an occurrence renders the spray system unavailable, in at least the CANDU-600 design, until the "dousing tank" is re-filled (which operation requires about 12 hours to complete).

2.2.5 One incident (27) involved a single inadvertent operator error; while another (448) involved several consecutive operator errors of omission. Human error in the other incidents contributed to varying degrees (design, training, omission, supervision...).

2.2.6 The primary actions identified in most cases were improvements in procedures, personnel training, and administrative controls.
3. Category B: Events Involving Isolation Valve Problems

3.1 Incident Summaries

See Appendix B.

3.2 Comments

About 39% of all the events analyzed in this generic study (26 out of 67) lie in the category of isolation valve problems. These 26 events will be addressed in the following three groups:

3.2.1 Events 6.02, 182, 190, 336, 361, 522 and 163.01 to 04

3.2.1.1 Hardware problems were identified in 8 out of the 10 events analyzed in this group: 7 of these involved mechanical failures (including one with an additional chemical control problem) and one involved an electrical failure.

3.2.1.2 Vibration-induced failures featured in 4/10, thus constituting the most prominent single mechanical problem in this group.

3.2.1.3 The presence of foreign materials was identified as the observed cause in 2 events.

3.2.1.4 Human factor problems were identified in 9/10 cases.

3.2.1.5 Design deficiencies were identified as the root cause in 4/10 events and maintenance deficiencies in 5 others.

3.2.1.6 No events were reported where errors were identified in the areas of management, organization or work planning.

3.2.1.7 In 7/10 cases a combination of hardware and human problems contributed. In particular, event 522 cites as contributing causes corrosion, foreign material, poor chemistry control, maintenance and procedural deficiencies, all of which led to multiple failures (excessive leakage of isolation valves).

3.2.1.8 In event 182 four different causes attributed to human error were identified: inadequate training, lack of clarity in the procedure, and carelessness in applying the procedure all contributed to the operator sequencing error, namely, opening the inboard MSIVs before opening the corresponding outboard valves.

3.2.1.9 The consequence of event 182 (referred to above) was a pressure transient which cracked the hydraulic speed control cylinders of both series valves.
3.2.1.10 Event 190 is the only one which resulted in an external release (20 Curies of noble gas through the plant stack); in most other cases the most adverse consequence was a reactor trip.

3.2.1.11 Leaktightness problems were identified in two reports (336, and 522): the loss of containment functions was highlighted as a potential consequence in 5 cases.

3.2.1.12 In 8 of the events, only single failures occurred, leading to degraded performance (instead of a complete loss) of the affected function; no system interactions were reported.

3.2.1.13 Based on the above general comments, two types of events were selected for in-depth analysis:

   a) events involving vibration problems and design deficiencies as the direct and root causes respectively, and
   b) events involving a degradation in leak-tightness.

   a) Events involving vibration and design deficiencies

These problems are reflected in incident reports 163.01 through 04. The concern here is the mechanical separation between the MSIV main disc and its respective stem; this type of failure was reported at the US plants of Brunswick 1 & 2, Edwin L and Hatch. These events (and three other similar ones) were described in Section V-C of "Nuclear Power Experience" (issued by Stoller Corporation) and in the September 3, 81 US NRC IE Information Notice No. 81-28. Yet another similar event was reported on May 17, 1987 on an outboard MSIV in Cofrentes, Spain.

In one particular case (163.04) the failure occurred on two inboard valves of two steam lines, possibly simultaneously; in 3 of the 4 events a maintenance deficiency was also reported during the valve re-assembly.

In all cases, the main valve disc became free of the stem and in two cases it dropped into the body causing a pressure transient and a subsequent reactor trip due to the resulting spike in power.

The separation in these incidents was mainly due to excessive vibrations on the main steam lines causing a rupture of the anti-rotation pins and of the threaded connections between the main disc and the stem of the valve. The design deficiencies, identified as the root cause, had the following as contributors:

--- wrong or missed installation of the anti-rotation pins, and/or,
--- instabilities in the steam flow, due to the configuration of the piping which connects to the inlet of the inboard MSIV.
Valve design modifications were implemented as corrective actions.

Apparently, the most significant safety-related concern in this regard is the potential unavailability of both series MSIVs, caused by a plant transient, and resulting in a leakage path bypassing the containment.

b) **Events involving degradation in leak tightness**

Significant degradation in leak tightness was reported in events 336 and 522, mainly due to problems related to maintenance and corrosion respectively. Report 336 mentions that for some isolation valves, grease was used despite the manufacturer's recommendations. Unacceptable local leak rate test (LLRT) results led to an investigation which identified the problem. Tests conducted after the valves had been cleaned revealed excessive leakage rates on almost all of them. Furthermore, report 522 attributes the excessive leakage past several local penetrations (identified in LLRTs) to the following:

- significant corrosion of the carbon steel pipes in the containment purge system and in the closed cycle cooling water;
- rust fragments and dirt build-up in the isolation valve seats resulted in damaging the teflon gaskets during the valve closure;
- frequent actuation (once every two days) of the purge system valves;
- the presence of dirt and impurities in the water of the closed cycle cooling system; and
- maintenance and procedural deficiencies.

Corrective actions included repairs/replacements as necessary of the valves and gaskets as well as painting the inner surface of the purge system pipes with an anti-rust material.

Tests performed in 1987 on US plant containments, particularly Oyster Creek and Brunswick, yielded results that are similar to those reported in 522 (for Coarso, Italy); namely, some penetrations could not be pressurized to the test pressure, because of excessive leakage. In some other tests (Shoreham 1 and Brunswick 1) leakage past one penetration was beyond the instrumentation range.

From the above we can conclude that:

- the combination of unprotected carbon steel piping and/or butterfly valves, and water (or humid air) could result in an adverse effect on containment local leak tightness, and hence should likely be avoided;
- maintenance and test procedures should stipulate that test results be collected and analyzed both before (as found) and after (as left) maintenance, to evaluate any degradation during the period between any two consecutive tests as well as to evaluate the overall maintenance and test programmes.
All the above issues were analyzed in Italy, as a follow-up to event 522. The pertinent results are presented in detail in the following papers.

(1) "Significant aspects concerning the leakage tests of containments: operational experience evaluation", by G Grimaldi, presented at the CSNI, specialist meeting on Water Reactor Containment Safety held in Toronto, Canada, October 1983;

(11) "Pre-existing containment openings", by S Ciattaglia et al., presented at the ANS meeting on Operability of Nuclear Power Systems in Normal and Adverse Environments, held in Albuquerque, US, September 1986;

(11) "Leak tightness capability conservation of Caorso WPP primary containment" by P Barsanti and G Grimaldi, presented at the NEA PWG 1 meeting held in Paris, France, September 1987.

Specifically, in paper #11, a "search" was effected in the LER file and in the "Nuclear Power Experience" for events similar to the one described in report 522. Whenever quantitative leak rate data were reported, the corresponding equivalent leak areas were calculated according to the formula (taken from the Mechanical Engineers Handbook):

\[ A = 100 G/1.38 \text{ P}_{\text{a}} \]

where

- \( A \) = Equivalent area (\( \text{cm}^2 \))
- \( G \) = Leakage flow (kg/s)
- \( \text{P}_{\text{a}} \) = Internal test pressure (kg/cm\(^2\))

A probability was then assigned to each value of equivalent area and the resulting plot is shown in the accompanying figure.

3.2.2 Events 82.05, 376, 389, 398, 406, 616, 662.02, 673 and 743

3.2.2.1 Four of these incidents (82.05, 376, 389 and 662.02) involved closing failures in PWR main steam isolation valves. Two and three MSIVs failed to close during shutdown testing in events 389 and 662.02 respectively, while in 82.05 and 376 the required closure times (of less than 5 seconds) were exceeded. The following observations are offered, concerning these four events:

a) Root Causes

The root causes for events 82.05, 376, 389 and 662.02 were:

(1) Heavy stem galling was reported in 376 to have extended to the junk ring located inside the packing gland. Junk ring steel is softer than the stainless steel stem, therefore galling may occur. This tended to bind the stem, slowing valve closure.
$P_{c_i}$ = Cumulative probabilities on demand, for areas exceeding a given $A_i$ (events/year). $A_i$ is the upper bound of the $i^{th}$ range of the preexisting opening.

$C_{c_i}$ = Cumulative failure rate, (subscriptes have the same meaning as for $P_{c_i}$).

A - Areas in the range A represent single penetration failures and should be additional to the area corresponding to T.S.

B - Areas in the range B are dominated by large failures of penetration exceeding the areas corresponding to T.S.

Probability of preexisting openings as a function of the equivalent area.
(i) Two apparently unrelated causes were identified in 389:

- Binding in the valve packing. The inspection of the valve packing revealed no apparent hardening or other degradation.
- Separation of the valve disk from the disk arm because of failure of all four disk fasteners.

(ii) Excessive dirt in the oil of the actuator and excessive oil pressure were pointed out as the root causes in 82.05 and 662.02. Two reasons were postulated for these problems: inadequate maintenance and the temperature in the actuator being above the limits determined by the manufacturer.

b) Reliability of Individual Components - Periodic Tests

The STS for Westinghouse plants require verification of the time for full closure according to ASME XI Code (once every three months; although this frequency may be increased to once per month). In addition, ASME Section XI IWV-3400 requires that MSIVs be tested by partial stroking during operation and by full stroking during each cold shutdown.

c) Design Features

The following information concerns Asco 2. The MSIVs were manufactured by Rockwell International. The actuator was manufactured by Greer Hydraulic. Fig 1 shows a schematic of the MSIV. The valve body and the piping form a 45 degree angle. The seat is designed for bi-directional flow. The actuator operates by applying high pressure nitrogen to a hydraulic piston. This pressure tends to close the valve and is counterbalanced by oil. Valve closure begins when this oil is released. This discharge is controlled to regulate the valve closure speed.

Calvert Cliffs had originally the same MSIV type. Because of operational problems a MSIV Project team was formed in early 1985 and concluded that the existing MSIV internals and actuation system were inadequate and should be changed. The internals were supplied by Rockwell International and the actuator is a Rockwell type A design. Information about the features of these new devices is unavailable.
d) Other Comments

When the incident in Asco occurred, the following additional information concerning Calvert Cliffs was supplied by the US NRC to the Spanish CSN:

On July 24, 1985 the MSIV number 21 did not close. The root cause was an oil deficienc due to a gas bubble in the hydraulic fluid header. Post-maintenance testing indicated:

-- The bladders of both high pressure hydraulic suppressors were replaced.
-- The check valve which, following a stroke, isolates the cap end of the actuator from the rest of the actuation system was replaced.

On December 12, 1985 the MSIV number 11 was rendered inoperable. The root cause was an increasing hydraulic fluid leak from the lower end seal on the piston, which could damage the valve seat during the closure.

On December 12, 1985 the MSIV number 12 failed to close (80% shut) when automatically initiated from the remote station. Several valves and procedures were changed to enhance the MSIV reliability.

3.2.2.2 In incident 398, routine surveillance testing was being performed on the steam and feedwater rupture control system (SFRCS). A wiring anomaly caused a failure in the optical isolator in a relay driver card in the SFRCS channel 4, previously undetected, thus causing the loop 2 MSIV to close.

Following the high flux trip (caused by the subsequent overcooling) main steam safety valve (MSSV) SP17A4 failed to close fully, MSSV SP17A1 failed to open when it should have and an auxiliary feedwater valve (AFW) failed to open. The MSSV problems were due to equipment failures, one being the failure of a cotter pin that secures the release nut at the top of the stem. The AFW valve failure was due to the torque switch setting. The event thus can be attributed to three root causes: installation/ construction errors, equipment failure and a maintenance deficiency.

As a consequence of the incident, the cotter pins in all other MSSVs were replaced, and maintenance procedures were modified to ensure new pins are used after any maintenance or testing in the future.

The cooldown rate exceeded B&J guidelines, but was within Technical Specification limits, therefore some Plant Procedures have been modified to incorporate lessons learned from this event.
The AFW valve problem was solved by changing the motor operator torque switch settings.

As far as periodic testing is concerned, STS for Westinghouse plants require that full MSIV closure time be verified according to ASME XI code once every three months (although this frequency may be increased to once per month). In addition, ASME XI Section INW-3400 requires that MSIVs be tested by partial stroking during operation and full stroking during each cold shutdown.

STS for Westinghouse plants require that the set points of each safety valve be verified at least once every five years.

STS for Westinghouse plants require that verifications be done at least once every 18 months, to ensure that each automatic valve in the AFW flow path actuates to its correct position upon receipt of an AFW test signal; furthermore, pumps should be tested properly from a safety point of view: startup adequately, provide the necessary flow,... Finally, every month it should be checked that no valve is blocking the flow path.

3.2.2.3 In events 406 and 673 spurious and/or incorrect signals led to MSIV closure. The MSIVs though, actuated properly during both incidents.

3.2.2.4 In incident 616 a loss in instrument air pressure caused, among other things, one MSIV to drift closed. This though can be considered to have been a "fail safe" condition that did not adversely affect the containment function.

3.2.2.5 The last event to be analyzed in detail in this group is # 743, which involved the failure open of a motor-operated valve (MOV), during a refuelling outage, due to motor burnout.

The root cause was determined subsequently to be related to the use of new less viscous grease in the valve operator; this grease had entered the spring pack area and apparently created a condition of "hydraulic lock-up" which prevented the spring pack from performing its intended function. The following observations are offered:

(1) **Design features**

This MOV is a limotorque SMB motor operator. The normal load application to operate the valve assembly, is through the Bellville spring pack compression, in reaction to the motor (or handwheel on SMB-0 and larger). The usual protection for the closing operation of the valve, would be the torque switch trip based on the predetermined load and corresponding spring pack deflection, based on the spring constant. If the spring pack area fills with grease that cannot escape when the spring compresses, then the spring constant effectively relates to the compressibility of the volume of grease (essentially incompressible) or "hydraulic lockup". Thus, the loads can increase dramatically and with no relationship to the spring deflection necessary to actuate the torque switch for protection of the MOV or completion of the desired MOV function. The old grease used was Mobil EP-1 and the new one is EXXON NEBULA EP-0.
The corresponding corrective action was the installation of relief kits, that would provide a flow path for the grease.

(11) **Reliability of individual components**

The STS stipulate the following periodic tests for containment isolation valves:

- Once every 18 months a containment isolation test signal must be verified to actuate each isolation valve to its isolation position.
- The isolation time of each power operated or automatic valve shall be verified according to ASME Section XI IVW-3400 (once per 3 months).
- After maintenance, repair or replacement work each containment isolation valve shall be demonstrated OPERABLE by cycling the valve and by verifying the specified isolation time.

(111) **Other similar occurrences**

Two events involving hydraulic lockup have occurred at Limerick Unit 1. The first event was in February 1984 and involved the MOV HV-57-111. This MOV had incorporated the design change explained in the DESIGFN FEATURES section. New relief paths were made in the torque limiter sleeve to solve the problem. EXXON WEBULA EP-0 grease was used in this MOV. The second event occurred on December 18, 1985 and involved containment spray header outboard isolation valve HV-51-1F016A. EXXON WEBULA EP-0 was used in this MOV. Relief paths were provided in the valve.

(iv) **Safety significance**

The "hydraulic lockup" phenomenon is a possible common mode failure mechanism for limitorque SMB motor operators. The phenomenon also appears to have the potential for many types of MOV assembly failures. In addition to motor burnout, it could possibly lead to valve operator component or valve body failures due to excessive loads. In Vermont Yankee, 40 MOVs, 32 with safety functions, had the potential for hydraulic lockup. These valves belonged to the reactor recirculation system, RHR, reactor water cleanup system, HPCI, core spray, RCIC and main steam and feedwater system. From these events, it seems that plant operation conditions, rate of load applications, type of grease used in the motor operator and motor operator design characteristics may have an impact on whether hydraulic lockup will occur.
However, the events at Limerick suggest that
the design change (internal relief path)
introduced by the manufacturer (Limiter) of the motor operator will not be effective to
prevent hydraulic lockup under all circumstances
of MOV operation.

3.2.3 Events 288, 360, 414.09, 559.01, 559.02, 565 and 738

Out of this group, some events were taken as typical examples for
detailed analysis.

3.2.3.1 Event 288

During a routine test, a drywell/suppression chamber vacuum breaker
at Quad Cities - 2 failed to close; steam from a break inside the drywell would
thus bypass the suppression pool such that quenching would not occur. In
principle, this situation would have the same consequences, in the event of a
LOCA, as the Oskarsham incident # 360 described in Section 3.2.3.2 below.

The direct cause of the incident was binding of the stainless steel
packing and stuffing box bushing on the valve shaft, that caused the valve to be
stuck open. The valve was later freed from its shaft and lubricated properly.
Since the root cause of the incident was not discussed in the report, it could
only be assumed that the binding was either due to inadequate maintenance (poor
lubrication) or bad design. Also not discussed in the report are the long term
corrective actions.

Based on the information given in the report, it seems that the
drywell/suppression chamber vacuum breakers are of the single rather than the
double design (i.e. two series check valves) used in all the Swedish and Finnish
pressure suppression type BWR plants. To achieve opening and closure of the
vacuum breakers with a high confidence level, given a single failure, the number
of vacuum breakers lines as well as the number of check valves per line, should
be analyzed in detail.

3.2.3.2 Event 360

At the beginning of the annual (1982) outage of Oskarsham -2, an
inspection of the reactor containment identified an open door, between the
drywell and the wetwell, having an area of 1.5 m². This door, which had
possibly been open from September 1981 to July 1982, could have led to a major
leakage through the diaphragm.

In this case, the direct cause was human error, which was due to
the root cause of inadequate work control and surveillance. Some design
modifications were effected to avoid a recurrence.
3.2.3.3 Event 414.09

During a safety inspection conducted by the Finnish authorities (STUK) at Loviisa, during March 1984, it was found that some of the flow channels of the ice condensers were partially blocked. Similarly to event 360 (above) the direct and root causes were human error (carelessness) and inadequate work control respectively.

It should be pointed out that an unobstructed path is necessary from the lower to the upper part of containment, through the ice condenser, to prevent excessive pressure build up in the lower part during the early stages of an accident; to prevent the steam from escaping to the upper part uncondensed, the ice must be evenly distributed and the flow channels through the ice condenser must be as narrow as possible.

Carelessness must have occurred in the process of adding ice to the ice baskets (to counteract sublimation) during the annual maintenance outage.

Procedures were developed to prevent a recurrence.

3.2.3.4 Event 565

The unit (Catawba) entered the hot shutdown state with 23 out of the 24 ice condenser lower inlet doors blocked closed. About 10 days later, during startup testing, a health physics technician discovered the situation when he entered the ice condenser to perform a biological shield survey.

The following incident, almost identical to the one above, occurred at the Loviisa plant.

On June 2, 1983 at 10.45 hrs. it was discovered in the control room that some of the lower doors of the ice condenser could not be kept closed after an examination had been carried out in the ice condenser. The shift supervisor notified the group that had performed the examination and asked them to close the doors. Earlier, difficulty was experienced in creating the necessary pressure difference between the ice condenser and the containment atmosphere (during normal operation, the pressure in the lower part of the containment is a little lower than that in the ice condenser, which keeps the doors tightly closed). Because the necessary pressure difference did not come about easily at that time, and because the workers' lunch hour was about to begin, they closed the lower doors in one half of the ice condenser with wedges. Their intention was to remove the wedges after they had eaten.

A process operator in the shift staff noticed the wedges during an inspection round at 12.15 hrs whereupon he removed them immediately.

This enabled opening of some of the doors. It was then found out that the problem was caused by the fact that the lever-arm shutters in the floor sumps of the ice condenser had not closed properly. Therefore, the necessary pressure difference could not be attained between the containment and the ice condenser.
To close the lower doors, the process operator put some sheets of paper on the leaking floor sumps, after which the correct pressure conditions were attained and the doors could be made to remain closed. This temporary solution would have allowed the opening of the doors, which is important to safety, and therefore, it was more justifiable than the closing of the doors with wedges.

As a further action, the leaking shutters of the sumps were cleaned and the paper sheets were removed.

3.2.3.5 Some Generic Observations related to the four events mentioned in Sections 3.2.3.2 through 3.2.3.4

(1) Root causes

In the three events which occurred at ice condenser plants, the deficiency was due to human error and the direct causes can be traced back to work control deficiencies. The event that occurred at Oskarsham can be included in the same group, as well. In all the events it seems that no control of the work done or believed to be done had been performed. Therefore, these degradations of safety systems were discovered either fortuitously or after some time, via an inspection. Thus the major root cause was inadequate work control and surveillance.

(ii) Corrective actions

In all cases, corrective actions were immediately carried out to restore the safety function capability. Design modifications have been performed (Oskarshamn) and procedures have been developed (Loviisa, Catawba) to eliminate the root causes as well.

(iii) Design features

The problems associated with ice condensers are the same with all ice condenser containment units. Because of the ice sublimation effects there is a need for adding ice into the ice baskets during every annual outage in order to keep the total amount of ice at an adequate level to meet the Tech Spec requirements. Because of the high temperature differences between the ice condenser and the reactor containment inner atmosphere in some parts of the ice condenser the sublimation effects can be very dominant. This could lead to an uneven ice distribution in a manner that could jeopardize the ice condenser operation. The annual loading of ice is a costly operation and requires good work control to ensure that the requirements set for the total amount of ice as well as for the cleanliness of the flow channels between the ice baskets, will be met. There are hardly any major design modifications to be imagined that would give a final solution to these problems - the emphasis should be put on working procedures and work control.
3.2.3.6 Event 738

The reactor-building-to-torus vacuum breakers, that were found inoperable, are designed to protect the steel structure of the torus against negative pressure loading should a vacuum be drawn in the torus.

A lack of detail in the design drawings resulted in the vacuum breaker tubing being incorrectly connected; the differential pressure transmitter which controls the valve opening was installed backwards. The tubing "as found" condition conformed to the "as built", hence the conclusion that the root cause was a design error. In that configuration, the vacuum breakers would have failed to satisfy the design intent and would have opened on increasing torus pressure (relative to the reactor building). Studies undertaken by the licensee showed that in the event of a LOCA inside containment, the torus could have been pressurized and primary containment integrity could have been jeopardized.

This event is an example of possible communications breakdown between the NSSS vendor and the Architect/Engineer/Constructor. Generally the NSSS vendor provides the design drawings of the pressure suppression system and the Architect/Engineer follows that information; in this case, deficient design drawings were followed.
4. Category C: Events Involving Containment Breach, Unplanned Releases and Airlock Problems

4.1 Incident Summaries

See Appendix C.

4.2 Comments

Eighteen events, or about 27% of the total analyzed in this report, were classified in this category. They are addressed below in three groups.

4.2.1 Events 36, 69, 224, 276 and 508.01 to 06

4.2.1.1 Human error was a contributing cause in 60% of the cases. Errors included improper valve lineup, failure to follow procedures and poor communications.

4.2.1.2 In 60% of the events containment integrity was impaired via airlock door seal malfunctions.

4.2.1.3 In three of the incidents (Tricastin 1 and 2 and Bugey 2-F) cross links leading to a common mode failure were identified on the air supply to two doors of an airlock. These led to design changes which include redundant air supplies, automatic air supply isolation on low pressure and modifications to the "airlock trouble" alarm.

4.2.1.4 In nine of the events (50%) there was a complete unavailability of the third barrier of containment: six via the airlock doors, one due to a passing valve (that links RHR with the pool cooling and cleanup system), one due to rupture of a sight glass in the primary circuit, and one due to over-filling of a storage tank.

4.2.1.5 The six incidents described in 508 have resulted in regulatory action related to:

a- Impress upon personnel the importance of maintaining the third barrier of containment;
b- Review of the airlock door seal air supplies;
c- Re-assessment of the control room alarms (related to containment);
d- Studying the feasibility of having "solid" airlock door seals, thus eliminating the need for air supplies.
4.2.2 Events 375 and 572

Both of these events resulted in an uncontrolled leakage of reactor coolant outside the primary containment. The cause was failure of the Scram Discharge Volume (SDV) drain valves to close fully, thus resulting in an overall containment bypass.

4.2.2.1 The initiating event in 375 was a disengagement of the main disc from the stem of a Rockwell MSIV; this phenomenon was discussed previously in Section 3.2.1. In this case, though, several other contributing factors aggravated the event:

- design improvements recommended by the manufacturer had not been implemented;
- a relief valve was blocked;
- high inventory loss through the main steam drain lines caused a significant level transient after the first scram signal (on high power);
- a missing threaded pipe cap on the drain system in the RCIC equipment room resulted in ingress of reactor coolant into that room and a subsequent RCIC isolation on high temperature; and
- defective drywell load shedding logic caused isolation of the drywell chiller.

The following corrective actions were implemented after the event:

- MSIV design modifications;
- safety and SDV isolation valve repairs;
- RCIC instrumentation inspection, calibration, testing and/or replacement, as required;
- study of a new vacuum breaker design;
- improvements of SDV vent and drain valve surveillance procedures and future installation of redundant valves;
- upgrading of administrative controls over drain hubcaps; and
- review of BWR emergency procedure guidelines developed by the BWR Owners Group for this type of event (the results of which were analyzed by the NRC).

The following lessons learned are highlighted:

- the scram discharge system can be a potential path for leakage of reactor coolant;
- the operability of the SDV vent and drain valves should be more effectively assured and verified;
- a flexible load shedding logic is necessary;
- correct equipment restoration after maintenance is important;
- consideration of potential common cause failures due to effects external to the plant, in fact adverse environmental conditions caused the unavailability of RCIC, in this case.
4.2.2.2 In event 572, the observed cause related to the containment function was the failure of the two SDV drain valves to fully isolate after a scram signal due to malfunction of a pressure regulating servo-valve. This resulted in a sustained leakage of reactor coolant, as it was not possible to reset the scram signal because of pressure and level transients. The first valve failure was due to an improper stroke adjustment, as a result of a calibration error; the second one was the inadvertent opening of the closed valve because of the undersized actuator closing spring, as a result of an installation error.

The event highlighted a potential loss of safety functions as a consequence of the overall containment bypass. This effect was due to the fire deluge system actuation caused by the hot reactor coolant which drained into the RBEDT (Reactor Building Equipment Drain Tank) and flashed to steam.

The following corrective actions were implemented after the event:

- The pressure regulating servo-valve was rebuilt, the filters for the valve replaced and the oil tubing supplying the valve flushed;
- SDV drain valves were repaired: the stroke was adjusted on one, and the actuator spring was replaced with a properly sized one on the other;
- The event was incorporated in the operator training programme;
- The design of the pressure regulating valve was reviewed;
- The scram logic design, which does not permit bypassing the scram signal until reactor pressure decreases below 600 psig, was also reviewed.

The following lessons could be learned:

- The importance of discussing abnormal events with the maintainers and the operators, and the necessity of O.A. improvements during installation and replacement operations (to avoid the installation of incorrect parts);
- The scram discharge system can be a potential path for leakage of reactor coolant;
- Area events pose potential common cause failures, in fact in this event flooding had the potential for damaging safety systems.

The aspects common to both events, highlighted from the analysis are:

- Bypass of the containment function through the SDV drain lines;
- Problems in resetting the scram signal;
- Area events causing system interactions and actual or potential damage to safety systems.
4.2.3  Events 111, 115, 306, 551, 626 and 688

4.2.3.1 In each of 111, 115 and 306 a breach of containment occurred due to both doors of a reactor building airlock being open simultaneously. These containment breaches lasted for 45 seconds, 33.5 minutes and 3.7 minutes respectively and were terminated by operator action. In incident 115, pressure inside the reactor building rose by about 50%.

The incidents were caused by incorrect modifications during commissioning, later rectified (111), operator error and poor annunciation design (115) and simultaneous failure of four valves in the airlock door seal air supply circuit (306).

4.2.3.2 In event 551, inadequate fan startup procedures, later revised, resulted in a temporary ventilation and hence pressure imbalance in the reactor building, which in turn caused strong air suction through the equipment airlock leading to damage to the door as well as to the interlock circuit. At the time of the incident, equipment was being moved through the airlock (no injuries or releases occurred).

4.2.3.3 The air supply valve, for the door seals of one of the reactor building interlocks, had been inadvertently left partially open during repairs on that airlock (event 626). Partial seal deflation resulted causing a breach of containment. Inadequate annunciation design contributed to the length of the impairment.

4.2.3.4 Event 688, which is similar to 375 and 572 (see Section 4.2.2), resulted in a 23 minute uncontrolled coolant leakage outside the primary containment, which caused contamination of the lower 3 levels of the reactor building.

While recovering from a trip, channel B reactor protection could not be reset (only channel A was reset). In this configuration the air pressure system (operating the scram discharge system valves) was at a lower than normal pressure. This resulted in the scram discharge volume vent and drain valves being open while the scram inlet and outlet valves were unable to close. This valve lineup allowed reactor cooling water to flow through the control rod drive seals into the reactor building through the open scram discharge volume vents and drains.

The following were the contributing causes:

- Maintainer error in accidentally moving a circuit card; this resulted in closure of the turbine control valves and a subsequent reactor trip.
- Stuck contacts on the reactor mode switch caused the failure of channel B to reset.
- Air leakage in the scram pilot air solenoid valves resulted in degraded air pressure.
- Inadequacies in the scram recovery procedures permitted resetting the reactor protective system without prior assurance that the scram discharge volume vent and drain valves were closed.
- Lack of operator training to recognize the real condition and to take prompt corrective actions.
5. Category D: Events Involving Structural Defects or Problems in Primary Containment Buildings

5.1 Incident Summaries

See Appendix D.

5.2 Comments

In this category, five events were analyzed in the following two groups:

5.2.1 Events 372, 02, 455 and 562

In all of these incidents, mechanical problems seemed to constitute the major causes.

Corrosion was a contributor in events 455 and 562, causing a degradation of the containment integrity. Cracks were furthermore identified in 562 and 372; hence both causes were found in 562.

Human factor problems were also identified in all cases. The root causes were:

- in event 455: a design deficiency as well as construction and management organization or work planning errors (a lapse in construction management overview resulted in missing structural welds);
- in event 562: installation errors and inspection/maintenance errors (the defective installation of some anchor tendons and their missed inspection permitted their extensive corrosion);
- in event 372: procedural deficiencies and inspection/maintenance errors caused inadequate temperature control of the containment inverting system liquid nitrogen, resulting in thermal stressing of the containment vent header and a subsequent through-wall crack.

In each event a common cause hardware failure was identified (corrosion in 2 cases and freezing due to liquid nitrogen in another).

Furthermore a multiple failure was identified: namely the one reported in IRS 562, where several anchor heads, anchors and wire tendons were found broken.

In all events, there were no serious consequences, no releases, and no personnel injuries, as the problems were discovered during the planned inspection with the plant shutdown.
In the analyzed events, corrosion appeared as the most recurring problem, as it was identified in 2 out of 3 events, (455 and 562). Furthermore it seems applicable to other plants as well.

In event 455, areas affected by pitting, corrosion and missing structural welds at the ring girder flange to torus shell inner web reinforcing plates were found during the torus cleaning operation.

As mentioned earlier, the identified direct cause for incident 455 was corrosion; the predominant root causes were design deficiencies, viz chemical properties of the water and incorrect choice of the torus lead coating.

The following were the corrective actions taken:
- torus was recoated with an epoxy coating;
- better quality demineralized water was used.

In addition inspections and periodic controls to detect the status of the affected parts and the corrosion rate were intensified.

A program to evaluate and to repair the pitted areas of the torus was established as well. Weld filling was deposited on the affected zones of the torus shell after ensuring that the nonconforming material was removed completely.

It is presumed that the above corrective actions will minimize future corrosion.

This problem may be expected in other Mark I type containments.

In event 562 failures of the containment tendon field anchor heads are reported, together with the rupture of numerous wires from a 170 wire tendon.

While conducting a pre-integrated leak rate test walkdown, an alert utility worker noted grease leakage and a deformed vertical tendon anchor grease cap.

A program was initiated to visually inspect anchor heads, anchors and tendons. Laboratory tests were also performed.

The above failures were due to hydrogen stress cracking.

The hydrogen source for this phenomenon was the corrosion cells established between steel and zinc in the presence of water in the tendon assembly.

The use of high strength steel subjected to sustained tensile stresses, in the presence of particles of zinc from abrasion during tendon installation and moisture (because of grease cap ineffectiveness) contributed to the observed causes.
The following root causes were identified:

- errors during tendon and grease cap installation; and
- inspection and maintenance errors due to missed surveillance of the tendons.

The long-term planned improvements were:

- anchor heads with new coating;
- regreasing of the completed anchor head assembly; and
- periodic inspection of the coating.

The analysis highlighted the following lessons to be learned:

- correct installation of the grease caps to prevent moisture penetration;
- inspection of the installed parts;
- periodic control of environmental conditions (moisture control, visual control to verify presence of water).

Because the post-tensioned concrete containment structure integrity was based on a highly redundant system of numerous tendon elements (several hundred), it was thought that failure of one such element would not jeopardize the containment structural capability.

The identified problem should not be considered as a plant specific one, but may rather be expected wherever prestressed concrete is used in containment structures.

5.2.2 Events 113 and 120

These incidents involved problems related to vacuum degradation in the vacuum building (CANDU design). Of particular interest is event 113, where an inadequate inspection frequency led to deterioration in the rubber seal between the vacuum building perimeter and its roof, as well as in the tensioned tendons holding the seal in place. There are similarities, between this event and 562 analyzed in Section 5.2.1 above, in the areas of causes, effects, lessons learned and actions taken.

Event 120 was caused by poor procedures and an inadequate valving and instrumentation design; both of these led to a rapid rise in vacuum building pressure accompanied by a rapid fall in containment pressure and a near miss personnel injury, due to deterioration of the water seal separating the vacuum building atmosphere from the duct connecting it to the reactor buildings.
6. Category E: Events Involving Containment Coolers or Transients/Incidents that Affected or had the Potential of Affecting the Containment Integrity and/or Functions

6.1 Incident Summaries

See Appendix E.

6.2 Comments

In this category, a total of 8 events were analyzed in the following two groups:

6.2.1 Events 6.03 and 370

In event 6.03, inadequate tightening of the bearing bolts on a fan motor in reactor building closed cooling water system resulted in high vibration which led to motor damage and a break in the air vent pipe; this latter in turn caused a 200 ton water leak into the drywell, before the leak could be isolated. The operators were alerted to the incident via the control room alarms "drywell floor drain leakage high" and "reactor building closed cooling water surge tank level low."

No assessment of drywell temperature, pressure, etc. was given in the report. There are similarities between this incident and some aspects of 566, 610 and 691 described in Section 6.2.2 below.

In event 370 (CANDU), steam entered the protective relay panel of a transformer, through the floor cable penetrations, causing it to trip. A simultaneous breaker failure (to clear the transformer fault) resulted in a twelve-minute unavailability of 3 out of 4 of the containment area cooling units located in one (out of two) fuelling machine vault (a confined area, within containment, housing one fuelling machine and all the feeders and end fittings associated with one reactor face).

6.2.2 Events 176.02, 241.03, 479, 566, 610 and 691

6.2.2.1 In incident 691, the polyurethane layer on the containment drywell exterior wall was inadvertently ignited by coming into contact with hot molten material from a welding arc cutting activity on a pipe penetration, during a shutdown. Figure 2 shows a typical drywell penetration detail - note the 5 centimetre gap between the sleeve and the penetration pipe which provides a direct path to the polyurethane foam. The presence of this material in the drywell expansion gap was not considered in the licensee's fire protection reviews conducted for this plant and similar ones having Mark I containments. Furthermore, this gap is not airtight; thus the fibreglass panels installed over the polyurethane foam do not form a barrier that prevents air from coming in contact.
FIGURE 2: Typical Dresden 3 Pipe Penetration
Investigation into the extent of damage resulting from the fire revealed that in several areas inside the drywell (a few square feet in area and several feet apart), discoloured paint was visible on the drywell liner which is approximately 2.86 cm thick carbon steel. Because of the burn intensity, it is suspected that higher temperatures than expected were reached inside the drywell expansion gap at the points of discolouration. It is not known how much polyurethane foam material was consumed by the fire or how far the fire spread vertically or horizontally around the drywell. The burn lasted for 4-5 hours, so substantial burning is believed to have occurred.

It is suspected that 130000 litres were consumed to extinguish the fire.

Because of the extent of the actual damage at Dresden, corrective actions (the report does not specify what kind of actions) and plant locations where polyurethane or other combustible foam materials are installed in concealed spaces, remain under investigation by both the licensee and the NRC.

6.2.2.2 Events 566 and 610

Both events involved BWR Mark II containments, and in both cases a drywell high pressure/temperature transient occurred. In incident 610, over an eight-hour period, the containment temperature exceeded the Tech. Spec. limit (57.2°C) by 17.8°C. Drywell pressure was kept below the isolation pressure of 1.69 psig by venting. A Sargent and Lundy analysis showed that this transient had no significant impact on equipment. Investigations showed that design, maintenance and installation errors had led to fatigue failure of the 18 circulation water pump discharge valve gear operator mounting bolts; this in turn caused a rapid valve closure creating a pressure spike which ruptured a rubber expansion joint between the valve and the pump. The resulting greenhouse flooding prompted a shutdown and caused, among other things, a loss of cooling to the station air compressors and the primary containment air coolers, the latter being the reason for the drywell high pressure/temperature transient. Reactor cooldown was effected according to procedures.

In incident 566, extreme cold weather had resulted in the use of only one reactor building closed cooling water (RBCCW) heat exchanger due to concern for recirculation pump seal embrittlement. When the unit tripped due to deteriorating condenser vacuum (caused by a failed turbine to condenser rubber expansion joint), an increased load on the RBCCW system (relief valves, vessel letdown to the condenser, etc.) resulted in a decreased efficiency of the drywell coolers and a subsequent high primary containment pressure/temperature (the values were not specified). There are no tech. spec. requirements related to the RBCCW system since it is not a safety-grade system.

6.2.2.3 Event 176.02

The control room operator noticed that control power had been lost to the "B" train hydrogen recombiner. This was caused by a short circuit due to inadvertent actuation of the fire protection deluge system while welding in the electrical penetration area.
The following are the surveillance requirements to demonstrate the hydrogen recombiner system operability:

i) Once per 6 months a functional test to verify the recombiner can reach the adequate heater sheath temperature, later to increase the power setting to maximum power for 2 minutes and verify that the power meter reads at least 60 Kw.

ii) Once per 18 months by making a channel calibration, a visual examination and verify the integrity of all heater electrical circuits.

In addition, there is a limit of 30 days to restore the operability of a recombiner.

This event resulted in the improvement of the maintenance personnel training and the test procedures. Maintenance personnel were advised concerning proper ventilation for welding in enclosed areas and burning permits are under revision to verify that ventilation requirements are met.

Seven out of nine Spanish LWR have hydrogen recombiners. Two reportable events have been found related to them. The first happened in 1985 and shows the failure of a thermocouple, which was replaced. The second happened in 1987 and shows the failure of a power meter, which was replaced. Note that these events do not imply the inoperability of the recombiner contrary to the Trojan event.

6.2.2.4 Event 479

Incorrect wiring inside the control cabinet resulted in a burnt pump motor (later replaced) in the containment hydrogen monitoring system. Inadequate mechanical design of the pump resulted in diaphragm failure, which was not detected during routine analyzer calibration (see figure 3). The two-stage diaphragm pump was later replaced by two redundant one-stage diaphragm pumps; a new flow meter was also installed in the suction side.

The number of trains composing the H2 monitoring system is not specified in the report. Otherwise, the reliability of the system has been enhanced by adding two redundant pumps, a flowmeter and a better design.

The STS for BWRs with two analyzers requires one channel calibration per 92 days. The STS for PWRs with two analyzers requires a channel check once per hour, an analog channel operational test once per 31 days and a channel calibration per 92 days.
FIGURE 3
Related Events in Spanish Reactors

The only Spanish BWR-4 with a Mark I containment has one train for the H2 monitoring system. Once per week it is checked to ensure that the H2 concentration is below 4%.

Three PWRs, 1000 Mwe each, have two trains. Fig 4 shows the flow diagram.

Two PWRs, 1000 Mwe each, have one train. This train is tested once per three months. The only reportable event that occurred in one of these units was in June 1982. One isolation valve failed closed resulting in the impossibility to take H2 samples in one point of the containment. The other points of the system were operable.

One PWR, 160 Mwe, has one train; in the case of failure, atmospheric samples can be taken by other means.

6.2.2.5 Event 241

Frozen condensate in the containment atmospheric control system of the nitrogen inerting system, prevented utilizing the vaporizer to perform nitrogen makeup to the drywell.

The measured oxygen concentration in the drywell exceeded the plant's technical specification limit of "less than 4 percent". The event lasted for 4.3 hours, with the highest measure of oxygen at 4 percent.

During the investigation of the event, it was noted that the required line trace heating and insulation lagging on the frozen line section were missing - no explanation could be found.

The containment atmospheric control system is not safety related, therefore Tech. Spec. have no provisions for periodic tests. The oxygen concentration should be measured once per week at least according to STS provided that at least one train of the oxygen analyzer is operable. Otherwise atmospheric samples should be taken every 4 hours.

Comparison with Spanish designs - related events in Spain

No similar events have occurred in Spain. The report gives no details on the system, which normally would have no function in the event of an accident. In the Spanish BWR Mark I containments, such a system has two main parts: one dedicated to the shock inerting with large diameter pipes (typically 14" to 20"), and the other dedicated to nitrogen make-up, with 2" piping.
ASCO

2 independent lines and 2 analyzers

2 pumps: capacity 0.6 m³/h

FIGURE 4: FLOW DIAGRAM IN ASCO NPP
GENERIC STUDY ON THE
EVENTS REPORTED THROUGH THE NEA-IRS AND INVOLVING
A LOSS OF CONTAINMENT FUNCTIONS

Prepared by a
CSNI/PWG1 Group of Experts

PART 2: APPENDICES

Paris, July 1988
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<th>UNIT STATUS (% F.P.)</th>
<th>CAUSES</th>
<th>CONSEQUENCES</th>
<th>LESSONS LEARNED AND/OR ACTIONS TAKEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Several operator errors result in equipment unavailability.</td>
<td>Trojan, Mar. 82, PWR</td>
<td>292</td>
<td>Following routine testing, train B of the spray pumps was not returned to &quot;available&quot; status.</td>
<td>The system would have failed to satisfy the design intent due to unavailability of Train B of the containment spray system.</td>
<td>Institute or improve existing administrative controls on safety system restoration. Develop or improve operator aids. Improve operator training emphasizing the importance of following procedures.</td>
</tr>
<tr>
<td>Temporary startup filters remained in the spray pumps suction for 3 years.</td>
<td>Summer Aug. 85, PWR</td>
<td>684</td>
<td>Organisational and/or planning fault - ambiguous technical drawings contributed failure to remove spray temporary filters.</td>
<td>In the long term, a spray system loss could occur since the filters present were not designed for post-accident conditions.</td>
<td>Remove the filters - Identify other similar situations. Send NRC letter alerting other stations. Emphasize with plant staff the importance of correctly putting safety systems into operation.</td>
</tr>
<tr>
<td>Inadvertent operation of containment spray and water accumulation in containment.</td>
<td>Sequoyah 1, Feb. 81, PWR</td>
<td>27</td>
<td>Operator error apparently due to mis-communications. Erroneous actuation.</td>
<td>Accumulation of 385000 litres of water in containment. Eight workers required some decontamination. A small external release occurred.</td>
<td>Improve communications between the plant staff.</td>
</tr>
<tr>
<td>EVENT HIGHLIGHTS</td>
<td>IDENTIFICATION</td>
<td>UNIT STATUS (% F.P.)</td>
<td>CAUSES</td>
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<tr>
<td>Containment spray system inoperability for more than one year.</td>
<td>287, U.S.</td>
<td>Ref*</td>
<td>Containment spray header isolation valves were found locked closed. An undocumented design modification contributed as well as an inadequate procedure for valve position verification.</td>
<td>Potential for off site doses being above limits (10 CFR 100) in the event of a LOCA or main steam line break with inoperable spray system.</td>
<td>- Lock valves open. - Perform analysis of potential event consequences. - Improve procedures for valve position verification. - Emphasize importance of putting safety systems correctly into operation.</td>
</tr>
<tr>
<td>Inadvertent containment spray operation during a test.</td>
<td>75, France, Bugey 2, Apr. 81, PWR</td>
<td>100</td>
<td>Deficiency in containment spray chain test procedure.</td>
<td>Accumulation of 16,000 litres of water in the reactor building basement.</td>
<td>- Correct test procedure. - Emphasize the importance of following correct procedure.</td>
</tr>
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</table>

* Refuelling outage.
<table>
<thead>
<tr>
<th>EVENT HIGHLIGHTS</th>
<th>IDENTIFICATION</th>
<th>UNIT STATUS (%)</th>
<th>CAUSES</th>
<th>CONSEQUENCES</th>
<th>LESSONS LEARNED AND/OR ACTIONS TAKEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Containment spray system</td>
<td>416.01</td>
<td>U.S.,</td>
<td>Containment spray header discharge</td>
<td>Potential for off site doses being above limits</td>
<td>- Lock valves open.</td>
</tr>
<tr>
<td>inoperability for more than one</td>
<td>Ind. Pt. 2,</td>
<td>100</td>
<td>valves found locked closed.</td>
<td>(10 CFR 100) in the event of a LOCA or main steam line break with inoperable spray system.</td>
<td>- Review position indication for safety-related valves.</td>
</tr>
<tr>
<td>month (both trains unavailable simultaneously).</td>
<td>Nov. 83,</td>
<td>PWR</td>
<td>Personnel carelessness in following procedures.</td>
<td></td>
<td>- Emphasize importance of putting safety system correctly into operation.</td>
</tr>
<tr>
<td>Five hour simultaneous unavailability of both containment spray system trains.</td>
<td>416.02</td>
<td>S/D</td>
<td>The B train was unavailable due to loss of indication for the DC power supply to the B train containment pressure control system transmitter.</td>
<td>A five hour violation of the action statement of Tech. Spec. Section 3.6.2 Potential consequences were failure to meet the release limits in the event of a LOCA or a main steam line break.</td>
<td>- Replace power supply and repair deficient pump.</td>
</tr>
<tr>
<td></td>
<td>U.S.,</td>
<td></td>
<td></td>
<td></td>
<td>- Improve procedures related to declaration of inoperability of all support systems.</td>
</tr>
<tr>
<td></td>
<td>Sept. 83,</td>
<td>PWR</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>13 day inoperability of containment spray system.</td>
<td>446.</td>
<td>100</td>
<td>Containment spray pump manual discharge isolation valves were found locked closed due to their omission from the checklist used, which had been inadequately reviewed because of deficient administrative controls.</td>
<td>Potential for off site doses exceeding limits in case of a LOCA or main steam line break with inoperable spray system.</td>
<td>- Revise procedures to ensure proper safety system valve availability.</td>
</tr>
<tr>
<td></td>
<td>U.S.,</td>
<td></td>
<td></td>
<td></td>
<td>- Improve administrative controls of procedure writing and modification.</td>
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<tr>
<td></td>
<td>San On 3,</td>
<td></td>
<td></td>
<td></td>
<td>- Improve training programme for proper system alignment.</td>
</tr>
<tr>
<td></td>
<td>Mar. 84,</td>
<td>PWR</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td>EVENT HIGHLIGHTS</td>
<td>IDENTIFICATION</td>
<td>UNIT STATUS (F.P.)</td>
<td>CAUSES</td>
<td>CONSEQUENCES</td>
<td>LESSONS LEARNED AND/OR ACTIONS TAKEN</td>
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<td>------------------------------------------------------------------------</td>
<td>----------------------------------------------------------------------------</td>
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<tr>
<td>Inadvertent containment spray</td>
<td>437.05</td>
<td>S/D</td>
<td>Following a shift change, the valve lineup check was done on unit 2</td>
<td>Outage extension due to the accumulation of about 14500 litres of water</td>
<td>- Assess containment building equipment for damages.</td>
</tr>
<tr>
<td>operation and water accumulation</td>
<td>U.S.</td>
<td>Cal. Cliffs 1,</td>
<td>instead of 1, leading to the containment spray header manual</td>
<td>in containment.</td>
<td>- Change procedure to require independent valve verification.</td>
</tr>
<tr>
<td>in containment.</td>
<td>Apr. 87</td>
<td>PWR.</td>
<td>isolation valve being left open.</td>
<td></td>
<td>- Make all operators aware of this event.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>- Evaluate colour coding feasibility for different units.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>- Assign different supervisors to watch stations of opposite units.</td>
</tr>
</tbody>
</table>
## APPENDIX B

### Category B Event Summaries

<table>
<thead>
<tr>
<th>EVENT HIGHLIGHTS</th>
<th>IDENTIFICATION</th>
<th>UNIT STATUS (% F.P.)</th>
<th>CAUSES</th>
<th>CONSEQUENCES</th>
<th>LESSONS LEARNED AND/OR ACTIONS TAKEN</th>
</tr>
</thead>
</table>
| MSIV unexpected closure during test.     | 6.02, Japan.   | 94                   | Insufficient maintenance led to accumulation of foreign material from the pneumatic control system inside the MSIV operator. | High main steam flow and subsequent reactor scram. | - Replace pilot valve assemblies with a different type.  
  - Ensure that pneumatic control systems do not cause MSIV problems. |
| Failures of Rockwell MSIV.               | 163.01-163.04, U.S. | N/A                  | Vibrations possibly caused by steam flow instability led to disengaging of the main valve disk from the steam. | Sudden main steam line obstruction resulting in void collapse, a power spike and a reactor scram. | - Repair valves and modify their design.  
  - Apply manufacturer's recommendations.  
  - Emphasize the importance of proper design review.  
  - Emphasize the potential of losing both MSIVs on the same line. |
| Cracked speed control cylinders on two MSIVs. | 162, U.S. | 5/D                  | Procedural deficiency and operator error during previous startup (11 days earlier, the inboard MSIVs were opened before the outboard ones). | MSIVs unavailability during 11 days of operation. | - Repair MSIVs.  
  - Inspect and test leak rate of inboard MSIVs.  
  - Revise procedures and operating instructions.  
  - Improve operator training. |
<table>
<thead>
<tr>
<th>EVENT HIGHLIGHTS</th>
<th>IDENTIFICATION</th>
<th>UNIT STATUS (% F.P.)</th>
<th>CAUSES</th>
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<th>LESSONS LEARNED AND/OR ACTIONS TAKEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inadvertent decrease of main steam line pressure causing MSIV closure.</td>
<td>190, Japan, Fuku, Daichi 1, Jul. 82, BWR.</td>
<td>97</td>
<td>Insufficient maintenance caused a malfunction of the electrical pressure regulating servo valve.</td>
<td>Reactor scram, turbine trip, and a small external release.</td>
<td>Replace valve and test. Check input/output characteristics of turbine servo valve. Replace servo valves during periodic inspections. Emphasize potential of MSIV spurious closure due to pressure regulator failure.</td>
</tr>
<tr>
<td>Improper use of lubricant on isolation valve seats.</td>
<td>336, U.S., Dres. 2, Apr. 83, BWR.</td>
<td></td>
<td>In violation of the manufacturer's recommendations, grease was used in the isolation valve re-assembly. Maintenance deficiencies also contributed.</td>
<td>Local leak rate test results were outside acceptable limits.</td>
<td>Inspect and clean valves and repeat leak tests. Develop standard isolation valve maintenance procedures. Improve maintenance training programme. Develop a repair manual.</td>
</tr>
<tr>
<td>Leak tightness degradation of primary containment.</td>
<td>522, Italy, Caorso, 83-84, BWR</td>
<td></td>
<td>The accumulation of carbon steel piping corrosion products inside valve seats caused deterioration of the teflon gaskets for butterfly isolation valves. Contributors were poor water chemistry and frequent use of some of the valves. Problem identification was delayed because of several procedural and maintenance deficiencies.</td>
<td>Degradation of containment integrity and leak tightness.</td>
<td>Repair/replace valves as necessary. Apply special coating to inside of some piping. Revise test procedures. Emphasize the compatibility of piping materials, fluids &amp; valve types. Emphasize the importance of test procedure results for the &quot;as found&quot; &amp; &quot;as left&quot; (before &amp; after maintenance) states.</td>
</tr>
<tr>
<td>EVENT HIGHLIGHTS</td>
<td>IDENTIFICATION</td>
<td>UNIT STATUS</td>
<td>CAUSES</td>
<td>CONSEQUENCES</td>
<td>LESSONS LEARNED AND/OR ACTIONS TAKEN</td>
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</tr>
<tr>
<td>Slow MSIV closure times due to</td>
<td>376,</td>
<td>S/D</td>
<td>Erroneous choice of materials</td>
<td>Degradation of containment isolation function.</td>
<td>Replace ring with another of proper material.</td>
</tr>
<tr>
<td>inadequate QA on replacement</td>
<td>U.S.,</td>
<td></td>
<td>caused heavy stem galling to extend to the</td>
<td></td>
<td>Improve purchase documentation on replacement parts.</td>
</tr>
<tr>
<td>parts.</td>
<td>Cal. Cl. 162,</td>
<td></td>
<td>junk ring inside the packing gland.</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Jan. 81-</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Oct. 82,</td>
<td></td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>PWR.</td>
<td></td>
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<td></td>
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</tr>
<tr>
<td>Failure of 3 MSIVs to close</td>
<td>389,</td>
<td>S/D</td>
<td>Two upstream valves failed due to valve</td>
<td>Potential failure of containment isolation in case of an accident.</td>
<td>Notification by manufacturer to customers of problems.</td>
</tr>
<tr>
<td>under no flow conditions.</td>
<td>U.S.,</td>
<td></td>
<td>packing binding. Failure of all four disc</td>
<td></td>
<td>Implement appropriate design modifications.</td>
</tr>
<tr>
<td></td>
<td>Farley 2,</td>
<td></td>
<td>fasteners caused separation of disk from disk</td>
<td></td>
<td>For packing problems:</td>
</tr>
<tr>
<td></td>
<td>Sep. 83,</td>
<td></td>
<td>arm on the downstream valves. Design</td>
<td></td>
<td># inspect/replace MSIV packing as required,</td>
</tr>
<tr>
<td></td>
<td>PWR.</td>
<td></td>
<td>deficiencies contributed to all failures.</td>
<td></td>
<td># test MSIV movement during heatup after packing replacement, and</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td># repeat above 2 steps until problem is solved.</td>
</tr>
</tbody>
</table>

<p>| MSIV unexpected closure during a test.  | 398,           | 99          | A combination of equipment failure, installation and/or construction errors resulted in a wiring anomaly in the MSIV circuitry, which caused failure of a relay driver card optical isolation in a steam and feedwater rupture control system channel. | A reactor trip followed by a six day outage for repairs and event evaluation. | Replace faulty relay driver card. |
|                                         | U.S.,          |             |                                               |                                                  | Send NRC letter concerning the evaluation of the equipment failures on the plant. |
|                                         | Davis          |             |                                               |                                                  |                                        |
|                                         | Besse 1,       |             |                                               |                                                  |                                        |
|                                         | Mar. 84,       |             |                                               |                                                  |                                        |
|                                         | PWR.           |             |                                               |                                                  |                                        |</p>
<table>
<thead>
<tr>
<th>EVENT HIGHLIGHTS</th>
<th>IDENTIFICATION</th>
<th>UNIT STATUS (% F.P)</th>
<th>CAUSES</th>
<th>CONSEQUENCES</th>
<th>LESSONS LEARNED AND/OR ACTIONS TAKEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor scram due to MSIV closure.</td>
<td>406,</td>
<td>100</td>
<td>An amplifier failure in the MSIV control circuit caused its closure.</td>
<td>Reactor trip.</td>
<td>This constituted a &quot;safe&quot; failure, hence no adverse effect on containment functions.</td>
</tr>
<tr>
<td>Loss of station instrument air pressure causing MSIV closure.</td>
<td>616,</td>
<td>100</td>
<td>Loss of instrument air pressure.</td>
<td>Reactor trip followed by a safety injection.</td>
<td>The MSIV operated as designed with no effect on containment performance.</td>
</tr>
<tr>
<td>Reactor trip due to MSIV closure.</td>
<td>673,</td>
<td>53</td>
<td>Vibration in main steam piping was picked up by the detection lines which resulted in MSIV closure.</td>
<td>Reactor trip.</td>
<td>A design change in the piping supports was effected. This event had no effect on containment performance/integrity.</td>
</tr>
<tr>
<td>Failures of MSIVs to close.</td>
<td>662.02,</td>
<td>5/D</td>
<td>Contributing causes were excessive dirt in the actuator oil, excessive oil pressure due to poor maintenance, and operation above the temperature limits recommended by the manufacturer.</td>
<td>Recurrent degradation of the isolation function and of the containment leak tightness.</td>
<td>- Clean actuators. - Change actuator oil. - Reduce oil pressure. - Emphasise importance of proper maintenance and operating conditions for safety related components.</td>
</tr>
<tr>
<td>EVENT</td>
<td>IDENTIFICATION</td>
<td>UNIT STATUS</td>
<td>CAUSES</td>
<td>CONSEQUENCES</td>
<td>LESSONS LEARNED AND/OR ACTIONS TAKEN</td>
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<tr>
<td>Containment spray header outboard isolation valve failure to close during a leak tightness test.</td>
<td>743, U.S., Limerick 1, Dec. 85, BWR.</td>
<td>S/D</td>
<td>The root cause was unknown at the time the report was written. A contributing factor is hydraulic lock-up due to excessive grease in the spring pack area.</td>
<td>Degradation of the isolation function and of the containment leak tightness.</td>
<td>- Remove the excess grease. - Emphasise the hydraulic lock-up phenomenon as a potential common cause failure for motor operated valves.</td>
</tr>
<tr>
<td>Drywell to suppression chamber vacuum breaker failure to close during test.</td>
<td>285, US., Quad Cities 2, Oct. 82, BWR.</td>
<td>-</td>
<td>Binding of the stainless steel packing and stuffing box bushing on the valve shaft; Possible contributors are poor lubrication and/or poor design.</td>
<td>Potential for a break inside the drywell with no quenching action due to steam by-passing the suppression pool; excessive dynamic loads would result on the suppression chambers and associated piping.</td>
<td>- Repair valve. - Check similar valves. - Emphasise the importance of proper maintenance on safety related components.</td>
</tr>
<tr>
<td>Outer isolation valves in two steam lines left closed during a startup.</td>
<td>414.02, Finland, TVO2, Aug. 83, BWR.</td>
<td>-</td>
<td>Operator error in following procedures.</td>
<td>Reactor trip on high pressure.</td>
<td>Emphasise to the operators the importance of following procedures.</td>
</tr>
<tr>
<td>Inboard MSIV drifting closed during a test.</td>
<td>559.01, US., Hatch 2, Jan. 85, BWR.</td>
<td>-</td>
<td>Continuous MSIV actuation (it failed to close within the required time limit) resulted in isolation of the drywell pneumatic system supply valves which feed the MSIVs-procedural inadequacy.</td>
<td>Reactor trip (forced outage).</td>
<td>Add in the procedure a 2 minute waiting period between successive MSIV operations to prevent a high flow isolation of the drywell pneumatic system.</td>
</tr>
<tr>
<td>EVENT HIGHLIGHTS</td>
<td>IDENTIFICATION</td>
<td>UNIT STATUS (%) F.P</td>
<td>CAUSES</td>
<td>CONSEQUENCES</td>
<td>LESSONS LEARNED AND/OR ACTIONS TAKEN</td>
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</tr>
<tr>
<td>Failure of 3 MSIVs to stay closed as required after a reactor trip.</td>
<td>559.02, U.S.</td>
<td>Grand Gulf -1, Feb. 85, BHR</td>
<td>Failure of the dual solenoid valve in the automatic actuation circuit. The root cause was not mentioned.</td>
<td>Degradation of containment isolation function.</td>
<td>Replace MSIV dual solenoids. Implement an NRC approved schedule for increased exercising of the MSIVs prior to startup.</td>
</tr>
<tr>
<td>Inoperability of both reactor building to torus vacuum breakers (protecting negative pressure relative to drywell).</td>
<td>738, U.S.</td>
<td>Hope Creek, Aug. 96, BHR</td>
<td>Improper tubing of the breaker due to a design drawing deficiency. Communications breakdown between the KSSS vendor and the architect/ engineer/builder.</td>
<td>Degradation of containment integrity in case of a LOCA inside containment.</td>
<td>Implement design changes. Review similar applications elsewhere. Verify lineup of all reactor building instrument valves. Verify all temporary modifications for safety related equipment. Identify on a schedule the position of safety related instrument valves. Review event with licensed operators and include it in training program.</td>
</tr>
<tr>
<td>MSIV failure to close during a test.</td>
<td>82.05 S/D</td>
<td>Zion 1, Jan. 81, PWR</td>
<td>Concurrent and independent failure of two redundant BC solenoid valves actuating the MSIV. One failed due to oil impurities settling on the valve surface; the other seized due to a short circuit.</td>
<td>Degradation of containment isolation function in case of a main steam line break downstream of the MSIV's as assumed in the PSAK.</td>
<td>Replace solenoid valves with spares (one being of a different type). Replace other solenoid valves experiencing oil impurity problems with ones of a different design.</td>
</tr>
<tr>
<td>EVENT HIGHLIGHTS</td>
<td>IDENTIFICATION (F.P.)</td>
<td>UNIT STATUS</td>
<td>CAUSES</td>
<td>CONSEQUENCES</td>
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<tr>
<td>Unexpected MSIV partial closure.</td>
<td>361, France, Bugey 3, Jan. 83, PWR.</td>
<td>82.5</td>
<td>Broken MSIV control valve air supply connection.</td>
<td>Coincident signals of &quot;high steam flow&quot; &amp; &quot;low steam pressure&quot; resulting in a safety injection.</td>
<td>- Check similar components.  - Emphasize the importance of proper design and operating conditions for support systems (e.g. compressed air)</td>
</tr>
<tr>
<td>Open door between wet well and dry well in pressure suppression containment.</td>
<td>360, Sweden, Osk. -2, Jul. 82, BWR.</td>
<td></td>
<td>Inadequate design - locking mechanism too stiff and weak for application all contributed to the door being improperly locked.</td>
<td>The pressure suppression function was degraded to the extent that the containment structure could have been jeopardized in the event of a LOCA.</td>
<td>- Change the lock mechanism design.  - Install &quot;position&quot; mark on the handle.  - Introduce indicators in the control room for door and handle positions.</td>
</tr>
</tbody>
</table>
## APPENDIX C
### Category C Event Summaries

<table>
<thead>
<tr>
<th>EVENT HIGHLIGHTS</th>
<th>IDENTIFICATION</th>
<th>UNIT STATUS (% F.P.)</th>
<th>CAUSES</th>
<th>CONSEQUENCES</th>
<th>LESSONS LEARNED AND/OR ACTIONS TAKEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Leakage of reactor coolant outside containment after a safety injection.</td>
<td>36, France, Begey 5, Jan. 81, PWR.</td>
<td>S/D</td>
<td>Operator error in actuating a manual valve allowing make-up (as well as safety injection) pump to pump water into the recirculation loop whose tank overflowed.</td>
<td>Contamination of the auxiliary building due to the containment bypass.</td>
<td>Emphasise with operators the importance of following written procedures.</td>
</tr>
<tr>
<td>Leakage of reactor coolant into the fuel building pond.</td>
<td>69, France, Laurent B-2, Dec. 80, PWR</td>
<td>S/D</td>
<td>Valves isolating the NHR system from the irradiated fuel pond cooling system were inadvertently left open. The pond cooling system was overpressurised and one valve leaked. Valve lineup errors also contributed.</td>
<td>Discharge of 6000 litres of primary water into the fuel building pond. The pond cooling system was damaged. There was a potential for liquid radioactive release outside containment and loss of fuel pond cooling.</td>
<td>- Provide control room indications of actual positions of the isolation valves involved. - Install an alarm on pond cooling system overpressure. - Define in the tech. specs. the connection between the NHR and the pond cooling systems.</td>
</tr>
<tr>
<td>Loss of zero meter integrity for 45 minutes.</td>
<td>224.02, France, Tricastin -1, Aug. 82, PWR.</td>
<td>100</td>
<td>A plug ruptured in a pressure reducing valve in a compressed air line feeding the hatch door seals. Possible operator delay in diagnosing the failure.</td>
<td>A 45 minute loss of containment integrity occurred at the zero meter hatch. 1000 cubic meters of air were released to the environment - potential for a radioactive release.</td>
<td>- Modify the design of the compressed air system feeding the hatch door seals. - Improve the alarms related to hatch leak tightness.</td>
</tr>
<tr>
<td>EVENT HIGHLIGHTS</td>
<td>IDENTIFICATION</td>
<td>UNIT STATUS (% F.P.)</td>
<td>CAUSES</td>
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<tr>
<td>Primary circuit leak in the outer reactor containment building.</td>
<td>276, France, Gravelines 4, Aug. 82, PWR.</td>
<td>45</td>
<td>Bad closure of an inboard KSS valve, resulted in an outboard sight glass leaking. The double isolation criterion was not followed.</td>
<td>4000 litres of water were discharged into the auxiliary building. Had the activity of that water been higher, a high external release would have resulted.</td>
<td>- Ensure the use of double isolation upstream prior to any repair on high energy piping. - Use appropriate limit switches for position indication in control room. - Improve the safety of flow indicators fitted with sight glasses.</td>
</tr>
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</table>

**508.01, Tricastin 1, Aug. 82, PWR.**

**THIS EVENT IS SIMILAR TO EVENT NO 224.02 (IN THIS CATEGORY)**

| A two minute loss of the 8 metre hatch integrity. | 508.02, France, Bugey -2, Apr. 83, PWR. | 100 | A filter ruptured in a compressed air system line feeding the hatch door seals. | A 2-minute loss of containment integrity at the 8 metre hatch. | - Modify the design of the compressed air system feeding the hatch door seals. - Improve the alarms related to hatch leak tightness. |

<p>| Loss of zero meter hatch integrity during a test. | 508.03, France, Blayais -1, Dec. 83, PWR. | 100 | The cause is unknown since the incident was not reproducible. | A 4.5 hour loss of containment integrity at the zero hatch. | Revise test procedures. |</p>
<table>
<thead>
<tr>
<th>EVENT HIGHLIGHTS</th>
<th>IDENTIFICATION</th>
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</tr>
</thead>
<tbody>
<tr>
<td>Degradation of the 8 metre hatch at two units.</td>
<td>508.04, Tricastin U2.</td>
<td>U1, S/D, Apr. 84, PWR.</td>
<td>A compressed air line plug ruptured in unit 1 (modifications following event 508.01 had not been implemented yet). The replacement plug was taken from unit 2.</td>
<td>Degradation of containment integrity at the 8 metre hatch for nearly one hour.</td>
<td>Replace the plug. - Implement the modifications from the event of Aug. 82 ($508.01).</td>
</tr>
<tr>
<td>Loss of zero meter hatch integrity for three hours.</td>
<td>508.05, Chinon 82, May 84, PWR.</td>
<td>100</td>
<td>Several factors contributed: - Breakdown of the compressed air system feeding the door seals. - Wrong alarm setpoint. - Poor organization of modifications. - Inadequate communications. - Failure to inform the radiation protection safety engineer.</td>
<td>A three hour loss of containment integrity at the zero meter hatch. 1200 cubic meters of air were released to the environment with gas and halogen activities of 7.9x10^-6 TBq and 8.6x10^-6 GBq respectively.</td>
<td>Improve operating QA deficiencies. - Improve alarms on loss of hatch integrity.</td>
</tr>
<tr>
<td>Loss of zero meter hatch integrity.</td>
<td>508.06, Gravelines 3, Sep. 84, PWR.</td>
<td>100</td>
<td>Failure to follow the procedures applicable in case of a &quot;On hatch defect&quot; alarm resulting in opening both inner and outer doors and closing the seal isolation valves (preventing double isolation). Design defects in the alarm logic also contributed.</td>
<td>Loss of containment integrity at the zero meter hatch level.</td>
<td>Improve personnel training on the third barrier integrity. - Modify the alarm instruction sheets to prevent spurious seal isolation. - Review the door opening logic. - Modify the &quot;hatch defect&quot; alarms to include erroneous lineup of the seal isolation valves.</td>
</tr>
<tr>
<td>EVENT HIGHLIGHTS</td>
<td>IDENTIFICATION</td>
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<tr>
<td>All the events described in report 508.</td>
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<td>- Study the feasibility of using solid seals.</td>
</tr>
<tr>
<td>Uncontrolled leak of reactor coolant outside primary containment via the scram discharge system and the drainage system.</td>
<td>375, U.S.</td>
<td>S/D</td>
<td>The following were contributors: Rockwell MSIV failure due to deficiency &amp; missed implementation of recommendations by Rockwell (see also 163.01-04 in Category B). Mechanical blockage of a safety valve causing primary building pressure increase. Loose connection between scram discharge volume drain and discharge valves and operator allowing loss of primary water. Defective drywell load shedding logic. Missing pipecap in RCIC room drainage system.</td>
<td>Overall containment bypass and sustained uncontrolled leakage of primary water. The reactor tripped. Pronounced level and pressure transients occurred. Drywell high pressure and defective load shedding logic led to drywell chiller isolation. Primary water flashing from the RCIC room drainage led to RCIC isolation.</td>
<td>- Implement the required MSIV design changes. - Improve SDV isolation valves surveillance procedures. - Upgrade administrative controls over drain hub caps. - Improve personnel training on bypass signals. - Improve SDV hardware arrangements. - Review emergency procedures. - Study SD system as potential path for reactor coolant leakage. - Emphasize the importance of load shedding logic flexibility. - Emphasize the importance of correct equipment restoration following maintenance (see also reports 292 &amp; 448).</td>
</tr>
</tbody>
</table>
## Uncontrolled leakage of reactor coolant outside the primary containment via the scram discharge system and the reactor building equipment drain tank.

- **Event:** Uncontrolled leakage of reactor coolant outside the primary containment via the scram discharge system and the reactor building equipment drain tank.
- **Identification:** 572, U.S., Oyster Creek, Jan. 85, BWR.
- **Unit Status:** 100
- **Causes:** Failure of the SDV drain and discharge valves to fully isolate due to calibration and installation errors. Defective scram logic also contributed.
- **Consequences:** Overall containment bypass, Reactor trip, Significant level and pressure transients, Flashing of primary water from the equipment drain tank actuated the fire deluge system causing flooding & potential loss of safety systems.
- **Lessons Learned:** Repair SDV valves, Review such events with operators and maintainers, Improve installation QA, Study the SD system as a potential path for reactor coolant leakage.

## Loss of one airlock integrity for 45 seconds.

- **Event:** Loss of one airlock integrity for 45 seconds.
- **Identification:** 111, Canada, Bruce 2, Aug. 81, CANDU.
- **Unit Status:** 88
- **Causes:** Incorrect airlock modifications during commissioning allowed the outer door seals to deflate while the inner door was open.
- **Consequences:** Loss of the 4 unit containment integrity at that airlock for 45 seconds.
- **Lessons Learned:** Modify the airlock alarm & repair interlock logic, Review the adequacy of personnel training, airlock procedure, and the containment breach detection system.

## Loss of one airlock integrity for 33.5 minutes.

- **Event:** Loss of one airlock integrity for 33.5 minutes.
- **Identification:** 115, Canada, Bruce 2, Sep. 81, CANDU.
- **Unit Status:** 5/D
- **Causes:** Operator error in closing the isolating valve to the door seals compounded by poor alarm system design.
- **Consequences:** Loss of the 4 unit containment integrity at that airlock for 33.5 minutes.
- **Lessons Learned:** SEE EVENT 111 ABOVE
<table>
<thead>
<tr>
<th>EVENT HIGHLIGHTS</th>
<th>IDENTIFICATION</th>
<th>UNIT STATUS (F.P.)</th>
<th>CAUSES</th>
<th>CONSEQUENCES</th>
<th>LESSONS LEARNED AND/OR ACTIONS TAKEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>4 minute loss of airlock integrity during a test.</td>
<td>306, Canada, Bruce 2, Jan. 83, CANDU.</td>
<td>0.1</td>
<td>Four compressed air valves, which had been replaced a day earlier, developed leaks during a test resulting in a deterioration of the air supply to the airlock door seals.</td>
<td>A 4-minute containment impairment.</td>
<td>Replace faulty valves. Change the procedure and test pressure. Review adequacy of personnel training, airlock procedures and containment breach detection and alarms.</td>
</tr>
<tr>
<td>Damage to equipment passage doors due to imbalance in the reactor building ventilation.</td>
<td>551, Japan, Tokai, Feb. 85, BWR.</td>
<td>S/D</td>
<td>Poor HVAC starting procedures resulted in improper HVAC switching and a subsequent negative pressure in the reactor building of about 400 mm H2O.</td>
<td>Loss of containment integrity and damage to equipment passage doors.</td>
<td>Repair and/or replace failed doors and affected equipment. Review the HVAC starting procedures.</td>
</tr>
<tr>
<td>Failure of reactor building airlock door seals.</td>
<td>626, Canada, Pickering 5, Dec. 85, CANDU.</td>
<td>100</td>
<td>An air supply valve for the door seals was inadvertently left in the partially open position. A contributor was a design deficiency in the power supply for the seal status annunciator.</td>
<td>The containment was impaired.</td>
<td>Review the incident with the operating staff. Increase the airlock surveillance. Emphasise the importance of the seal status alarm signal.</td>
</tr>
<tr>
<td>EVENT HIGHLIGHTS</td>
<td>IDENTIFICATION</td>
<td>UNIT STATUS (%) F.P.</td>
<td>CAUSES</td>
<td>CONSEQUENCES</td>
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<tr>
<td>Uncontrolled reactor coolant leakage outside the primary containment via the scram discharge system and the reactor building drain tank. (see also 375 &amp; 572 this category).</td>
<td>688, U.S., Dresden 3, Sep. 85, BWR.</td>
<td>83</td>
<td>Stuck contacts in reactor protective system channel B prevented resetting of a trip condition and caused a reduction in air pressure to the actuators of the SDV drain &amp; discharge valves causing their opening.</td>
<td>The containment was partially by-passed in that a sustained loss of hot pressurized reactor coolant occurred outside the primary containment for 23 minutes.</td>
<td>- Improve the scram recovery procedures. - Improve training to increase operator awareness of actual plant conditions. - Study the SDV system as a potential path for reactor coolant leakage.</td>
</tr>
</tbody>
</table>
### APPENDIX D

**Category D Event Summaries**

<table>
<thead>
<tr>
<th>EVENT HIGHLIGHTS</th>
<th>IDENTIFICATION</th>
<th>UNIT STATUS</th>
<th>CAUSES</th>
<th>CONSEQUENCES</th>
<th>LESSONS LEARNED AND/OR ACTIONS TAKEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Through-wall crack in the torus vent header.</td>
<td>372.02, U.S., Hatch 2, Feb. 84, BWR.</td>
<td>S/D</td>
<td>Brittle fracture caused by low temperature nitrogen, from the containment inerting system, impinging on the vent header. The root causes are believed to be a combination of containment inerting component failures, poor management, and inadequate procedural controls.</td>
<td>Degradation of the containment pressure suppression system. In case of a LOCA, potential for primary containment overpressurization due to reduction of the amount of steam condensed by the torus.</td>
<td>Modify hardware &amp; procedures related to the inerting system. Issue an NRC IE bulletin alerting licensees to the problem. Improve visual inspection of components.</td>
</tr>
</tbody>
</table>

<p>| Torus corrosion pitting - missing structural welds. | 455, U.S., Oyster Creek 1, Oct - Nov 83, BWR. | S/D | Corrosion pitting of the torus was due to local failures of original coating - possibly poor water chemistry. A lapse in construction management supervision led to the missing welds. | Degradation of containment structural integrity. | Recoat torus and repair welds. Use high quality demineralized water. Emphasize the importance of periodic inspection &amp; controls of torus structures. |</p>
<table>
<thead>
<tr>
<th>EVENT HIGHLIGHTS</th>
<th>IDENTIFICATION</th>
<th>UNIT STATUS (% F.P.)</th>
<th>CAUSES</th>
<th>CONSEQUENCES</th>
<th>LESSONS LEARNED AND/OR ACTIONS TAKEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Failures in vacuum building roof tendons &amp; deterioration of seal rubber- (common to 4 units).</td>
<td>113, Canada, Bruce A, Sep. 81, CANDU.</td>
<td>88</td>
<td>Poor choice of materials and poor protection against weather conditions resulted in cable corrosion and seal deterioration.</td>
<td>Minor degradation in containment structural integrity.</td>
<td>Increase the seal inspection frequency.</td>
</tr>
<tr>
<td>Vacuum building pressure excursion during maintenance.</td>
<td>120, Canada, Bruce A, Jan. 82, CANDU.</td>
<td>85</td>
<td>Defective work planning and poor procedures. Inadequate instrumentation &amp; improper valving contributed.</td>
<td>Minor degradation of vacuum in vacuum building which is common to the 4 units. Potential for serious personnel injury. Pressure excursion in containment.</td>
<td>Improve the appropriate vacuum building instruments. Review procedures. Improve personnel training.</td>
</tr>
</tbody>
</table>
## APPENDIX E
### Category E Event Summaries

<table>
<thead>
<tr>
<th>EVENT HIGHLIGHTS</th>
<th>IDENTIFICATION</th>
<th>UNIT STATUS (% F.P.)</th>
<th>CAUSES</th>
<th>CONSEQUENCES</th>
<th>LESSONS LEARNED AND/OR ACTIONS TAKEN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inoperability of a hydrogen recombiner.</td>
<td>176.02, Trojan.</td>
<td>90</td>
<td>Loss of control power due to inadvertant actuation of the fire deluge system while welding in an electrical penetration.</td>
<td>Unavailability of a hydrogen recombiner.</td>
<td>Replace the shorted power transformer. Improve training of maintenance personnel related to ventilation while welding.</td>
</tr>
<tr>
<td>High oxygen concentration in drywell for 4.3 hours.</td>
<td>241.03, Brunswick 2.</td>
<td>80</td>
<td>Frozen condensate in the containment atmospheric control system (nitrogen inerting) prevented nitrogen makeup to the drywell. The required trace heating &amp; lagging on the frozen line were missing. No explanation was found.</td>
<td>No actual consequences.</td>
<td>Repair the inerting system. Emphasise the importance of the inerting system on hydrogen &amp; oxygen concentration control in the drywell.</td>
</tr>
<tr>
<td>Undetected failure of the containment hydrogen monitoring system.</td>
<td>479, Wülsen.</td>
<td>100</td>
<td>Inadequate mechanical design led to diaphragm failure of the system pump.</td>
<td>System unavailability in the event of any incident involving a hydrogen release.</td>
<td>Modify the affected pump. Install a redundant pump. Modify the alarms such that the pump suction is monitored.</td>
</tr>
<tr>
<td>EVENT HIGHLIGHTS</td>
<td>IDENTIFICATION</td>
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<tr>
<td>High primary containment pressure and temperature following a reactor trip.</td>
<td>566, U.S., Quad Cities 2, Jan. 85, BWR.</td>
<td>S/D</td>
<td>Very cold river water temperature coupled with the concern for recirculation pump seal embrittlement led to the availability of only one reactor building closed cooling water heat exchanger, which resulted in system overloading.</td>
<td>Potential for equipment damage due to the high containment temperature and pressure.</td>
<td>Emphasize the impact of a non-safety grade system (such as the RBCW) on the containment.</td>
</tr>
<tr>
<td>High primary containment pressure &amp; temperature following a reactor trip.</td>
<td>610, U.S., Lasalle 1, May 85, BWR.</td>
<td>10</td>
<td>Manual depressurization of the reactor using safety valves (with RCIC as well) in order to reach cold shutdown.</td>
<td>High temperature in drywell for about 8 hours. A Sargent &amp; Landy analysis showed that the impact on equipment was insignificant.</td>
<td>Emphasize the potential impact of plant transients on the containment.</td>
</tr>
<tr>
<td>4.5 hour insulation fire in the containment drywell expansion gap.</td>
<td>691, U.S., Dresden 3, Jan. 86, BWR.</td>
<td>S/D</td>
<td>Hot molten material from an arc cutting activity in the RWCU heat exchanger room came in contact with and ignited the polyurethane foam in the drywell expansion gap. The presence of this material was not considered in the licensee’s fire protection reviews.</td>
<td>Substantial burning occurred. It was suspected that the drywell expansion gap temperature was higher than expected.</td>
<td>Corrective actions were still under investigation at the time the report was written. - The resins in the drywell expansion gap will be included in the fire analysis</td>
</tr>
<tr>
<td>EVENT HIGHLIGHTS</td>
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<tr>
<td>Partial blockage of ice condenser flow lines for several months</td>
<td>414.09, Finland, Loviisa</td>
<td>S/D</td>
<td>Inadequate work control and surveillance during the 1983 maintenance outage resulted in carelessness while adding ice to the ice baskets.</td>
<td>Reduced capability to condense steam in the event of a break inside containment.</td>
<td>Improve procedures, work control and surveillance.</td>
</tr>
<tr>
<td>23 out of 24 ice condenser lower inlet doors blocked closed for 9 days during hot shutdown.</td>
<td>565, U.S., Catawba I, Dec. 84, PWR.</td>
<td>S/D</td>
<td>Personnel missed door position verification. Poor work control and verification contributed.</td>
<td>SEE EVENT 414.09 ABOVE.</td>
<td>Improve mechanical maintenance procedures related to the installation of door blocking devices.</td>
</tr>
<tr>
<td>Power supply failure and leakage inside drywell.</td>
<td>6.03, Japan, Fukushima Daiichi 4, Feb. 80, BWR.</td>
<td>100</td>
<td>Inadequate tightening of fan motor bolts in dry well ventilation system resulted in high vibration and a break in the air vent pipe.</td>
<td>The unit was shutdown. About 200 tons of water leaked into drywell. Damages were sustained to the fan-motor bearing and the air vent pipe. No effect on containment integrity.</td>
<td>Effect repairs of damaged/broken equipment.</td>
</tr>
<tr>
<td>EVENT HIGHLIGHTS</td>
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<tr>
<td>12 minute loss of 3 out of 4 west fuelling machine vault air-cooling units.</td>
<td>Canada, Pickering 4, Sep. 63, CANDU.</td>
<td>100</td>
<td>Loss of power supply due to steam entering a class 3, 4hv transformer protective relay panel. A standby bus breaker problem also contributed.</td>
<td>Degradation of containment area cooling function.</td>
<td>- Check similar equipment. - Emphasise the importance of efficient power supplies to the containment support systems (eg. air coolers).</td>
</tr>
</tbody>
</table>
APPENDIX F

Brief Descriptions of Typical Containment Designs involved in this Study

1. General

In meeting the requirements of the national codes and standards, containment systems also usually conform to principles developed by such institutions as the American Society of Mechanical Engineers (ASME) for the steel components, and the American Concrete Institute (ACI) for the concrete structures. Local external phenomena and/or conditions, are generally taken into account in the national codes, in the licensee's own design guides, in regulatory requirements, or in any combination thereof; examples of such phenomena are seismicity, siting conditions and aircraft crashes.

In many of the US, Japanese, and European light water reactors, a double containment is used with the outer shell being concrete and the inner one either steel or prestressed concrete. The outer reinforced concrete shell protects the inner one (whose function is to resist internal pressure and ensure leaktightness) against external loads and makes it possible to collect and filter leakages escaping past the inner shell; unplanned releases are thus considerably reduced even in the case of large leaks. A containment annulus ventilation system is used to recover leaks from the inner wall, treat them in iodine traps and finally release them through the stack, which process ensures satisfactory dispersal in the atmosphere.

Containment systems in the CANDU plants are of a different design and will be described in a later section of this Appendix.

2. BWR Containments

BWR containments are of the pressure suppression type, incorporating drywells and wetwells as pressure suppression chambers. Following a LOCA, the steam/water flow causes a rapid increase of pressure and temperature in the drywell.

The pressure difference between the drywell and wetwell clears the ventpipe systems and drywell steam flows into the wetwell pool. Steam condensation occurs and non-condensible gases are collected in the wetwell airspace.

This condensation process is the key element in the limitation of the maximum accident pressures. The overall objective of BWR containments is to maintain low design pressure with relatively small containment volumes.
Several different design concepts for pressure suppression systems are used. They consist mostly of a cylindrical prestressed concrete structure with an embedded steel liner which is protected by additional concrete. The drywell space surrounds the reactor vessel and heat transport piping up to the second isolation valve. Some designs have the first isolation valve inside the drywell and the second one outside but connected to the drywell. Many large diameter vent pipes connect the drywell with the wetwell pool and provide the path to condense LOCA induced steam/water mixtures. A separate pressure relief valve also dumps energy into the wetwell pool during primary coolant system pressure control. The containment system is protected against negative consequences of drywell sub-atmospheric pressures by vacuum breaker swing check valves which allow pressure equalization between the two regions. The drywell design pressure is typically 4 bar compared with maximum expected LOCA pressures of less than 3 bar. The wetwell design pressure depends on the size of the wetwell and ranges from 2 - 4 bars. The wall of the reactor building serves as a secondary containment and the space between it and the containment, if not lined concrete, is slightly below atmospheric pressure to control the leakage to the environment.

The Mark 1, 2 and 3 containment types represent evolutions in the design described above; the main concept though of pressure suppression remained unchanged. Containment auxiliary systems will be described in a later section.

3. PWR Containments

In most PWRs the "dry containment" design is adopted, in that reliance to contain the radioactivity and energy released following a LOCA is on a large free volume; pressure, temperature and radioactivity control is effected via the auxiliary systems.

The design evolved from a single to a double walled containment building, with the two-metre annulus being maintained at a slightly negative pressure (see details in Section 1 of this Appendix). Some PWR containments use an "ice condenser" as a heat sink.

The ice condenser containment is divided into two main compartments. The containment internal structures separate the upper compartment from the lower one so that only narrow flow paths exist between the two. If the lower compartment pressure increases, for example as a result of a high-energy pipe rupture, the ice condenser vent doors will open and a flow path from the lower into the upper compartment established through the ice condenser. When the high-pressure mixture flows between the columns of borated ice, the steam is condensed on the ice surface.
The ice condenser system is designed such that:

- the heat transfer from the steam to the ice columns is sufficient in all postulated accident conditions
- the ice condenser structures maintain their geometry, under any accident loadings
- the vent doors open in a reliable way, and
- the bypass flow paths from the lower to the upper compartment are not larger than assumed in the containment analysis.

4. CANDU Containments

Multi-unit CANDU nuclear power stations are serviced by a central, negative pressure containment system. The reactors are contained in individual reactor buildings connected to a large, common vacuum building via pressure relief ducts and self-actuating pressure relief valves. The valves are normally closed, isolating the reactor buildings from the vacuum in the vacuum building. In the event of an accident involving a rise in reactor building pressure, the reactor building’s atmosphere is vented into the vacuum building where the vacuum reserve plus an internal dousing system quickly terminate the overpressure excursion. The activity release from containment during the short overpressure is low and constitutes only a small fraction of the nuclear safety siteing guideline limits. In the long term, the containment pressure will rise slowly towards sub-atmospheric pressure as the vacuum reserve in the vacuum building is depleted due to containment in-leakage as well as instrument air in-flow. To maintain the containment sub-atmospheric, a filtered-air discharge system is provided to permit controlled venting via a filtered pathway, which removes most radiiodines but does not prevent the release of radioactive noble gases. The period available before first venting and the required rate of venting varies for different station designs, but in all cases the calculated public doses are well within the regulatory dose limits.

5. Containment Auxiliary Systems

Generally speaking, the containment designs described above rely on auxiliary systems to perform the following functions in the event of an accident:

a) isolate the pipes penetrating the containment walls to minimize radioactive releases;

b) spray water into the containment to reduce pressure, temperature and airborne iodine concentration; and

c) maintain hydrogen and oxygen concentrations below specified limits (this is also applicable in non-accident conditions).

Furthermore, under normal operating conditions, auxiliary systems maintain containment pressure and temperature within specified limits.
APPENDIX G

THE ITALIAN APPROACH TO SYSTEMATIC ANALYSIS OF EVENTS
Form adopted for the analysis

EVENT

OBSERVED CAUSES

CONTRIBUTING ELEMENTS

EFFECTS

CORRECTIVE ACTIONS

IMPLEMENTED OR PLANNED IMPROVEMENTS

ROOT CAUSES

LESSONS LEARNED
### CAUSES OF FAILURE

#### MECHANICAL FAILURE
- Corrosion, erosion, fouling
- Vibration
- Break, rupture, crack, weld failure
- Blockage, restriction, obstruction, binding, foreign material
- Information, distortion, displacement, movement, misalignment
- Other
- Poor chemical control
- Design deficiency (including material)

#### ELECTRICAL FAILURE
- S.1
- S.1.5
- S.1.7
- S.1.9
- S.2
- S.3
- S.3.5
- S.3.1
- S.3.2
- S.3.3

#### CHEMICAL OR CORE PHYSICS FAILURE
- S.8.1
- S.8.2
- S.8.3
- S.8.4
- S.8.4.2
- S.8.4.4

#### HUMAN FACTORS
- Manufacture, construction, installation error or deficiencies
- Procedure deficiency
- Operator error, carelessness, confusion
- Violation of technical specification, or any other procedures
- Inspection, maintenance, testing or calibration error
- Management, organization, or work place, g error
- Indiscretion training

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**NOTE:** (1) Both observed and root causes are reported, such to highlight all phenomena involved in the event.
(2) Event 375 was split into three occurrences, each one independently analyzed.
### DISTRIBUTION OF FAILURE TYPE VERSUS IRS REPORTS

#### TYPE OF FAILURE

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EVENT
MSIV UNEXPECTED CLOSURE DURING A TEST

OBSERVED CAUSES
5.1.1.6 - FOREIGN MATERIAL INSIDE MSIV OPERATOR

EFFECTS
HIGH MAIN STEAM FLOW AND SUBSEQUENT REACTOR SCRAM

CORRECTIVE ACTIONS
PILOT VALVE ASSEMBLY REPLACING

IMPLEMENTED OR PLANNED IMPROVEMENTS
MSIV PILOT ASSEMBLIES SUBSTITUTION WITH OTHERS OF DIFFERENT TYPE DURING SUBSEQUENT MAINTENANCE OUTAGE

LESSONS LEARNED
PROBLEMS TO MSIVs FROM THEIR PNEUMATIC CONTROL SYSTEM

CONTRIBUTING ELEMENTS
PISTON SLEEVE AND PILOT PISTON SCURED BY SMALL DEBRIS FROM AIR CONTROL SYSTEM

ROOT CAUSES
5.2.8.5 - INSUFFICIENT MAINTENANCE
EVENT
FAILURES OF ROCKWELL MAIN STEAM ISOLATION VALVES (MSIVs)

OBSERVED CAUSES
DISENGAGING OF MAIN VALVE DISC FROM VALVE STEM

CONTRIBUTING ELEMENTS
- VIBRATIONS:
  - FATIGUE FAILURE OF PINS
  - LOSS OF THREADED JOINT PRELOAD
  - LOOSENESS OF THREADED CONNECTIONS
  - STEAM INSTABILITIES

EFFECTS
REACTOR SCRAM

CORRECTIVE ACTIONS
VALVES REPAIRS AND MODIFICATIONS

IMPLEMENTED OR PLANNED IMPROVEMENTS
- VALVE DESIGN MODIFICATIONS
- CONSTRUCTOR RECOMMENDATIONS TO FACILITIES

LESSONS LEARNED
- IMPORTANCE OF PROPER DESIGN REVIEW
- POTENTIAL OF LOSS OF BOTH MSIVs ON SAME STEAM LINE

ROOT CAUSES
5.2.8.1 - DESIGN DEFICIENCY OF ROCKWELL ISOLATION VALVES
5.2.1.5 - VIBRATIONS
EVENT
FAILURE OF ROCKWELL MAIN STEAM ISOLATION VALVES ON "C" AND "D" MAIN STEAM LINES

OBSERVED CAUSES
VALVES MAIN DISC SEPARATION FROM STEM AND DROP INTO BODY SEAT

CONTRIBUTING ELEMENTS
- VIBRATIONS:
  - FATIGUE FAILURE OF PINS
  - LOSS OF THREAD JOINT PRELOAD
  - LOOSENESS OF THREADED CONNECTIONS
- STEAM INSTABILITIES

EFFECTS
- POWER SPIKE AND SUBSEQUENT REACTOR SCRAM
- LOW REACTOR LEVEL

CORRECTIVE ACTIONS
VALVES REPAIRS AND MODIFICATIONS

IMPLEMENTED OR PLANNED IMPROVEMENTS
- VALVE DESIGN MODIFICATIONS
- CONSTRUCTOR RECOMMENDATIONS TO FACILITIES

LESSONS LEARNED
- IMPORTANCE OF PROPER DESIGN REVIEW
- POTENTIAL OF LOSS OF BOTH MSIVs ON SAME STEAM LINE

ROOT CAUSES
5.2.8.1 - DESIGN DEFICIENCY OF ROCKWELL ISOLATION VALVES
5.2.1.5 - VIBRATIONS
EVENT
CRACKED HYDRAULIC SPEED CONTROL CYLINDERS ON TWO MSIVs

OBSERVED CAUSES
5.1.8.4.2 - OPERATOR ERROR INBOARD MSIVs ERRONEOUSLY OPENED BEFORE OUTBOARD ONES

CONTRIBUTING ELEMENTS
- DEFICIENCY OF THE START-UP PROCEDURE
- PROCEDURE NOT FOLLOWED

EFFECTS
- MSIVs INOPERABILITY
- DEGRADATION OF CONTAINMENT FUNCTION

CORRECTIVE ACTIONS
- HYDRAULIC CYLINDERS AND SPEED CONTROL ASSEMBLY REPAIRS
- INBOARD MSIVs INSPECTION AND LEAK RATE TEST PERFORMANCE

IMPLEMENTED OR PLANNED IMPROVEMENTS
- PROCEDURES AND OPERATING INSTRUCTION REVISION
- ADDITIONAL OPERATOR TRAINING

ROOT CAUSES
5.2.8.4.4 - VIOLATION OF STARTUP PROCEDURES
5.2.8.3 - PROCEDURAL DEFICIENCY
5.2.8.9 - INADEQUATE TRAINING

LESSONS LEARNED
- PROPER OPERATOR TRAINING
- DETAILED OPERATING INSTRUCTION FOR REACTOR START-UP
EVENT
INADVERTENT DECREASING OF MAIN STEAM LINE PRESSURE CAUSING MSIV CLOSURE

OBSERVED CAUSES
MALFUNCTION OF ELECTRICAL PRESSURE REGULATOR SERVO VALVE 5.1.2.0-OTHER ELECTRICAL CAUSE

CONTRIBUTING ELEMENTS
SERVO VALVE DEFICIENT IN ELECTRICAL CHARACTERISTIC

EFFECTS
- REACTOR SCRAM
- TURBINE TRIP
- SMALL EXTERNAL RELEASE

CORRECTIVE ACTIONS
VALVE REPLACEMENT AND TEST

IMPLEMENTED OR PLANNED IMPROVEMENTS
- TURBINE SERVO VALVE CHECK OF INPUT-OUTPUT CHARACTERISTIC
- SERVO VALVES REPLACEMENT EVERY PERIODICAL INSPECTION WITH SPARES CHECKED BY MANUFACTURER

LESSONS LEARNED
MSIVs SPURIOUS CLOSURE DUE TO PRESSURE REGULATOR FAILURE

ROOT CAUSES
5.2.8.5 -INSUFFICIENT MAINTENANCE
EVENT
IMPROPER USE OF LUBRICANT ON VALVE SEATS

OBSERVED CAUSES
MAINTENANCE SURVEILLANCE DEFICIENCY

EFFECTS
NON CONSERVATIVE LLRT RESULTS

CORRECTIVE ACTIONS
VALVES INSPECTION, CLEANING AND LEAK RATE TEST REPETITION

IMPLEMENTED OR PLANNED IMPROVEMENTS
- DEVELOPMENT BY QC DEPARTMENT OF STANDARD INSPECTION PLANS FOR ISOLATION VALVES MAINTENANCE
- MAINTENANCE PROCEDURES REVISION
- MODIFICATION OF MAINTENANCE TRAINING PROGRAM
- DEVELOPMENT OF A REPAIR MANUAL

LESSONS LEARNED
- NECESSITY OF MAINTENANCE SURVEILLANCE
- OPERATOR TRAINING ON CORRECT ISOLATION VALVES REASSEMBLING

CONTRIBUTING ELEMENTS
DIFFICULTY OF ISOLATION VALVES REASSEMBLING OPERATIONS

ROOT CAUSES
5.2.8.4.4 - VIOLATION OF TECHNICAL SPECIFICATION
5.2.8.5 - MAINTENANCE ERROR
EVENT
LEAK TIGHTNESS DEGRADATION OF PRIMARY CONTAINMENT IN THE OPERATING CYCLES

OBSERVED CAUSES
5.1.1.1 - CORROSION OF CARBON STEEL PIPES

CONTRIBUTING ELEMENTS
- ACCUMULATION OF CORROSION PRODUCTS INSIDE VALVE SEAT
- CHEMICAL CHARACTERISTICS OF WATER
- FREQUENT USE OF THE PURGE SYSTEM OF THE CONTAINMENT
- DETERIORATION OF THE TEFLOM GASKETS FOR BUTTERFLY VALVES
- PROCEDURAL AND MAINTENANCE DEFICIENCIES DELAYED THE EARLY IDENTIFICATION OF THE PROBLEM

EFFECTS
DEGRADATION OF CONTAINMENT INTEGRITY AND LEAK TIGHTNESS

CORRECTIVE ACTIONS
- REPLACEMENT OF THE STELLITE FILLING ON THE SHUTTER SURFACE
- LAPPING OF DISK SURFACES
- SUBSTITUTION OF TEFLOM GASKETS
- VALVES REPLACEMENT

IMPLEMENTED OR PLANNED IMPROVEMENTS
- PAINTING OF INNER SURFACE OF PURGE SYSTEM PIPES WITH PASSIVATORS
- VALVES REPLACEMENT WITH BETTER QUALITY ONES
- TEST PROCEDURES REVISION

ROOT CAUSES
5.2.3.5 - POOR CONTROL OF THE CHEMICAL CHARACTERISTICS OF THE WATER
5.2.1.8 - FOREIGN MATERIAL
5.2.8.3 - PROCEDURAL DEFICIENCY
5.2.8.5 - MAINTENANCE DEFICIENCY

LESSONS LEARNED
- COMPATIBILITY OF PIPES MATERIAL, CHARACTERISTICS OF FLUIDS AND VALVE TYPES
- TEST PROCEDURES EXPLICITLY ADDRESSED TO COLLECT AND TO ASSESS TEST RESULTS "BEFORE" AND "AFTER" MAINTENANCE
EVENT
UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE THE PRIMARY CONTAINMENT

OBSERVED CAUSES
FAILURE OF THE TWO SDV DRAIN VALVES TO FULLY ISOLATE

CONTRIBUTING ELEMENTS
- IMPROPER STROKE ADJUSTMENT OF ONE VALVE
- INADVERTENT OPENING OF THE SECOND CLOSED VALVE DUE TO UNDERSIZING OF THE ACTUATOR CLOSING SPRING

EFFECTS
- OVERALL CONTAINMENT BYPASS
- LEVEL AND PRESSURE TRANSIENTS
- POTENTIAL LOSS OF SAFETY FUNCTIONS DUE TO THE FIRE DELUGE SYSTEM ACTUATION

CORRECTIVE ACTIONS
- STROKE READJUSTMENT
- SUBSTITUTION OF THE SPRING

ROOT CAUSES
5.2.8.5. -CALIBRATION ERROR
5.2.8.2 -INSTALLATION ERROR

IMPLEMENTED OR PLANNED IMPROVEMENTS
- REVISION OF THE OPERATORS TRAINING PROGRAM
- SCRAM LOGIC DESIGN REVIEW

LESSONS LEARNED
- DISCUSSION WITH MAINTENANCE OPERATORS OF ABNORMAL EVENTS
- Q.A. IMPROVEMENT DURING INSTALLATION AND SUBSTITUTION
- SCRAM DISCHARGE SYSTEM AS POTENTIAL PATH FOR LEAKAGE OF REACTOR COOLANT
- AREA EFFECTS AS POTENTIAL COMMON CAUSE
EVENT
-Failures of torus coating and vent system
-Missing structural welds

OBSERVED CAUSES
5.1.1.1 - Corrosion pitting
5.1.8.8 - Error in construction management overview

CONTRIBUTING ELEMENTS
Corrosion:
-Materials: red lead coating
-Environment: moisture, chemical characteristics of the water

EFFECTS
Degradation of containment structural integrity

CORRECTIVE ACTIONS
Welds repairs

IMPLEMENTED OR PLANNED IMPROVEMENTS
-Torus recoated with an epoxy coating
-Use of high quality demineralized water

ROOT CAUSES
5.2.8.1 - Design deficiency
5.2.8.2 - Manufacturing, construction...error

LESSONS LEARNED
-Controls after construction (welds inspection)
-Inspections and periodic controls to detect condition of corrosion
EVENT
- FAILURE OF DOME TENDON ANCHOR HEADS
- FAILURES OF CONTAINMENT TENDON FIELD ANCHORS

OBSERVED CAUSES
5.1.1.7 - HYDROGEN STRESS CRACKING

CONTRIBUTING ELEMENTS
- HYDROLYSIS OF THE WATER (HYDROGEN DEVELOPMENT AND ITS DIFFUSION INSIDE STEEL)

MATERIALS:
- HIGH-STRENGTH STEEL SUBJECTED TO SUSTAINED TENSILE STRESSES
- PARTICLES OF ZINC FROM ABRASION DURING TENDON INSTALLATION

ENVIRONMENT:
- PENETRATED MOISTURE FOR GREASE CAP INEFFECTIVENESS

ROOT CAUSES
5.2.1.1 - CORROSION
5.2.8.2 - INSTALLATION ERROR
5.2.8.5 - INSPECTION, MAINTENANCE, ERROR

EFFECTS
DEGRADATION OF CONTAINMENT STRUCTURAL INTEGRITY

CORRECTIVE ACTIONS
REPLACEMENT OF AFFECTED PARTS

IMPLEMENTED OR PLANNED IMPROVEMENTS
- ANCHOR HEADS WITH NEW COATING
- REGREASE OF THE COMPLETED ANCHOR HEAD ASSEMBLY
- PERIODICAL INSPECTION OF THE COATING

LESSONS LEARNED
- CORRECTED INSTALLATION OF THE GREASE CAPS TO PREVENT MOISTURE PENETRATION
- INSPECTION AFTER INSTALLATION
- PERIODICAL CONTROL OF ENVIRONMENTAL CONDITIONS (MOISTURE CONTROL, VISUAL CONTROL TO VERIFY PRESENCE OF WATER)
EVENT
THROUG-WALL CRACK IN THE VENT HEADER IN THE TORUS

OBSERVED CAUSES
5.1.1.7 - BRITTLE-FRACTURE TYPE OF FAILURE

EFFECTS
DEGRADATIONS OF CONTAINMENT PRESSURE SUPPRESSION SYSTEM

CORRECTIVE ACTIONS
REPAIRS OF THE VENT HEADER

IMPLEMENTED OR PLANNED IMPROVEMENTS
DIFFERENT LOCATION OF THE INERTING SYSTEM NITROGEN INJECTION PIPE
PROCEDURE CHANGES TO IMPROVE THE STEPS INVOLVED WITH INERTING THE CONTAINMENT
REDUNDANT ISOLATION ON LOW TEMP. OF NITROGEN INJECTION
VISUAL INSPECTION OF THE COMPONENTS

LESSONS LEARNED
CONTROLS AND MAINTENANCE OPERATIONS ON INERTING SYSTEM

CONTRIBUTING ELEMENTS
- MATERIAL (CARBON STEEL) OF THE VENT HEADER SUBJECTED TO TEMPERATURE BELOW ITS NIHIL DUCTILITY TEMPERATURE
- THERMAL STRESSES
- FAILURES OR MALFUNCTION OF COMPONENTS OF CONTAINMENT INERTING SYSTEM
  - EVAPORATOR
  - HEATERS
  - ISOLATION VALVE OF NITROGEN INJECTION

ROOT CAUSES
5.2.8.3 - PROCEDURAL DEFICIENCY
5.2.8.5 - INSPECTION, MAINTENANCE... ERROR
APPENDIX H

THE FRENCH APPROACH TO SYSTEMATIC EVENT ANALYSIS
## FICHE "ANALYSE" DES INCIDENTS

<table>
<thead>
<tr>
<th>PARAMÈTRES</th>
<th>CONCEPTION</th>
<th>SYSTÈMES COMPOSANTS</th>
<th>PROCÉDURES</th>
<th>FACTEUR HUMAIN FORMATION DES OPERATIONS</th>
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</thead>
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| ** Causes ** | Injection de sécurité | | | Erreurs humaines :  
- non utilisation procédures  
- actions non demandées par consigne, irrespect consigne. |
| ** Conséquences** | - Perte de confinement | Débordement par trop plein du ballon 21000 ppm  
Contamination du bâtiment des auxiliaires nucléaires :  
Haute activité chimique | | |
| ** Enseignements tirés** | Système d'aide au diagnostic | | | Rappel aux opérateurs  
+ diagnostic  
+ application rigoureuse des consignes |
## FICHE "ANALYSE" DES INCIDENTS

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<td>Causes</td>
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<td>Vannes d'isolement RRA/Ptr non fermées</td>
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<td>Mauvais lignage.</td>
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<td>Conséquences</td>
<td>Potentiellement: - Perte refroidissement piscine combustible. Si cœur chargé, brèche primaire hors enceinte.</td>
<td>- Vanne sur la liaison RRA/Ptr* devenue fuyarde 6 m³ d'eau primaire perdu. - Vannes et étanchéité des pompes PTR endommagées</td>
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<tr>
<td>Enseignements tirés</td>
<td>Retransmission position des vannes en salle de commande - alarme signalant une montée en pression.</td>
<td>- définition des cas d'utilisation des liaisons RRA/Ptr - prise en compte des définitions et précautions dans les procédures de conduite concernées. Précautions à prendre pour éviter de pressuriser le PTR.</td>
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* RRA : Circuit de refroidissement à l'arrêt.
PTR : Circuit de traitement et de refroidissement d'eau des piscines.
## FICHE "ANALYSE" DES INCIDENTS

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<th>PARAMETRES</th>
<th>CONCEPTION</th>
<th>SYSTEMES COMPOSANTS</th>
<th>PROCEDURES</th>
<th>FACTEUR HUMAIN FORMATION DES OPERATIONS</th>
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</thead>
<tbody>
<tr>
<td>Causes</td>
<td>Défaillance de mode commun</td>
<td>Rupture du bouchon du détendeur alimentant les joints du sas Om</td>
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<tr>
<td>Consequences</td>
<td>Perte Atachéts sas Om : perte 3ème barrière qui entraîne rejet 1000 m³ hors bâtiment réacteur</td>
<td>Joints du sas Om dégonflés.</td>
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<tr>
<td>Enseignements tirés</td>
<td>- Mise en évidence d'un défaut de mode commun sur l'alimentation en air des deux portes du sas.</td>
<td>- Dédoublement détendeur.</td>
<td>- Isolation automatique air en cas de baisse de pression</td>
<td>- Secours air avec compresseur mobile.</td>
<td>- Régloage alarme &quot;sas défaut&quot;.</td>
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</tr>
</tbody>
</table>
**FICHE "ANALYSE" DES INCIDENTS**

<table>
<thead>
<tr>
<th>Paramètres</th>
<th>Conception</th>
<th>Systèmes Composants</th>
<th>Procédures</th>
<th>Facteur Humain Formation Des Opérations</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Causes</strong></td>
<td>Fuite sur filtre réseau d'air : le resserrage entraîne sa rupture.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Conséquences</strong></td>
<td>- Perte d'étanchéité du sas 8 m - perte 3ème barrière pendant 2 mn et 14 s.</td>
<td>Les deux joints des portes du sas 8 m se dégonflent.</td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Enseignements tirés</strong></td>
<td>Mise en évidence d'un défaut de mode commun sur l'alimentation en air des joints des portes, spécifique aux tranches de ce site.</td>
<td>Modification du système d'alimentation en air des joints sur les quatre tranches.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>INCIDENTS</td>
<td>PARAMÈTRES</td>
<td>CONCEPTION</td>
<td>SYSTÈMES COMPOSANTS</td>
<td>PROCÉDURES</td>
</tr>
<tr>
<td>-----------</td>
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</tr>
<tr>
<td>Causes</td>
<td>- Démantellement non identifié - Suite manœuvres conformes à la procédure on a porte externe étanche et porte interne fermée non verrouillée, joints non gonflés sur sas 0m.</td>
<td>- Manoeuvrabilité des sas personnels.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Conséquences</td>
<td>- Perte d'étanchéité du sas 0m - perte 3ème barrière pendant 4h 30.</td>
<td>Dégondrage des joints porte externe non expliqué.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Enseignements tirés</td>
<td></td>
<td></td>
<td>Gamme d'essai revue.</td>
<td></td>
</tr>
<tr>
<td>PARAMÈTRES</td>
<td>CONCEPTION</td>
<td>SYSTEMES COMPOSANTS</td>
<td>PROCÉDURES</td>
<td>FACTEUR HUMAIN FORMATION DES OPERATIONS</td>
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</tr>
<tr>
<td>Causes</td>
<td>Mode commun toujours existant sur l'alimentation en air des joints des portes.</td>
<td>Fuite d'air au niveau du bouchon d'un détendeur du sas 8 m sur tranche 1.</td>
<td>Modifications suite à Incident du 03.08.82 sur Tricastin 1 non encore réalisées. Emprunt sur tranche 2 du même bouchon décidé par chef de quart.</td>
<td></td>
</tr>
<tr>
<td>Conséquences</td>
<td>Situation d'agrandissement des sas 8 m des deux tranches.</td>
<td>Portes internes fermées étanches portes externes ouvertes des sas 8 m des deux tranches.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Enseignements tirés</td>
<td>Remplacement des bouchons des détendeurs.</td>
<td></td>
<td>Décision de mise en œuvre des modifications pour supprimer le défaut de mode commun (voir IRS 224).</td>
<td></td>
</tr>
</tbody>
</table>
## FICHE "ANALYSE" DES INCIDENTS

<table>
<thead>
<tr>
<th>Incident : IRS 508 V-F CHINON B2 28.05.84</th>
<th>PARAMETRES</th>
<th>CONCEPTION</th>
<th>SYSTEMES COMPOSANTS</th>
<th>PROCEDURES</th>
<th>FACTEUR HUMAIN FORMATION DES OPERATIONS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Causes</td>
<td>Désordre alarme (seuil réglé trop bas) saas non étanche.</td>
<td></td>
<td></td>
<td>- Mauvaise organisation modification.</td>
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<td></td>
<td></td>
<td></td>
<td></td>
<td>- Mauvaise transmission: orale et non écrite.</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>- Non rédaction de consigne temporaire.</td>
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<td></td>
<td></td>
<td></td>
<td></td>
<td>- Oubli d'informer l'Ingénieur Sécurité et Radioprotection.</td>
<td></td>
</tr>
<tr>
<td>Conséquences</td>
<td>Perte étanchéité saas 0m - perte 3ème barrière qui entraîne rejet de 1200 m³/hours enceinte.</td>
<td>Joints des portes du saas 0m dégonflés.</td>
<td></td>
<td></td>
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</tr>
<tr>
<td>Enseignements tirés</td>
<td>Réglage seuil d'alarme.</td>
<td></td>
<td>Revue de l'organisation de la qualité.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Incident : IRS 508 VI GRAVELINES 3-F 24.09.88</td>
<td>PARAMETRES</td>
<td>CONCEPTION</td>
<td>SYSTEMES COMPOSANTS</td>
<td>PROCEDURES</td>
<td>FACTEUR HUMAIN FORMATION DES OPERATIONS</td>
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<tr>
<td>Enseignements tirés</td>
<td>Réexamen de la modification propre à Gravelines sur l'ouverture des sas.</td>
<td></td>
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</tr>
</tbody>
</table>
**FICHE "ANALYSE" DES INCIDENTS**

<table>
<thead>
<tr>
<th>PARAMÈTRES</th>
<th>CONCEPTION</th>
<th>SYSTEMES COMPOSANTS</th>
<th>ORGANISATION QUALITÉ ET PROCÉDURES</th>
<th>FACTEUR HUMAIN FORMATION DES OPÉRATIONS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Causes</td>
<td>Filtres provisoires pour les essais non enlevés à l'aspiration des pompes d'aspiration enceinte.</td>
<td>Mauvais montage dû aux plans techniques incomplets.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Conséquences</td>
<td>À terme, après LOCA, perte du système d'aspiration car filtres non prévus pour fonctionner dans ces conditions.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Enseignements tirés</td>
<td>Filtres enlevés. Lettre NRC à toutes les centrales sur ces événements. Revue de tous les documents de conception où des filtres auraient pu être installés.</td>
<td></td>
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<tr>
<td>Incident : IRS 292 TROJAN-USA 02.03.82</td>
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<td>----------------------------------------</td>
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<td>PROCEDURES</td>
<td>FACTEUR HUMAIN FORMATION DES OPERATIONS</td>
</tr>
<tr>
<td>Causes</td>
<td></td>
<td></td>
<td></td>
<td>Non respect de la procédure de remise en conformité des pompes d'aspersion voie B après E.P.</td>
</tr>
<tr>
<td>Conséquences</td>
<td>Etat dégradé du système d'aspersion.</td>
<td>Pompes d'aspersion voie B indisponibles.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Enseignements tirés</td>
<td></td>
<td></td>
<td>- Améliorer et fournir une meilleure mise en œuvre des procédures, - Développer et améliorer les aides et les outils des opérateurs, - Garder l'opérateur de tracasseries avec des contrôles administratifs superflus.</td>
<td>Formation avec insistance sur le suivi des opérations</td>
</tr>
</tbody>
</table>
APPENDIX I

THE FINNISH APPROACH; EMPHASIS ON GENERIC ANALYSIS FROM THE HUMAN FACTOR POINT OF VIEW
1. **Methodology**

Eight IRS reports are analysed in this section: 75, 414, 416.01, 416.02, 437.05, 448, 559 and 574 with the help of a special "human error" form (copy attached), one per event, to demonstrate its usefulness despite its current "experimental stage" status. The main objective was to present the problems identified, suggest some solutions and finally describe related Finnish experience, in order to stimulate international discussions.

In the attempt to recognize as many common features as possible, in order to facilitate drawing some generic conclusions, some difficulties were encountered primarily because of the insufficient information in some of the reports (to enable the use of the "human error" form) and the small number involved (ideally each participant should be asked to make his/her own observations concerning 50-100 events, to reduce misunderstandings arising from lack of clarity in some cases).

2. **Analysis**

a) Detailed analysis of the eight "human error" forms indicated that:

- five incidents involved procedural inadequacies;
- three incidents involved a failure to follow procedures;
- two to four incidents involved a lack of training; and
- in four incidents information on deficiencies was not sufficiently adequate to enable its use in the analysis performed.

b) Deficiencies in training and procedures:

In all the cases analyzed except 559, training deficiencies constituted at least one of the root causes identified.

In seven cases, the corrective actions taken were revisions and modifications of procedures and in seven cases, as well, additional training to the personnel had been given.

Generally speaking, it is very difficult to make any conclusions about the quality of a given training programme. Additional training could consist of any number of simulated transients discussed and analyzed or it could only be a repetition of the instructions, which should have been followed during the incident. Without adequate information about the types of training given an analysis of the sufficiency of training can hardly be done.
It was typical to the incidents analyzed, that components resetting were totally missing in the procedures, because it had been assumed that it would be performed anyhow. With lack of any independent inspection or, when the inspection fails to be done, the faults may remain undetected.

In one of the incidents the operative management of the plant shorted the procedure used to verify component positions and status and that caused the incident. Other deficiencies in training observed in the analysis were: lack of familiarity with the plant and poor understanding of system interactions.

3. Comparison of Similar Problems in Finland

There are four nuclear power plants in Finland: two boiling water reactors supplied by the Swedish Asea-Atom; and two pressurized water reactors supplied by the Soviet-Union.

The plants are very different from each other. The deficiencies in procedures have been more common at the PWR-units. An elimination of the observed deficiencies in the work order practices as well as in the instrumentation system for component status indications has been going on during the whole 80's. The problems in the work order practices have been almost the same as those described in the IRS-reports analyzed above. The observation has been made that the resetting of valves after surveillance or maintenance operations have sometimes been forgotten. That is why columns for control signatures have been added to all the working procedures. An acknowledgement shall be made in these columns always when a process component's state has been changed. In addition to the above acknowledgements, a so called 'normal-state-inspection' shall be conducted prior to a plant's start-up, in order to verify the correct positions of all the safety-related manual or motor-operated valves. The normal-state-inspection is always performed by two persons with a special check list. The most important manual valves will be locked in their positions and in the case of motor-driven valves they can be de-energized.

One important point to be mentioned in this context is the revision of working procedures and PI-diagrams.

Typically, these revisions take some months. During this time the incorrect procedures may guide the user to totally wrong actions. These problems are usually dependent on the personnel resources in the power company organization. A total revision and updating of all the procedures every 3 years has been a requirement in Finland in order to ensure that even all minor modifications are taken into consideration in all procedures.
In the training programs for the operations of the Finnish plants the importance of following the formal routines has been taken very well into consideration. At both plants they are emphasized in the familiarization of the personnel with the plant systems and in the theoretical training. In addition there is special training for operating procedures, disturbance procedures, plant's emergency plan and formal work practices.

For example, in the Finnish BWR's during the annual training course for 1987 2-3 days were reserved for work order practises and procedures. The corresponding part of PWR's training program consists of 5 hours of work planning practises, 3 hours of operating procedures and formal routines in control rooms and 4 hours of the Technical Specifications during the annual outage of the plant. Those issues also occur in some other training programs, for example, in the courses given because of structural changes at the plant.

4. Generic Conclusions

The following generic problems have been recognized in the incidents analyzed: valves were forgotten in wrong positions, personnel training was inadequate and, faults and deficiencies were found in the procedures used. Many problems are associated with a lack of work motivation of personnel, who should understand the importance of following formal routines. They should also be aware of the necessity of updated procedures even if everyone is aware of the modifications done to the plant systems. The personnel's education and their willingness to become familiar with the plant has never been a problem in Finland. There are many ways of correcting these deficiencies and every country has its own solutions. However, the following generic solutions could be suggested:

- A periodic revision and updating of procedures and instructions. This work must be coordinated very carefully so that the procedures could reach consistent forms and a high level of reliability. A homogenous group of procedures is easier to use under abnormal operating conditions.

- A complete and accurate check list for verification of valve positions prior to a start-up. These 'normal-state-position' check lists would always be updated so that any 'permanent deviations' can be avoided.

The root causes of any operational upsets should always be analyzed and the incident reports from the NEA and IAEA should be taken into consideration in the personnel training as much as possible because, when everything 'goes alright' at a plant, the importance of formal routines will usually be forgotten.
After system modifications the procedures should be updated as soon as possible in order to avoid the use of any incorrect procedures.

Independent inspections should be added to the internal routines of the plant, for example, a special Safety Inspection Group could make some operational inspections.

On the basis of incident 416.02, one generic conclusion can be made related to the work orders for surveillance and/or maintenance. The administrative controls should ensure that Technical Specifications require that work on redundant trains or components be done one at a time, and that resetting of one redundant part be effected before work is started on the other.
OPERATIONAL DISTURBANCES AND INCIDENTS INVOLVING HUMAN ERROR, FORM FOR STATISTICAL ANALYSIS

Docket number  ______________  Year  ______________  Unit  ______________

Date  ______________  Unit type  ______________  Hour  ______________

The title of incident:

____________________________________________________________________
____________________________________________________________________

Components and systems associated with the incident in order of the relative importance:

____________________________________________________________________
____________________________________________________________________
____________________________________________________________________
____________________________________________________________________

A short description of the incident:

____________________________________________________________________
____________________________________________________________________
____________________________________________________________________
____________________________________________________________________
1 PLANT STATUS PRIOR TO THE INCIDENT
   1.1 Raising power
   1.2 Nominal power
   1.3 Reducing power
   1.4 Hot stand-by
   1.5 Cold shut-down

2 TYPE OF DISCOVERY
   2.1 Alarm
   2.2 Measuring display
   2.3 Periodical maintenance
   2.4 Periodical inspection
   2.5 Occasional inspection
   2.6 Visual observation
   2.7 Others

3 CONSEQUENCES OF THE INCIDENT
   3.1 Reactor scram
   3.2 Degradation of power
   3.3 Degradation of safety systems
   3.4 Unplanned release of radioactive materials
      3.4.1 inside the plant
      3.4.2 outside the plant
   3.5 Spurious actuation of safety systems
   3.6 No significant consequences

4 ROOT CAUSE(S) OF THE INCIDENT
   4.1 Construction deficiency
   4.2 Human error
   4.3 Environmental cause
   4.4 Component failure
      4.4.1 single failure
      4.4.2 multiple failure
   4.5 Others
# Analysis of the Human Error

## 5 Classification of the Human Error

### 5.1 Effect of the error on the incident
- **5.1.1** missing action leading to the incident
- **5.1.2** spurious action causing the incident
- **5.1.3** spurious action impairing the incident
- **5.1.4** defective action impairing the incident

### 5.2 Detailed classification of the error
- **5.2.1** preparing of working routines
- **5.2.2** designing of the components and systems
- **5.2.3** administrative controls
- **5.2.4** maintenance error, mechanical
- **5.2.5** maintenance error, electrical
- **5.2.6** mechanical external damage (no connections to maintenance)
- **5.2.7** inspection
- **5.2.8** testing
- **5.2.9** calibration
- **5.2.10** operator error, erroneous manipulation
- **5.2.11** operator error, incorrect control/regulation
- **5.2.12** sabotage/terror
- **5.2.13** unknown reason

### 5.3 Character of the measures
- **5.3.1** routine
- **5.3.2** no-routine
- **5.3.3** emergency situation
5.4 Point of decision making process the error occurred

5.4.1 observation
5.4.2 interpretation
5.4.3 decision
5.4.4 action

5.5 Type of erroneous action

5.5.1 negligence
5.5.2 lacking knowledge
5.5.3 erroneous timing
5.5.4 erroneous order of measures
5.5.5 opposite action
5.5.6 substitute action
5.5.7 others

6 ROOT CAUSES OF THE HUMAN ERROR

6.1 Procedures

6.1.1 lacking procedures
6.1.2 procedure not used
6.1.3 error in procedure
6.1.4 erroneous understanding of procedures
6.1.5 others

6.2 Communication

6.2.1 erroneous understanding of spoken instructions
6.2.2 erroneous spoken instructions
6.2.3 erroneous understanding of written instructions
6.2.4 erroneous written instructions
6.2.5 others
6.3 System of marking components and instrumentation

6.3.1 erroneous marking
6.3.2 deficient/lacking marking
6.3.3 misunderstanding of marking(s)
6.3.4 others

6.4 Indications

6.4.1 erroneous measurement or position indication
6.4.2 erroneous interpretation of measurement
6.4.3 others

6.5 Training

6.5.1 deficiency in training
6.5.2 erroneous training
6.5.3 lack of experience
6.5.4 others

7 CORRECTIVE ACTIONS

7.1 Additional training
7.2 Alterations to procedures
7.3 Improvements to the organisation
7.4 Modifications to the components or systems
7.5 Modifications to the control room panels
7.6 Improvements to the marking systems
7.7 Improvements to the security systems
7.8 Improvements to the inspections and/or supervision
7.9 Others