

Nuclear Power Plant Operating Experience



from the IAEA/NEA
International Reporting System
for Operating Experience

2012–2014



IAEA

International Atomic Energy Agency



NEA

NUCLEAR ENERGY AGENCY

NUCLEAR POWER PLANT
OPERATING EXPERIENCE
FROM THE IAEA/NEA
INTERNATIONAL REPORTING SYSTEM
FOR OPERATING EXPERIENCE
2012–2014

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

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INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2018

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FOREWORD

Incident reporting is an important aspect of the operation and regulation of all public health and safety related industries. Diverse industries, such as aeronautics, chemicals, pharmaceuticals and explosives industries, depend on operating experience feedback to provide lessons learned about safety.

The International Reporting System for Operating Experience (IRS) is an essential element of the international operating experience feedback system for nuclear power plants. IRS reports contain information on events and important lessons learned that assist in reducing the recurrence of events at other plants. The IRS is jointly operated and managed by the Nuclear Energy Agency (NEA) — a semiautonomous body within the Organisation for Economic Co-operation and Development — and the IAEA. For the system to be fully effective, it is essential that national organizations allocate sufficient resources to enable the timely reporting of events important to safety and to share these events in the IRS database.

Five editions of this report have been previously published, in 2000, 2003, 2006, 2010 and 2012. These editions cover experience from 1996 to 1999, 1999 to 2002, 2002 to 2005, 2005 to 2008 and 2009 to 2011, respectively.

This sixth report on nuclear power plant operating experience from the IAEA–NEA IRS covers the 2012 to 2014 period and follows the format of the previous editions. This edition highlights important lessons learned from a review of the 258 event reports received from the participating countries during this period.

This report is intended to provide senior safety managers in regulatory bodies and in industry with information related to the safety of nuclear power plants to help those managers in their decision making roles.

EDITORIAL NOTE

Guidance provided here, describing good practices, represents expert opinion but does not constitute recommendations made on the basis of a consensus of Member States.

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SUMMARY

In the 2012–2014 period, Member States submitted 258 International Reporting System for Operating Experience (IRS) reports to the IAEA. In April 2015, the IAEA convened a consultants meeting in Vienna to prepare this publication. The participants reviewed the IRS events reported during the period and identified and selected those events with lessons and information of interest to the broader nuclear community. The selected events were grouped into common categories, for example ‘leaks and leakage’, or common plant activities, such as ‘maintenance’.

However, the safety significance of some reported events merited a review independent of the categories involved. The most safety significant event that occurred during the reporting period is addressed separately at the beginning of Section 2. The relevant categories of operating experience used later in this publication are provided in parentheses in the event description.

A brief overview of some of the event categories covered in this publication is presented below:

(a) Human performance

During the 2012–2014 period, a large number of events related to human performance were reported. Most of these events were the result of several contributing factors. It is important to thoroughly investigate the circumstances in which human error occurred during an event because other contributing factors or deeper safety culture issues will most likely be found.

For many of the reported events, human error either had significant consequences for the plant or exacerbated the course of the events. Some reported events did not have direct consequences but could have had serious consequences if a loss of core coolant or a seismic event had occurred.

Most of the events were triggered by errors made during non-routine maintenance activities performed either by licensee staff or contractors, rather than being triggered by the errors of regular operating personnel. The descriptions of these events indicate that maintenance or contract workers are not consistently informed of the potential consequences of their work on plant operation. Careful planning and close monitoring of the work done by maintenance staff and contractors could reduce the number of events significantly.

(b) Ageing management

There has been a notable increase in events related to ageing in recent years as plants get older and many are granted life extensions. In many cases, subsidiary

equipment and components that have been present since construction have not been considered for ageing issues and have therefore not been part of an ageing management programme. In one instance the primary service water system that delivered water to the heat sink for several systems was compromised, and in another instance the concrete structure that acted as a barrier to the radioactive material and protected personnel from ionizing radiation during normal and accident conditions had been degraded.

Important to nuclear and operational safety are (a) the identification of equipment and components that have been in service for many years and could be subject to degradation and (b) the inclusion of these items in an ageing management programme. Inspections, condition monitoring and replacement of degraded components are essential for safe, reliable operations.

(c) Leaks and leakage

Issues associated with leaks and leakage in nuclear power plants have started to increase recently and pose challenges to nuclear and operational safety. Leaks that are not corrected in a timely manner can deteriorate over time, compromise the reliability of safety systems and cause damage to associated equipment. In the reporting period, many leaks occurred in subsidiary systems not considered significant; however, these leaks affected other more important systems.

It is essential that leaks, particularly those that occur in important safety systems, are identified quickly and that the necessary actions are taken to remedy the situation in a timely manner to avoid adverse effects on operational safety and plant reliability.

(d) Seismic issues

The inspections and design reviews of seismically qualified structures continued to reveal deficiencies in the reporting period. Frequently, these deficiencies could be traced back to the original design. In other cases, design modifications or changes in operating practices invalidated the assumptions made by the original design. Constant efforts to reassess the seismic hazard and other external events should be maintained.

(e) Fire issues

Weaknesses in design and plant procedures contributed to several of the fire events reported during this period. A design deficiency in a main turbine lubricating oil heating system led to the single point failure of a heater and a

fire in one system. Another significant fire occurred during containment air testing because of a procedural violation that led to arcing in an electrical motor connection. Two other reported events were related to seismically induced fires. The lessons learned from these events demonstrate the importance of performing a systematic analysis of the secondary effects that may be induced by seismic events.

(f) Degraded cooling

Maintaining reactor core and spent fuel pool cooling capability is one of the main safety objectives in the operation of nuclear power plants. Two degraded cooling related events were reported during the 2012–2014 period. Both events took place during outages, and in both cases, operator response to the loss of core and spent fuel cooling would have been significantly facilitated if preapproved plans had existed to deal with outage situations in which the only equipment readily available for core cooling fails in service. Experience shows that this situation can and does happen in shutdown and outage conditions. The events reported during this period show that the time it takes for plant staff to find and implement an alternative cooling method can be rather long.

(g) Ineffective use of operating experience

A number of reported events might have been avoided if previous similar operating experience had been adequately considered. For several events, an assessment of external or internal operating experience had identified recommendations; however, the recommendations had either not been implemented at all or not been implemented in a timely manner. Also, a major contributor to some events was a failure to complete a thorough operating experience assessment. It is important for any organization dedicated to safety to take advantage of all available information to prevent the occurrence of safety significant events.

(h) Fuel damage or baffle jetting

Three of the events highlighted in this publication report the recurrence of baffle jetting related fuel failure. Baffle jetting fuel failures were first identified in the early 1970s, and heightened industry awareness of this issue over the following decades resulted in plant operators making modifications designed to eliminate this failure mechanism. However, several units that were identified with baffle jetting related damage during outages in the reporting period were thought to be resistant to baffle jetting because they incorporated an improved

design with additional bolting of the baffle plates. Moreover, two units that made baffle jetting related reports were at two of the plants previously believed to be at low risk for baffle jetting. The plant operator of one of those units stated that changes in the material properties of the baffle plates and bolting due to ageing mechanisms may have resulted in the gaps widening at the baffle joints over time and set up a condition in which baffle jetting could occur.

1. THE INTERNATIONAL REPORTING SYSTEM FOR OPERATING EXPERIENCE

1.1. WHAT IS THE INTERNATIONAL REPORTING SYSTEM FOR OPERATING EXPERIENCE?

In 1978, the Nuclear Energy Agency (NEA), a semiautonomous body within the Organisation for Economic Co-operation and Development (OECD), established an international system for exchanging information on safety related events in nuclear power plants. In 1981, OECD countries formally approved the operation of this system, called the Incident Reporting System (IRS), and in 1983 the IAEA extended the system to all Member States with a nuclear power programme. Since then, the IRS has been jointly operated by the IAEA and the NEA. However, with the creation in 1995 of the first comprehensive database on the IRS, the Advanced Incident Reporting System, the responsibility for handling information on events (including quality checking) was transferred to the IAEA.

In 2009, to reflect the evolution of the IRS to a system that included an expanded view and use of operating experience feedback, the name of the system was revised to the International Reporting System for Operating Experience. The system has, however, retained the abbreviation IRS.

The IRS, as a worldwide system, is designed to complement national schemes. The information reported is assessed, analysed and fed back to all interested parties in the nuclear industry to help prevent similar occurrences. The ultimate objective is to enhance the safety of nuclear power plants by reducing the frequency and severity of safety significant events. Currently, 33 countries with a nuclear power programme participate in the IRS (see Table 1).

The IRS is also helpful in identifying precursors. Precursors are conditions of apparently low safety significance that, if not properly monitored, have the potential to escalate into more serious incidents. Through the analysis of data reported to the IRS, the identification of these precursors can be facilitated and appropriate actions taken to mitigate their consequences. It is also important to detect and report on low level events and near misses as well as recurrent events.

Operating experience is therefore a key element of the ‘defence in depth’ philosophy, which is a fundamental building block for safety throughout the nuclear power industry.

The role of the IRS was reinforced by the obligation under Article 19 of the Convention on Nuclear Safety that Contracting Parties take the appropriate steps to ensure that:

“programmes to collect and analyse operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies”.

TABLE 1. COUNTRIES PARTICIPATING IN THE IRS

Argentina	Hungary	Russian Federation
Armenia	India	Slovakia
Belgium	Italy	Slovenia
Brazil	Japan	South Africa
Bulgaria	Korea, Republic of	Spain
Canada	Lithuania	Sweden
China	Mexico	Switzerland
Czech Republic	Netherlands	Ukraine
Finland	Pakistan	United Arab Emirates
France	Poland	United Kingdom of Great Britain and Northern Ireland
Germany	Romania	United States of America

1.2. WHAT ARE THE BENEFITS?

The IRS increases worldwide awareness of potential and actual problems in nuclear power plant operations. The heightened awareness generated by feedback from operating experience has resulted in numerous improvements to equipment, procedures and training in many nuclear power plants, thereby reducing the potential for subsequent failures that could result from unusual events.

The analysis of IRS reports can also assist in determining whether a particular event is generic or recurring in nature. Recurring events may reveal

several types of problem related to the safety of nuclear power plants. A recurring event is defined as one that has actual or potential safety significance, that is the same as or similar to a previous nuclear industry event (or events) and that has the same or similar causes as the previous event (or events).

The IRS database contains event reports that provide detailed descriptions and preliminary analyses of each event's causes that may be relevant to other plants. These analyses may lead to corrective action by plant management or regulatory authorities. IRS resources include the publishing of topical studies on events of particular interest. Recent studies have focused on topics such as external flooding and the effect of the failure of a non-safety system on plant safety.

Countries that participate in the IRS benefit from exchanging information related to the root cause analyses and lessons arising from incidents at nuclear power plants. Feedback on how to adequately remedy or avoid possible precursors is of paramount importance to operational safety. For example, abnormal pipe thinning in short piping bends that is not identified in time could eventually lead to a pipe break, which in turn could result in an accident.

Another potential use of IRS data is the application of operational feedback in the design of the next generation of nuclear power plants. Nuclear power plant operating experience has demonstrated that design modifications documented in IRS reports can have a significant effect on safety.

1.3. HOW CAN THE IRS BENEFIT DECISION MAKERS?

Decision makers in the nuclear operating organizations, regulatory bodies and technical support organizations around the world face a challenging environment that includes deregulation, privatization and other economic pressures in electricity markets. This environment forces decision makers to seek new strategies and manage risks and resources with the objective of achieving and maintaining high standards of safety. The IRS contributes by providing information on safety significant events from the global nuclear community.

In managing risks and resources, decision makers need credible and reliable systems information on which to base the prioritization of their programmes. To maintain an acceptable level of safety, decision makers need to receive early warning of deteriorating safety performance in the field. They also need to share their experience and lessons learned with others and thus make efficient use of their resources.

Regulators require the industry to report on actual or potential hazards so that the regulators can develop effective requirements, guides or standards that, when implemented, will limit the risk to the public.

The IRS is a global contact network and forum that enables safety experts around the world to share and review information on lessons arising from reported events. It can provide world experts with information on individual and generic issues of safety significance and advance information on deteriorating safety performance. The IRS can also be used, together with other databases, to prioritize those issues of safety significance that have been reported and identify areas in which further resources or research is appropriate.

1.4. HOW DOES THE IRS WORK?

1.4.1. Event reports

Each of the 33 member countries with an operating nuclear power plant designates a national IRS coordinator. Reporting to the IRS is based on the voluntary commitment of the participating countries. An event report is submitted to the IRS when the event is considered by the national coordinator to be of international interest. Events of safety significance and events from which lessons can be learned are reported according to the IRS guidelines. An event is defined as safety significant if it:

- (a) Is serious or important in terms of safety because of a significant actual or potential reduction in the plant's defence in depth;
- (b) Reveals important lessons learned that would help the international nuclear community prevent the event's recurrence as a safety significant event under aggravated conditions or avoid the occurrence of a serious or important event;
- (c) Is similar to an event previously reported to the IRS but highlights new, important lessons learned.

When information is considered time sensitive, a short preliminary report can be distributed within one month of the event. Subsequently, a main report is produced, and in some cases a follow-up report is generated and distributed when additional relevant information becomes available.

The main event report contains basic information, including the title and date of the event, the characteristics of the plant and an abstract. The main event report also includes a narrative description of the event, a preliminary safety assessment (the direct causes, consequences and implications), a root cause analysis, potential corrective actions, lessons learned and guide words containing

essential information that can be easily searched and retrieved. Often, a written description of the event is supported by graphics (diagrams of affected parts of the plant, etc.).

When an event or series of events indicates a generic problem, the national coordinator may produce a 'generic event report'.

1.4.2. Sharing information

Each IRS report becomes part of the web based IRS, which was created to facilitate data input and report availability and speed up access to information. Passwords are provided to users according to their access level to ensure a high level of security. Once a new report is posted on the web based system, the users are informed by email. The routine receipt and distribution of reports on incidents form the basis for in-depth studies on implications and remedies and assist in identifying safety issues common to nuclear power plants.

1.5. HOW IS THE IRS USED?

1.5.1. Topical studies

Topical studies are one component of IRS related activities. Such studies are intended to provide the basis for in-depth evaluations and identify topical or generic issues by a team of nuclear experts. These issues begin with a national assessment by the reporting country that is, when warranted, then studied in depth by experts at the international level.

1.5.2. Annual meetings

National coordinators meet each year to review the information received and the operation of the system in general. The committee of national coordinators selects for further analysis topics and reports of those events that it considers to be of particular safety interest to the international community. The conclusions of the committee are distributed to participating countries. A joint IAEA/NEA meeting to exchange information on unusual events is also held annually. These meetings serve to strengthen the mechanisms for the exchange of experience in the assessment of incidents and in improvements made to reduce the frequency of similar events.

1.5.3. Restricted access

Access to IRS reports is restricted. Because the system is designed to be of value mainly to technical experts working in the nuclear power field, the information reported is not intended for distribution to the general public. This restriction encourages openness within the nuclear community, including the disclosure of incident details and related plant actions.

1.5.4. Other systems

The IAEA and NEA also developed the International Nuclear and Radiological Event Scale in 1990. Its primary purpose is to facilitate communication and understanding among the nuclear community, the media and the public of the safety significance of events at nuclear installations. The scale was modified in 1992 to include any event associated with radioactive materials or radiation or the transport of those materials.

1.5.5. Other activities

Activities within the IRS extend beyond the exchange and feedback of event information. Both the NEA and the IAEA have assigned expert working groups that meet annually and discuss the safety relevance of such events, the regulatory perspective and the application of lessons learned.

1.6. WHAT HAS BEEN ACHIEVED?

There are now more than 4000 event reports within the system. Additional events are added at a rate of approximately eighty per year. The annual reporting rate since 1980 is shown in Fig. 1.

The reports are made available in a user friendly, web based system, with a full-text database and a powerful search engine to allow full-text searching. The enhanced capacity for data input, storage and access to written, numerical and graphical information has improved the reporting and subsequent analytical capabilities of the database and is making the IRS more effective in the enhancement of nuclear safety.

Over the years, the IRS has expanded from being primarily a vehicle for information exchange to becoming a source for analysis, in-depth discussions, generic studies and meetings for the exchange of information related to operating experience.

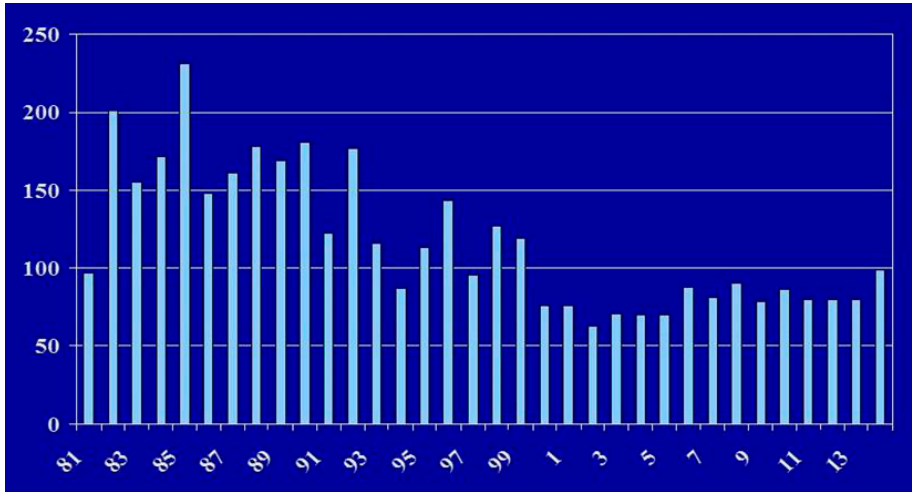


FIG. 1. Annual reporting rate since 1980.

2. EVENTS AND EXPERIENCE GAINED FROM THE IRS DURING THE REPORTING PERIOD

This section starts with a summary of the most safety significant event that occurred during the 2012–2014 period. Related categories of operating experience used later in the section are provided in parentheses.

The safety significant event took place at a two unit site. Unit 1 was shut down, cooled down and about one week into a refuelling outage. Unit 2 was operating at 100% power with no technical specification limitations in effect. One of Unit 1's outage activities involved the replacement of the main generator stator. The plant operator hired a contractor to lead this activity, and the main contractor subcontracted another company to carry out the project's heavy lift operation. The temporary heavy lift assembly being used for the main generator stator replacement collapsed while lifting the stator. The main generator stator weighed approximately 500 t (500 10³ kg). One contract worker was killed and several others were injured as a result of the collapse. The impact of the stator drop was felt throughout both plants. Substantial damage was sustained by electrical switch gear and mechanical equipment located on the level below the turbine deck. The complications caused by the stator drop and related equipment damage included the following:

- (a) The electrical switch gear was damaged, leading to a loss of off-site power at Unit 1. All emergency diesel generators started and re-energized vital loads as designed. Unit 1 off-site power was restored six days later.
- (b) The physical impact of the main generator stator drop on the turbine deck caused a vibration that induced a relay to trip a reactor coolant pump in Unit 2. This reactor coolant pump trip resulted in an automatic Unit 2 reactor trip.
- (c) The rupture of a 20 cm diameter fire main during the event caused a water leak, which sprayed on adjacent electrical equipment and resulted in a partial loss of off-site power, a loss of reactor coolant pumps and a switch-over to natural circulation cooling in Unit 2. (Experience with degraded cooling)
- (d) The fire water leak was not immediately secured by the operators on duty. It took over 30 minutes to secure the temporary fire water pump that was running and providing flow into the damaged system. It was later determined that procedural controls associated with the 1999 plant modification in which the temporary fire pump was installed were inadequate because they did not direct operators to turn the temporary pump off in the event of a fire system leak or rupture.

Unit 2 returned to operation about one month after this event, following repairs to the turbine building wall and the affected switch gear. Unit 1 restarted about four months after the event, following significant repairs to the turbine building and off-site power connections, as well as the implementation of measures to address the causes of the event. The plant operator determined that the direct cause of the event was a failure of the temporary heavy lift assembly. The plant operator also determined the root cause of the event to be an inadequate design by the heavy lift subcontractor, coupled with a failure to perform a load test of the special lift assembly. (Experience with design and installation; experience with modifications)

A similar assembly had been used for identical applications at other sites, and the subcontractor, contractor and plant operator accepted this experience in lieu of performing the load test required by plant procedures and industry standards. In fact, the temporary assembly used at Unit 1 differed slightly from the assembly that had been used successfully elsewhere. An inaccurate assumption made by a subcontractor masked the fact that one end of the crane structure would not actually be able to bear the load of the main generator stator lift. This error went unnoticed during reviews by the subcontractor and the contractor. (Experience with human performance)

The regulator conducted separate inspections at the site and identified several issues of varying safety significance. The inspectors determined that

the plant operator did not provide a sufficient level of oversight for the lift assembly design and load testing process and did not ensure that appropriately qualified personnel performed the necessary independent reviews of the design calculations required by site procedures. Therefore, as the plant operator had failed to follow the procedural requirements specified in the material handling programme, the regulator assigned one finding of substantial safety significance to each unit. (Experience with maintenance programmes)

Failure to rapidly secure all fire main water pumps resulted in the release of a large volume of water during the event (initially about 9500 L/min with two fire pumps running). The water leak revealed deficiencies in the site's flood mitigation measures. Contrary to the plant design, fire main water was able to migrate from the turbine building into the Unit 1 auxiliary building. Deficiencies with unsealed and degraded penetrations, unisolable floor drains and open ventilation ductwork were identified during the regulatory event response and as part of regulator prescribed flooding inspections post-Fukushima Daiichi nuclear power plant accident. The deficiencies were determined to affect plant performance in response to internal and external flooding and resulted in the regulator assigning one additional flooding related finding of substantial safety significance to each unit. (Experience with flooding)

As can be seen, the stator drop event involved elements of several of the experience categories found later in this report.

2.1. EXPERIENCE WITH HUMAN PERFORMANCE

During the 2012–2014 period, a large number of events related to human performance were reported; most had several contributing factors. For many of the reports presented, human error had either had significant consequences for the plant or had exacerbated the course of the events. Human performance is therefore a major factor to consider in the prevention of events.

At one plant, when the unit was at cold shutdown, an operator initiated the actuation of the emergency core cooling system by mistake because of the close proximity of the hand switches. The heavy water injection from the emergency core cooling system did not terminate on sensing the low level and pressure of the heavy water accumulator in the emergency core cooling system because the operator had previously changed the logic settings for the isolation of the accumulator while shutting down the unit. This lack of termination of the heavy water injection eventually led to the spillage of coolant into the reactor building and the disturbance of shutdown cooling flow through the core. An investigation revealed that in addition to the human error, procedural adherence issues contributed to the event.

In another event, a fault in a 4160 V cable caused a fire in a non-safety-related bus and, eventually, a reactor trip on degraded reactor coolant flow because voltage fluctuations affected the reactor coolant pumps. In this case, the operators did not immediately recognize the full impact of the fault on the electrical distribution system and plant equipment and later inadvertently re-energized the fault while attempting to restore the plant to a normal shutdown electrical alignment. This re-establishment of the fault caused a second fire and significant damage to surrounding equipment. The review of the event found that several equipment issues had been compounded by inadequate operator response due to training deficiencies.

At one plant, after an event in which operators noticed increasing dry well floor leakage, an investigation determined that the reactor vessel head studs had not been fully tensioned before startup operations. This situation was the result of errors made while operating the reactor vessel head stud tensioning equipment and in the validation process to ensure that the head was properly tensioned. The root causes of this event were determined to be (a) the lack of procedural guidance for the correct interpretation of the data used to validate that the reactor vessel head studs were properly tensioned and (b) the failure to provide proper training.

In another event, operations personnel were executing a routine safety system test. During the test, the operator misread the procedure and closed parallel motorized valves in the emergency coolant injection system while the unit was operating and therefore deactivated the system. The shift supervisor found the impairment and instructed the operator to back out of the test procedure and open the motorized valves. An investigation found that a new revision of the safety system test, which took into account that Unit 1 was no longer out of commission for refurbishment, had not been issued. Other contributing factors were a lack of questioning attitude by operating staff and the lack of barriers and administrative controls.

In another plant, during a refuelling outage, loss of off-site power occurred. One emergency diesel generator failed to start and another was out of service for scheduled maintenance, resulting in a plant blackout. The loss of off-site power was caused by human error during a protective relay test related to the main generator: the test personnel misunderstood the supervisor's instructions. The regulatory investigation of this event also revealed deeper safety culture issues.

One nuclear power plant was in a planned shutdown to change two pressure tubes. While the tasks were being carried out, a worker erroneously loosened the bolts of the wrong tube and caused a significant leak of heavy water from the primary heat transport system. The sequence of events revealed human error during planning, a lack of self-checking, communication and supervision issues, and indications of a lack of safety culture.

In the event described at the beginning of this section, while a stator was lifted for replacement, the temporary crane that was being used collapsed, and the stator fell. The dropped stator created a lot of debris before coming to rest on the hauler in the train bay (Fig. 2). Licensee staff had not checked the adequacy of the temporary crane before starting the job and instead relied on the contractor's statement that this equipment had been used previously to lift a load of the same weight. A design problem with the temporary crane was therefore missed. The investigation of the event revealed several industrial safety violations by the licensee and the contractors, as well as deficiencies in contractor supervision by the licensee.

Several other human performance related events, in addition to the ones described here, were reported. The most common contributing factors to human error were inadequate instructions or procedures, training deficiencies, self-checking issues, indications of a lack of safety culture and inadequate contractor supervision. Other less common contributing factors were a lack of questioning attitude, administrative controls or work preparation, as well as management issues.



FIG. 2. Dropped stator (partially shown on the right).

2.1.1. Safety significance

Overall, the events related to human performance had a wide range of consequences. Some of the events described here did not have direct consequences. However, the consequences could have been more serious in a different plant configuration or if a seismic event or a loss of core coolant, for example, had occurred concurrently with any of those events. In another case, the reactor was put in an unanalysed condition. Other events had consequences of various severity, including fire in the electrical equipment, electrical power supply trips, a reactor trip or scram, radioactive spills, control rod sticking issues, inadvertent activations of emergency core coolant, disturbances in shutdown cooling flow, losses of off-site power, containment breaches and physical injuries to workers, and — in one case — even death.

In some cases, the severity of the event was increased by the inadequate response of operating staff.

2.1.2. Lessons learned

Issues with the ergonomic design of equipment can and do cause human error. After one particular event, a transparent cover box was provided for some control room switches. Their colour was changed to reduce the potential for human error.

In most events with human performance elements, human error does not happen on its own. It is important to thoroughly investigate such errors because they will likely be associated with other contributing factors or reveal deeper safety culture issues within the organization. The errors may reoccur if these other factors are not corrected.

Most of the events submitted were triggered by non-routine maintenance activities done either by licensee staff or contractors, rather than errors by regular operations personnel. The descriptions of these events indicate that maintenance or contract workers are not consistently informed of the potential consequences of their work on plant operation. These consequences should be carefully explained during the preparation of the work to be performed.

Some of the events submitted indicate deficiencies in contractor supervision. Before work is started by a contractor, the licensee should do a thorough review of all relevant documentation related to the design, inspection and maintenance of the equipment that will be used, to ensure it is appropriate. The work procedure to be used by the contractor should also be verified, and the work should be closely monitored for quality and appropriate execution.

In one case, the event was caused not by an individual but by the behaviour of the plant management as a whole. This case shows the effect that management and organizational factors can have on nuclear safety.

2.2. EXPERIENCE WITH MODIFICATIONS

In the 2012–2014 period, the proportion of events caused by deficiencies in the modification process continued to be relatively high. The subject is rather broad and covers experiences from quite simple tasks — such as the change of a turbine speed set point — to more complex tasks, such as major mechanical, instrumental and electrical changes in the pressurizer spray system.

In one event that involved a turbine-driven auxiliary pump, the aim was to solve a vibration problem on the pump. The personnel lacked knowledge of or ignored the design requirements. The safety issue was only solved after persistent questioning from the regulator.

In another event, an error in the relay logic induced a loss of off-site power that also affected the twin unit site that was in refuelling outage. The cause of the faulty relay was an inadequate design input specification and insufficient control over vendor outsourcing. The vendor review of the modification of the relay was at a high level and did not identify the weakness. In addition, the licensee engineering personnel did not specify all of the critical design inputs required for proper operation of the new zone relay scheme. As a result, the design error was not detected during site review or post-modification testing.

Two similar relay malfunctions were detected in other reported events during separate emergency diesel generator periodic tests. The relay actuation was delayed up to a couple of minutes in the tests. The consequence would have been that the startup of emergency diesel generators would have been delayed or potentially disabled. The relays were commissioned one or two years before the deficiency was detected. The cause of the malfunction was manufacturer failures in a certain production batch. The quality problem was known by the vendor but not communicated to the nuclear power plant. Moreover, the vendor had introduced software based technology without notifying the nuclear power plant.

At one plant, a fault that affected measurements caused several malfunctions. The unit lost all indication of emergency feedwater flow, which led to an increased steam generator level. The cause was a short circuit in one cabinet. The licensee determined that a modification in the design of the emergency feedwater control system had been carried out in the past with the aim of eliminating the vulnerabilities of this system to single failure. However, the modified design failed to address this concern, and the design review failed to recognize these vulnerabilities.

At another plant, to fulfil new environmental qualification requirements for accident conditions, a modification project was performed during a refuelling outage. High temperature cables replaced existing cables on 18 level switches belonging to the reactor pressure vessel (RPV) instrumentation and on 14 level switches belonging to the main steam system. During startup, recurring ground faults occurred and the unit was shut down. Those ground faults were of common cause and were initiated by two independent contributions aimed initially at the betterment of the plant: the use of excessive thermal insulation and the use of unqualified heat shrinkable tubing. The direct cause behind the series of earth faults was that the heat shrinkable tubing installed in the RPV level switch junction boxes was not qualified for the high ambient temperature (270°C) experienced by these junction boxes. The heat shrinkable tubing exhibited a rapidly degrading insulation resistance, which eventually resulted in the ground faults. The abnormally high ambient temperature was caused by new thermal insulation installed on the level switches during a refuelling outage that fully covered the switch junction boxes (see Figs 3 and 4). The managers of the high temperature cable replacement project were not initially informed about these changes.

In another case, major modifications were made to the pressurizer system during an outage. The modifications included the replacement of the existing eight on-off spray valves by two adjustable spray valves. This modification involved the replacement of several auxiliary lines, together with the necessary modifications in the instrumentation and controls and the electrical systems.



FIG. 3. Improper insulation installation.



FIG. 4. Proper insulation installation.

During testing, a previously unidentified design failure was detected in the spray valve position controller logic. Analysis showed that it was impossible to determine from the test records whether the opening and closing pressure set point requirements of the relief valve were met.

At one plant, a capped nozzle in the low pressure injection system started to leak as a result of corrosion caused by the incorrect choice of material for the plant conditions. The cap was introduced because of changes during the construction of the plant (a valve that was not installed was replaced by this cap). This modification was a temporary solution, but a final solution was not introduced until the leak was discovered.

During a planned general overhaul at another plant, extensive damage to cable insulation in the measuring circuits was found. New cables were not available at the power plant, and there was no time for their procurement and replacement in the outage schedule. The maintenance personnel shortened the existing cables after removing the damaged portions and thus changed the location of the temperature sensors. The maintenance personnel did not communicate this modification to the plant outage management group. The maintenance personnel were not aware that at the designed location of temperature sensors, air from the reactor pit blew onto the sensors.

2.2.1. Safety significance

In one event, the licensee's configuration control failed. The licensee's attempt to permanently capture changes made in a temporary modification led to

an event. In several events, modifications intended to improve plant performance and safety actually introduced vulnerabilities that could have negatively affected or did negatively affect nuclear safety.

The lack of information from vendors, together with an inability at nuclear power plants to fully test or understand the characteristics in relays, has caused errors that had the potential to cause the loss of emergency power.

2.2.2. Lessons learned

The management of technical modifications should have the same systematic and rigorous review as the initial design and installation phase of a nuclear facility. The systematic and rigorous review of modifications should also be independent of the complexity of the modification.

Changes that might appear to be simple can have significant effects on safety. The lack of a systematic review of modifications could result in implementing modifications that do not meet the design requirements. Design modifications always require the application of a questioning attitude and careful analysis.

Poor communication between different disciplines and a lack of knowledge have been identified in events related to modifications. In some events, good work was done by each discipline, but without a thorough understanding of how these works interact. There are also examples of a lack of communication about the test results between different disciplines, such as test operators, the testing organization and the maintenance organizations. Supervisory processes that did not consistently identify weaknesses and faults were allowed to pass without a clear, analysed statement or an understanding of the reasons. In one event, a temporary solution was regarded as final or was forgotten within the organization. In one case, maintenance workers solved emergent problems without proper communication within the organization and were not aware of the importance of the component design bases. Instead of reporting the unsatisfactory technical conditions, the maintenance workers decided to use an available solution without sufficient risk assessment. Some of the events were due to a lack of communication and understanding between vendors and licensees when changes were made to products. It is important for licensees to ensure that vendors inform the licensee about any quality issues or changes to the design of safety related components. Licensees should also ensure that internal processes are in place to identify and understand all technical features in new or modified systems and components.

2.3. EXPERIENCE WITH DESIGN AND INSTALLATION

The reviewed events from 2012 to 2014 indicate that design deficiencies can exist for a long time before causing significant problems. Some events show the importance of a holistic view and understanding of the original design and the interaction between systems. The lessons identified highlight the importance of understanding the design basis. This understanding is vital for the successful operation of the plant during the original design life, as well as for the extension of the plant's life.

In another event, a load rejection test (at reactor power 43%) was performed to verify that the main turbine speed automatic limiter operated as designed after a power uprate. Speed automatic limiters should prevent the overspeed of the turbine and initiate a turbine trip at 107% of the turbine nominal speed (1926 rev./min). When the pressure of the main turbine first step is greater than 8.48 kg/cm², a reactor protection system trip signal is actuated if the reactor power is greater than 24.8%.

The load rejection event test was initiated by manually opening the main generator output breaker. As a result of the load rejection test, the main turbine increased its speed and the digital electrohydraulic control system, which is responsible for controlling the turbine speed, initiated the closure of the governor valves. The turbine speed increased to 107% of the nominal speed, and the speed automatic limiter initiated a turbine trip.

The pressure of the main turbine first step decreased below the permissible limit of 8.48 kg/cm², so the automatic reactor trip was inhibited and the reactor did not shut down as expected in response to the main turbine trip. The root cause of the event was determined to be inadequate design change analysis. The system design did not consider the opening of the main generator breaker in its protections and trip settings. Moreover, the design did not consider the effect of the fast response of the control system on the reactor protection system set points. The reactor did not trip as anticipated on turbine trip, but it did trip on high reactor vessel pressure.

One event raised concern about a potential control circuit design deficiency in motor operated valves that could result in incorrect valve position indication and therefore in the valve being in an incorrect position during a loss of coolant accident. In this event, a design deficiency in the operation of certain motor operated valves was observed. Several motor operated valves remained partially open after the initiation of an automatic isolation signal in response to a design based loss of coolant accident. The identified design deficiency, which involved the 'hammering' of closed valves, had been highlighted in 1985.

Many safety related instruments that perform essential safety functions and provide operations personnel with information used as the basis for actions to

ensure adequate core cooling, use sensing lines. In one event, several instrument sensing lines within multiple systems were detected as improperly sloped.

In another reported event, a pump shaft was broken in the fillet area, where the diameter changed. The cause was a fatigue fracture that resulted from a design deficiency in which a lack of rounded edges caused stress concentration.

One event included several corrosion related fatigue failures of shafts, caused by inadequate material properties. The shafts of all primary service water pumps (including spare pumps) were made of steel that did not meet material specifications. Moreover, the piping of the system had been replaced some years ago, and the selected materials changed the electrochemical conditions and the loading–vibration behaviour of the system. The material selection created galvanic couples that probably accelerated the corrosion.

During a planned general overhaul at one plant, several of the main steam line level switches of the float level type were discovered to be faulty. An investigation showed that the actuating magnets had weaknesses. One magnet was contaminated with metallic clips. Several switches had a misaligned magnet due to manufacturing deficiencies. Further investigations showed a deficiency in the verification process, which had failed to identify manufacturing deficiencies. The experience was communicated to the manufacturer, who changed the verification process.

2.3.1. Safety significance

Motor operated valves might not automatically resume operation once a power supply is restored. There is also a possibility that the position indicating lights would incorrectly indicate that the valves are fully closed when the actual valve position could be as much as 15% open. Primary containment isolation valves are susceptible to this situation.

One event had a high degree of common cause failure because of manufacturing deficiencies, but the operability had been verified by procedures at the site. This event had the potential for greater safety significance because of the reduced failure margins and the common cause.

The use of material yield stresses and concrete compressive strengths that are less conservative than specified in the plant's design and licensing basis in structural design calculations can result in inappropriate reductions in the safety margins inherent in the specification and code requirements.

One event occurred during the test phase that, although it had no actual consequences for plant safety, had the potential for safety impacts. Indicated deficiencies in the design and change analysis process could lead to other more safety significant events.

Improper instrument sensing line sloping can cause highly significant safety problems. One event with design deficiencies in instrument lines caused a high pressure coolant injection system to be declared inoperable.

2.3.2. Lessons learned

The reviewed events indicate that latent design deficiencies can remain unidentified for a long time after the commissioning of the plant and cause significant problems only after many years of operation.

Some events highlight the importance of the original design calculations for the whole period of design plant life, as well as in decision making and actions related to plant life extension.

In some cases, the work done to improve the functionality of safety components after the identification of design deficiencies introduced another deficiency.

Instrument sensing lines within multiple systems at one plant were detected to be improperly sloped. One problem in this case was the misinterpretation of the related construction procedure, which was missing the detail required to define the alignment of the sensing lines.

2.4. EXPERIENCE WITH FLOODING

Several safety significant events related to both internal and external flooding were reported in the 2012–2014 period. Even after the accident at the Fukushima Daiichi nuclear power plant and the subsequent safety reviews and inspections conducted worldwide (which often included plant walkdowns and design reassessments aimed at detecting vulnerabilities to flood hazards), deficiencies in flood barriers or in the design against flooding continue to be observed.

The 2011 Tohoku Pacific Ocean tsunami triggered three events other than at the Fukushima Daiichi nuclear power plant in which the pre-existing vulnerabilities (missing or deficient watertight barriers) led to the flooding of rooms that housed safety equipment, resulting in the loss of different safety systems. In another external flood event, heavy rainfall exceeded the capabilities of the drainage system of the plant. The event also revealed that this drainage system was performing some safety related functions, even though it was considered a non-safety-related system.

The event described at the beginning of this section, involving the collapse of the lift equipment used to hoist the main generator stator, had significant implications for the flood mitigation features of the plant. Although the licensee

had identified significant issues during the plant walkdowns carried out after the Fukushima Daiichi accident, these walkdowns had not identified all of the deficiencies in the flood barriers at the site. Condition inspections were conducted after the failure of the auxiliary building hatches, and the issues identified included degraded hatch seals due to ageing or deficient installation, numerous (over one hundred) unsealed conduit penetrations through flood barriers, unisolable floor drains (which could become a point for water entry if there were an external flood), unisolated ventilation ducts or abandoned unsealed piping that penetrated flood barriers.

In another event, a failure in the sewage system control circuits led to an internal flood in the room housing the fuel oil tank pumps and caused prolonged unavailability (i.e. more than one day) of one of the diesel generators. Another internal flood was caused by the rupture of a rubber bellow in a diesel generator cooling pipe. An error in the design pressure of the bellow technical specification was the cause of the bellow failure. The flood could have spread to different rooms because the plant flood safety analysis had not included the case of a double guillotine break in low energy piping, according to applicable standards (only a limited leak rate). As a result, the room's sump volume and draining capacity were insufficient (see Fig. 5).

In addition to these actual flooding events, other vulnerabilities in safety assessments or equipment design have been identified, like the underestimation of flooding frequencies due to external catastrophic dam failures or the leaks observed in refuelling water storage tanks. These leaks reveal a potential flood threat due to the large water inventory of these tanks and, in some cases, due to their close proximity to safety related equipment.



FIG. 5. Examples of internal flooding.

2.4.1. Safety significance

Both external and internal floods have the potential to disable redundant safety systems if the barriers separating different areas are missing or inadequate, if the flood design bases do not adequately cover all the flooding sources, or if some possible water intrusion pathways have not been addressed.

The plant power supply, particularly the emergency diesel generators, has proven to be particularly vulnerable during the observed flooding events when auxiliary equipment (like fuel transfer pumps, cooling system components or control power panels) becomes submerged. In one case, two out of the three diesel generators were out of service for four hours, and there was the potential for the third generator to be lost as well.

The safety significance is greater if the flood is connected to a major external event, like the 2011 Tohoku Pacific Ocean tsunami in Japan, as in this case the probability of the loss of off-site power or of experiencing other equipment damage before the flooded systems can be recovered is much greater. Furthermore, the response of the plant staff is made more difficult because of the complexity of such events.

2.4.2. Lessons learned

A plant flood safety assessment should address all possible flow paths (both outgoing flows in internal flood events and incoming flows in external flood events), especially through cable troughs, piping penetrations, ventilation conduits, drains and so forth. Rigorous inspections should be conducted to ensure the integrity of all seals and barriers.

The design assumptions used in internal flood safety assessments should be sufficiently conservative. In particular, the failure of expansion joints or other similar fittings in low energy piping should be taken into account.

A rigorous and independent review of the procurement documentation for safety components is essential to reveal design errors that, otherwise, may propagate from the original design documentation.

The assumptions used in the original design to estimate the frequency and severity of external flooding should be periodically reassessed, both because new data that did not exist may now be available and because there may be errors in the original assumptions. All uncertainties and assumptions included in the safety case supporting analyses should be clearly stated and made visible so that they can be challenged by independent assessments and reviews.

General non-nuclear standards may not be adequate for the design of drainage systems as some aspects of these systems may have safety related

functions, either protecting safety equipment from damage caused by rainwater or avoiding releases of contamination to the environment.

Inspection or ageing management programmes should consider a wide range of environmental and mechanical modes of degradation of the reactor water storage tanks, or any other tanks that contain large inventories of borated water. Experience shows that leaks occur in welds that are in or near the bottom of the tank and that those leaks are mainly caused by weld fabrication flaws, stress corrosion cracking or high stress low cycle fatigue.

2.5. EXPERIENCE WITH MAINTENANCE PROGRAMMES

Four significant event reports about maintenance programmes were made during the reporting period. All four events related to insufficient awareness of standard practices for compliance with seismic qualification; a lack of training or self-checking practices; poor maintenance, testing and surveillance practices; and a lack of procedure compliance or an incomplete procedure. The events concerned the torquing of bolts, the use of incorrect screws, the improper installation and welding of exhaust headers, oil leaks from flanges with loose bolts, oil levels, the maintenance and adjustment of circuit breakers, the degradation of the breaker isolation gas (sulphur hexafluoride (SF₆)) and the improper filling of desiccant in the compressed air dryer.

These events highlight the need to regularly make maintenance teams aware of standard practices. These events also illustrate the need for accurate maintenance procedures that include all the information required to ensure the safety of structures, systems and components (SSCs) and compensate for any lack of worker knowledge about standard practices. These events reveal the impact of inadequate maintenance practices that are inconsistent with the expected level of professionalism and highlight the need to update and correct such practices.

The events also highlight that vendor technical recommendations can contain information that might not be suitable for all designs or be applicable to all conditions under which the equipment is expected to function. The ineffective use of vendor technical recommendations can cause or contribute to operational transients, scrams and component failures. Maintenance activities, procedural controls and corrective actions can identify the ineffective use of vendor technical recommendations.

2.5.1. Safety significance

The proper maintenance of SSCs is essential to ensure their operability in all design conditions. Seismically qualified equipment and components need

special attention for maintenance. A loss of seismic qualification in many valves can hamper the function of SSCs in seismic events and reduce safety margins. In one case, as a result of improper maintenance, desiccant powder reached the valve of the safety related system and could have hampered its safety function.

2.5.2. Lessons learned

The events highlighted the importance of high maintenance quality, the completion of maintenance programmes that include all necessary components, the monitoring and supervision of maintenance activities, and training and communication to ensure that maintenance teams are aware of standard practices (proper screw identification, torque tightening, suitable tools, etc.). The events also show the need for accurate maintenance procedures that include all the information required to ensure seismic qualification is maintained. The need to improve the performance of maintenance activities and monitor the effectiveness of maintenance programmes was realized. It was recognized that utilities should ensure that vendors' recommendations are considered.

2.6. EXPERIENCE WITH AGEING MANAGEMENT

As nuclear power plants age, it has become more important to manage and mitigate the effects of equipment degradation due to ageing (see Fig. 6). The management of equipment and component ageing aims to ensure that the effects of degradation do not compromise the equipment's safety function.

Several significant events relating to ageing and the management of ageing were reported during the 2012–2014 period. These reports show that issues with ageing are not particular to certain types of equipment and that all components installed in a plant since construction can be susceptible to ageing problems.

Several issues concerning the degradation of capacitors have been experienced lately. Ageing causes the epoxy insulation to harden and crack over time, which degrades the capacitor and allows a high flow of current and excessive heating. The high current flow and excessive heating may result in premature circuit damage or malfunction, which could compromise important safety systems.

Indications of the degradation of main cooling water system pipelines were found during surveys performed on one system. The pipelines were made of three components. Their leak-tightness was ensured by a steel liner with a wall thickness of 4 mm. The steel liner was clad with a layer of concrete, both on the inside and the outside, to prevent corrosion. The outer concrete had a steel reinforcement and provided additional component stability. The degradation

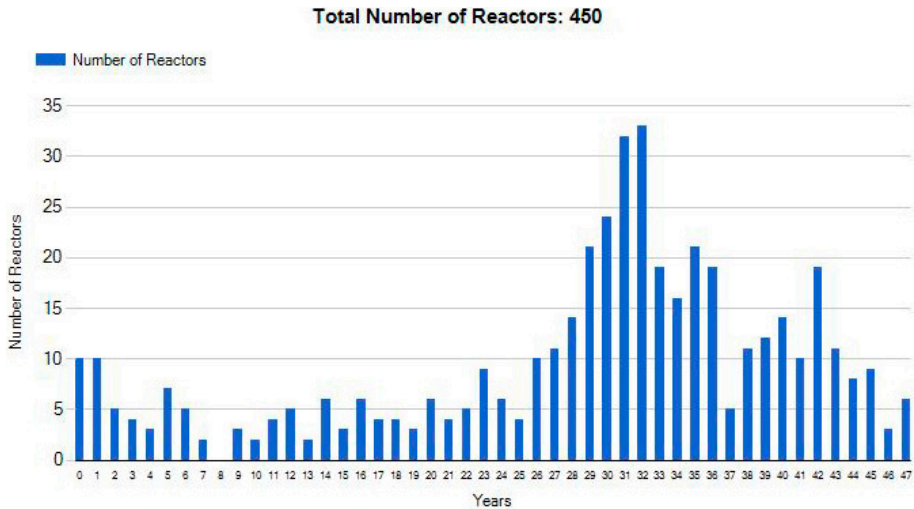


FIG. 6. Number of reactors by age (in years).

of the cooling water intake piping was found to be caused by the ageing of the concrete and steel layers as well as by deficiencies in the original design and construction of the pipelines.

Several plants discovered conditions that either increased battery design loads or decreased rated battery capacity so that the battery no longer met the design basis. The plants that identified these issues did not recognize the need to ensure the expected life of Class 1E batteries appropriately by accounting for the sizing requirements and post-accident direct current loading assumptions that were in the design basis documents.

Weathering and the intrusion of water over time can decrease the effective life of structures and components. In one instance, water contributed to the accelerated ageing of the concrete structures of the spent fuel storage system. Water entered cracks and crevices around the anchor bolt block out holes in the concrete structure and, when subjected to freezing temperatures, generated mechanical forces that produced cracks in the concrete.

Other significant events concerned three shaft breaks in the pumps of the primary service water system. Two breaks occurred in Unit 1 and one in Unit 2 on the same site. All breaks occurred in a two year period. The primary service water system ensures the transfer of heat from the fuel elements either in the reactor vessel or in the storage pool when the temperature of the primary circuit is below the minimum required for heat removal by the steam generators. The affected pumps had been in operation since the units were commissioned. No consideration had been given to the life expectancy of these shafts, and they were

not part of an ageing management programme or material condition inspection process.

2.6.1. Safety significance

The degradation of equipment and components due to ageing can significantly affect nuclear safety and plant reliability. Although the events mentioned here had no major safety consequences, a more significant event would have been possible if the issues had not been identified and addressed in a timely manner.

Capacitors are used in many safety systems and logic circuits. The failure of these capacitors could have many consequences for the safe operation of a plant.

The degradation of cooling water intake piping could cause a reactor trip and possibly degrade the core cooling should the pipeline fail as the temperatures in the system rise steeply in a short period of time.

As a result of the life reduction of Class 1E batteries, certain technical specification testing frequencies, specifically those associated with performance or modified performance discharge testing, were not sufficient to meet the requirements.

If remedial actions had not been taken to correct the deficiencies in the concrete structure of the spent fuel pool, the ability of the concrete structure to protect the spent fuel and protect personnel from ionizing radiation during normal and accident conditions could have been compromised.

Failures of pump shafts affected the same system, and there was a slightly increased probability of a simultaneous failure of the shafts of other operating pumps in the primary service water. However, normally, one primary service water pump is operating; a second pump starts only when there is increased demand for service water. No consideration had been given to the life expectancy of these shafts, and they were not part of an ageing management programme or material condition inspection process.

2.6.2. Lessons learned

Issues regarding the ageing of components and equipment have increased in recent years as plants age. Often, subsidiary equipment and components that have been present since construction have not been considered in ageing management programmes. It is important to recognize and identify systems, equipment and components, particularly those that are part of a safety system or that support a safety system, which may be subject to degradation over time. Changes in the assumed normal environment over a long time can accelerate the ageing process.

Inspections, condition monitoring and the replacement of degraded components are essential for safe, reliable operations and nuclear safety.

Periodic preventative maintenance programmes for critical equipment and components should take into account the time spent in storage as well as the time in service to address the adverse effects of ageing.

Buried pipes, in particular, can be subject to ageing. Soil (de)stabilization and changes in the load bearing capacity of the pipeline's bed should be monitored from the beginning of operation, using the appropriate monitoring systems, and should continue to be monitored and included in an ageing management programme. In many plants, remedial work has been undertaken to exhume these pipes so that they are above ground or to re-site them in a trench where they can be inspected.

Identifying potential moisture entry points, such as cracks, crevices, through-wall penetrations and joints in buildings and structures, can facilitate the incorporation of gaskets and sealing materials into both the design and maintenance of important buildings to minimize the premature degradation of structures and components important to safety.

The failed shafts of the primary service water pumps were made of steel and were highly affected by corrosion. Materials that do not fulfil their specifications can cause failures of components even after decades of inconspicuous operation. Especially at risk are components without a comprehensive and documented analysis of the batch of material used for their production. It is important for maintenance inspection intervals to be conservatively reassessed as components and equipment age, particularly after any major system change or modification.

2.7. EXPERIENCE WITH LEAKS AND LEAKAGE

During the reporting period, the number of events reported to the IRS database concerning leaks and leakage increased. Leaks and leakage even for short periods of time can compromise the reliability of safety systems, and not repairing those systems can be an indication of poor safety performance.

Three of the most significant events included a leak of primary coolant that proved problematic to identify. A plant shutdown was initiated to assist with the identification and enable repairs to be made. The source of the leak was discovered to be a crack in the upper housing assembly of a control rod drive mechanism caused by transgranular stress corrosion cracking. Another event concerned a leak from a flexible pipe onto the lubricating oil system of a diesel generator due to fatigue cracking of the welded ends; this leak rendered the diesel generator unavailable and increased the risk of a fire. Similarly, an oil leak from

the oil pump of a reactor coolant pump motor occurred and was followed by a fire and degradation of the reactor coolant pump.

Similar leakage events included steam leaks from an emergency turbine-driven feedwater pump, leaks in a heat exchanger, the leakage of heavy water from a failed fuel sample line, leakage across main steam isolation valves in excess of the technical specifications, and a leak from the cover of an in-core monitoring system detector. In another case, a hydrogen leak from a generator led to a reactor trip and the penetration of a radioactive medium to a system outside the boundary of the radiologically controlled area.

All these events were attributable to deficiencies in plant maintenance inspection programmes, poor maintenance practices and the inadequate consideration or assessment of risks.

2.7.1. Safety significance

Leaks from subsidiary equipment or components that are not recognized as part of the main system and are not subject to the same rigorous maintenance procedures or inspection processes in some cases resulted in either a reactor trip or a reduction in reactor power. One event resulted in the unavailability of both emergency feedwater system turbine-driven pumps due to a steam leak. Another event resulted in the unavailability of a diesel generator due to a leak from an oil pump.

A leak from the heat exchanger tubes (Fig. 7) in a hot water system required to maintain building temperatures could have led to loss of the reactor coolant system's boron make-up function if not dealt with quickly.



FIG. 7. Heat exchanger tubes, showing origin of leak.

In another event, a leak from a copper coupling attached to a PVC pipe on a potable water system caused water to flow through various levels of an electrical building. The leak resulted in an insulation fault on a 48 V direct current power supply and distribution switchboard and in malfunctions on reactor protection system sensors. A reactor shutdown was initiated, and the reactor was cooled by the residual heat removal system.

In another event, an oil leak resulted in a fire, the degradation of a reactor coolant pump and, finally, its unavailability. The complete degradation of the reactor coolant pump seal produced debris. Foreign materials reached the seal return isolation valve, prevented the complete closure of the valve and caused it to leak. The resulting debris produced foreign material throughout the chemical and volume control system. Further degradation of the seal(s) could have resulted in a loss of coolant accident.

Another significant event took place after work was performed on a supply system for the primary circuit. A leak in a valve resulted in the radioactive medium being present outside of the radiologically controlled area.

2.7.2. Lessons learned

Leaks can occur in many different systems and may involve various fluids or gases. In some cases, leaks occurred on subsidiary parts not considered important or part of the main system and therefore not subject to the same maintenance procedures or inspection processes (e.g. the flexible hose of the diesel generator lubricating oil system). This hose was subject to a limit of three years' operation, but no record was kept of how long it had been in service, and it was not included in the maintenance inspection programme.

The unavailability of the turbine-driven pumps for the emergency feedwater system was caused by deficiencies in maintenance processes and procedures. The failed flanges had not been aligned in accordance with standard maintenance practices. Similarly, events concerning the leak from the cover of the in-core monitoring system and the deficiencies identified with the heat exchangers can all be attributed to these same errors.

Rigorous risk assessments need to be performed when modifications are being carried out to ensure that any failure that may result in a leak poses no threat to other systems and equipment in the vicinity. This was the case with the potable water network that caused flooding throughout the electrical building. The modification was carried out without an adequate risk assessment because the potable water system was not identified as having safety significance.

Preventative maintenance inspection programmes, plant walkdowns and similar activities need to be robust and performed regularly to ensure the early detection of leaks. Subsidiary equipment and parts should also be considered if

they are susceptible to leakage that could compromise plant safety systems and equipment.

The use of subcontractors to perform maintenance activities has increased in recent years. It is essential that utilities ensure that all personnel, including contractors and subcontractors, who perform these activities understand that the quality of their work could have a direct impact on nuclear safety.

2.8. EXPERIENCE WITH SEISMIC ISSUES

Inspections and reviews of seismically qualified SSCs continue to reveal design vulnerabilities and some deficient operating practices that could affect SSC response during and after an earthquake.

Examples of vulnerabilities in the design include (a) the incorrect seismic analysis of the boron standby liquid control tank, (b) the incorrect design of seismically qualified cranes and special lifting devices, (c) deviation from the steel and concrete parameters prescribed by approved standards in the design of safety related SSCs and (d) the design of seismic monitoring instrumentation without an adequate power supply.

The events that highlighted vulnerabilities arising from deficient operating practices included (a) a full power operation with non-seismically qualified piping that was not isolated from a seismically qualified, safety related refuelling water storage tank, (b) the attachment of the floating roof of a seismically qualified tank to non-seismically qualified anchor points, (c) the inadequate testing and calibration practices of seismic monitoring instrumentation and (d) a failure to appropriately inspect special lifting devices.

One event showed an example of unexpected interaction between a seismic event and the later failure of an emergency diesel generator. The earthquake induced a fire involving electrical equipment and caused electrical disturbances that inadvertently damaged electrical circuits related to one of the diesel generators. The issue was discovered one month later, when the diesel generator failed to start during a test.

2.8.1. Safety significance

In the case involving the boron standby liquid control tank, the vulnerabilities in the design and seismic analysis, if not addressed and mitigated, could have resulted in the failure of the supports of the piping in the safety systems or safety equipment.

The event in which the inappropriate inspection of lifting devices, as well as inappropriate operational practice in lifting heavy loads, was reported could have

damaged neighbouring safety equipment during an earthquake and prevented the safe shutdown of the plant.

2.8.2. Lessons learned

The alignment of non-qualified and qualified systems may invalidate the original assumptions of the seismic design. If licensees choose to use temporary compensatory measures, these measures should be evaluated and justified.

The calculations for seismically qualified structural steel and concrete should use values for material yield stresses and concrete compressive strengths that comply with the standards referenced in the licensing and design bases. Special attention should be paid to avoid non-conservative extrapolations of concrete compressive strengths based on the concrete age hardening over time.

For cranes and special lifting devices, it should be verified that the calculations and the procedures used for load testing, dimensional testing and non-destructive examinations of welds and other critical parts satisfy the codes and standards referenced in the applicable licensing and design bases.

Operating practices should be consistent with the assumptions used for the seismic qualification of tanks, such as the boron standby liquid control tank, particularly regarding the volume of water contained by the tank during power operation.

Many nuclear power plants around the world use seismic monitoring instrumentation to decide whether or not to shut down the reactor after an earthquake or to make other decisions related to emergency planning and response. These systems should be provided with uninterruptible power supplies to avoid losing their readings when emergency diesel generators start and load during a loss of off-site power.

Experience shows that seismically induced fires are relatively likely: at least two events were reported in the 2012–2014 period. External event safety assessments should systematically consider this interaction.

2.9. EXPERIENCE WITH FIRE ISSUES

The impact of fires on nuclear power plants can be characterized by the possibility of simultaneous multiple equipment damage, which could challenge plant design provisions. In general, nuclear power plants are designed to withstand severe fire events. However, operational experience shows that there can be hidden threats and secondary effects not fully covered by design provisions. Therefore, there is a need to monitor and thoroughly analyse fire related operational events to identify possible weak points in the plant design.

An event was reported that involved a fire in a turbine hall due to the overheating of oil in a turbine lubricating oil purification system. One of three heating elements in the heater failed to switch off when demanded, which led to the oil exceeding its self-ignition temperature and therefore to the fire. The contributing problems were the absence of a high temperature alarm in the control room and a design weakness that allowed a single component failure to result in a fire.

At another facility, a major fire was caused by the inappropriate implementation of a containment air test procedure. According to the procedure, during the test there should be no transient fire loads present and no equipment electrically connected in containment. A violation of the procedure (one vacuum cleaner was energized) led to a fire in one of the electrical motors. The investigation of the event showed that the operation and supervisory management and the operational staff had not fully understood the challenge for the organization because the containment air test had been rescheduled. Organizational weaknesses, including unclear responsibilities, deficient communication and a lack of strategic planning, contributed to the event. Although the event did not result in any consequences for the public or environment, did not lead to fuel damage and did not affect safety systems, it did lead to a large fire that continued for 30 minutes and had the potential for fire or smoke propagation to safety related compartments. In addition, as a result of the fire, the whole containment was covered with black soot particles (Fig. 8) that

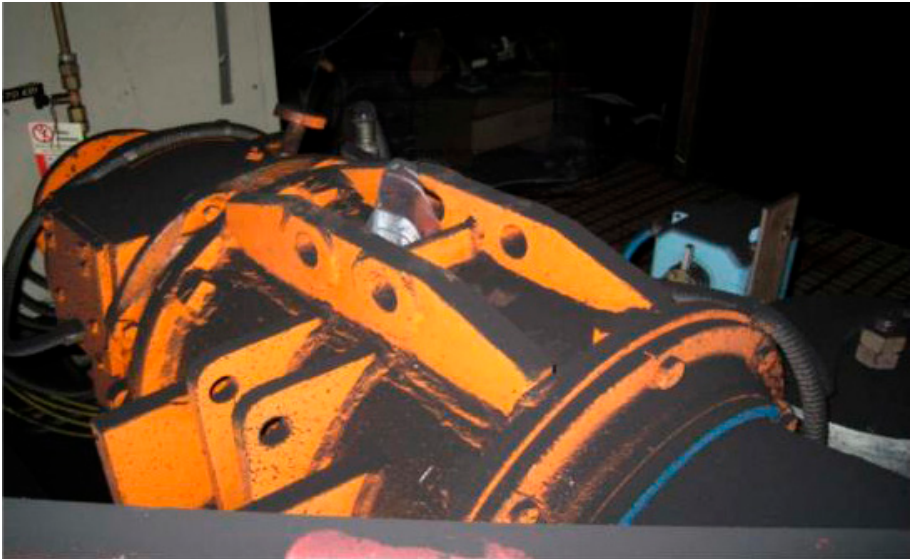


FIG. 8. Pump covered with soot after a fire during a containment air test.

resulted in an extended production loss during the cleanup and restoration of the containment and all the components located inside it.

Another event report indicated that the reported fire originated from a failure in the high voltage power supply switchboard during an earthquake. This event shows the coupling of a seismic event with a fire risk, which was also demonstrated by the seismically induced fire event at another nuclear power plant. In addition, the event shows that it is essential to extensively check the integrity of electrical circuits after a fire event, as electrical faults can easily propagate and cause damage in circuits apparently isolated from the origin of the fault.

Some facilities performed a self-assessment of their fire protection programmes and reported a potential for circuit failure induced secondary fire events or heat damage to equipment. It was determined that there may be specific unanalysed conditions related to existing deterministic fire protection programme requirements and analysis requirements involving ammeter circuits.

One reported event related to the spurious activation of a fire protection spray system. The event involved the double occurrence of spray directly resulting from a short circuit of the cable type thermal detector loop. The event was caused by a defect on the main transformer spray control system, which was therefore prone to triggering the spray in error. The main transformer spray control logic of one unit had a control mode that would immediately trigger the spray when the cable thermal detectors detected fire. This kind of control logic requires high equipment reliability to avoid a single failure leading to the spurious operation of the fire protection spray.

2.9.1. Safety significance

In general, the fire related events reported for the 2012–2014 period did not lead to safety significant equipment failures. However, the observed problems are considered to be potentially safety significant under certain conditions.

In particular, fires in turbine halls and containment buildings can lead to fire propagation within plant compartments, which could challenge plant safety. In addition, it was observed that there could be some effects, not systematically analysed, such as seismically induced fires or circuit failure induced secondary fire events, which could lead to the underestimation of the fire risk for a given facility.

2.9.2. Lessons learned

The lessons learned from the reported events indicate some measures and proposals aimed at decreasing the risks due to fires and related issues. Some selected lessons are described below:

- (a) Fire protection systems with one-way triggering control mode require extremely high equipment reliability, since a single fault may lead to spurious operation. The other way to resolve the problem is to avoid a one-way triggering control mode in fire protection systems.
- (b) Seismically induced fires can lead to additional failures in redundant systems. It is important to systematically analyse seismically induced fires through safety assessment (deterministic and probabilistic) and provide protective measures when necessary.
- (c) As shown by the event connected with the containment air test, it is important to ensure that there are no transient fire loads or energized equipment in containment during containment air test implementation.
- (d) As illustrated by the event connected with the fire in the turbine lubricating oil purification system, it is important to provide a remote high temperature alarm for such oil systems and procedures for control room operators.

2.10. EXPERIENCE WITH DEGRADED COOLING

Maintaining core and spent fuel pool cooling capability is one of the main safety objectives in the operation of nuclear power plants. Two degraded cooling events were reported during the 2012–2014 period. Both events took place during outages.

In the first event, during a refuelling outage at a pressurized water reactor nuclear power plant, a worker noticed smoke coming from the ‘A’ component cooling water system pump. Operators responded by securing both the ‘A’ component cooling water pump and the related reactor heat removal system pump ‘A’. This action resulted in a loss of forced cooling to the spent fuel pool and the fuel still inside the reactor vessel. In response to the event, core refuelling operations were stopped, temperature monitoring in both the spent fuel pool and the refuelling cavity was started, and the containment was isolated. At the time of the event the ‘B’ component cooling water train and a third common component cooling water pump (shared between the two trains) were both unavailable because of maintenance. The licensee had several methods available to restore the spent fuel pool and reactor cavity cooling. However, it took operations personnel

about seven hours before forced cooling to the reactor and spent fuel pool was restored through those available methods.

The second loss of cooling event occurred in a gas cooled reactor during an outage. The failure of a gas circulator component placed the plant in a condition outside of that allowed by technical specifications. The associated action condition required that action be taken to restore any plant declared temporarily unavailable and that the affected plant be returned to operable status. Senior outage personnel were not informed of this situation, and delays were incurred in returning the temporarily unavailable systems to operable status. Meanwhile, plant workers and the outage control centre continued to focus solely on the repair of the failed gas circulator component, and in doing so unnecessarily delayed the restoration of core cooling by failing to explore alternative means. This decision was compounded by difficulty in identifying the gas circulator fault, which prolonged the repair and led to a more extended period of inoperability. The next day an operational decision meeting was held to determine the best alternate method to restore cooling. The gas circulator was restored the following day.

2.10.1. Safety significance

These events had no significant impact on safety. Temperature increases measured in the spent fuel pool and refuelling cavity did not approach the established safety limits or technical specifications in either event. However, the longer it takes to restore core or spent fuel cooling, the greater the probability of exceeding an established safety limit.

2.10.2. Lessons learned

In both cases, operator response would have been significantly facilitated if preapproved plans had existed to deal with situations in which the only equipment available for core cooling fails in service. Experience shows that this situation can happen in shutdown and outage conditions, and that the time it takes for plant staff to find and implement an alternative solution can be rather long. The time available before fuel integrity is compromised depends on several factors but could be relatively short under certain conditions. Plant operators need to establish maintenance strategies that prevent the removal of a common or shared pump from service when one full train is to be made inoperable.

2.11. EXPERIENCE WITH RADIATION PROTECTION AND CONTAMINATION

Several events involving personnel exposure and the release of radioactive liquid outside the radiologically controlled area and to the environment were reported between 2012 and 2014. These events were the result of a number of factors, including non-adherence to radiation protection procedures and practices, weaknesses in handling unidentified objects in a radiologically controlled area, failure to recognize the radiological risks associated with plant tasks, inadequate radiation surveys, poor coordination between operations and health physics staff, absence of surveillance and maintenance programmes, delays in the repair of SSCs meant to handle and contain the radioactive liquids, and the non-usage of human error prevention tools during safety critical activities.

Three events resulted in workers being exposed to radiation beyond annual statutory dose limits or annual dose constraints.

One event occurred when a diver removed an unidentified object (an irradiated piece of guide tube from reactor core instrumentation) from the bottom of a pool in the fuel handling building. The diver, working underwater, did not perceive the alarms of the dosimeters he wore. Other causes of the event were an inadequate radiation survey inside the pool before the start of a planned job and weaknesses in handling unidentified material in a radiologically controlled area.

The second event occurred during the installation of a hook-up provision to inject water from outside the reactor building into the calandria (a low pressure tank containing the moderator) in a pressurized heavy water reactor (a post-Fukushima Daiichi accident modification). During the modification, openings to the calandria (which contained tritiated heavy water) were not temporarily sealed with rubber bungs. This lack of sealing caused a high tritium-in-air concentration at the job location and a worker to be internally exposed to tritium. The event was the result of a number of factors, including failure to assess the radiological risk if rubber bungs were not used, inadequate radiological monitoring during the modification, the non-use of protective equipment by personnel and poor coordination between operations and health physics personnel.

The third radiation exposure event occurred during the transfer of a canister containing cobalt self-powered neutron detector from the spent fuel pool to a transportation flask. This event highlighted the need to use human error prevention tools (self-check, peer check, verification, supervision, etc.) during the conduct of activities with a potential for radiological risk. Before the transfer, the canister top cover and its body were not properly secured by tightening the screws. As a result, the canister body containing the cobalt self-powered neutron detector fell inside the transportation flask during the transfer, but the canister's

top cover remained attached to the electromagnet. When the electromagnet was later retracted, the canister top cover plate, with loose cobalt contamination, became unshielded and caused a high radiation field at the work location and the exposure of a worker. The operator had noticed the dropping of the canister through a monitor but misinterpreted this event as resulting from the de-energization of the electromagnet.

In several events, the leakage of liquid effluents from SSCs (such as tanks, containers, buried or overground piping, dyke area floor drain valves, or active sump liners) resulted in the release of radioactivity outside the radiologically controlled areas and to the soil, the groundwater and the environment. These events occurred because the SSCs meant to handle and contain the radioactive materials were not in the desired condition or configuration because of the absence or delay of surveillance and maintenance activities. The detection of groundwater contamination may indicate a loss of integrity in the SSCs and a need for action to prevent the further spread of contamination. Also, the maintenance or repair of SSCs handling radioactive liquids should be prioritized according to the outcome of a risk assessment.

2.11.1. Safety significance

The inadequate assessment of potential radiological risks associated with various working activities and inappropriate planning led to an excess of statutory exposure limits and potential radiological consequences.

The release of radioactive liquids outside radiologically controlled areas and to the environment resulted in undue internal and external radiation exposure.

2.11.2. Lessons learned

The lessons learned from the radiological events include:

- (a) No material should be amended in a radiologically controlled area without suitable evaluation and proper control. Weaknesses in handling unidentified materials in radiologically controlled areas may lead to significant exposure. Nuclear power plants should have an approved procedure for handling such materials.
- (b) Personnel dosimeter alarms might not be audible to workers or divers when they are performing underwater jobs. Hence, the workers or divers who carry out underwater jobs should be provided with suitable devices to ensure that they are alerted of a higher than permissible radiation dose rate or dose.

- (c) The potential radiological risk associated with a task should be assessed thoroughly in advance. This assessment should consider possible human error and equipment failures. The system or procedure for the task should be commensurate with the risk.
- (d) The SSCs that are meant to handle, contain and provide barriers for radioactive liquids should be included in plant inspection, surveillance and maintenance programmes and always maintained in the desired configuration or condition through the timely execution of these programmes. The time for any necessary repair of these SSCs should be decided on the basis of a risk assessment.
- (e) Human error prevention tools (self-check, peer check, verification, supervision, etc.) should be extensively used whenever plant activities have the potential for radiological risks.

2.12. EXPERIENCE WITH INEFFECTIVE USE OF OPERATING EXPERIENCE: RECURRING EVENTS

The events discussed below illustrate that events recur when previously gained operating experience applicable to the same or similar components is not adequately considered. The lessons from these recurring events show that the analysis of direct and root causes based solely on an investigation of the individual event, without considering the past operating experience, will not always identify the underlying cause of the problem and thus prevent recurrence.

One event involved the potential draining of a spent fuel storage pool due to the absence of a siphon breaker on its cooling circuit line. Following an international organization recommendation on the role of fuel pool siphon breaker devices in late 2011, a plant operator decided to carry out a functional test of the siphon breakers on the discharge line of the spent fuel cooling system. During preparations for the test, it was noticed that the siphon breakers — which should have each had a 20 mm diameter orifice — were not present on the pipes submerged in the spent fuel pools of two units. The compliance inspection conducted before commissioning had not identified these discrepancies, and thus this condition had existed since construction. Siphon breakers on the fuel pool cooling system discharge lines of the units were installed within approximately a month of discovering the non-compliance.

The second event involved the valve stem failure of a primary system loop bypass valve due in part to the ineffective processing of operating experience. During unit startup, operations personnel identified a primary loop bypass flow indicator that showed no flow. During troubleshooting, a valve in the primary system loop bypass was found to have failed in the backseat (fully open) position.

Radiography subsequently determined that the valve wedge had separated from the valve stem and was stuck between the body seats, blocking the flow. The direct cause of the failure was the thermal ageing embrittlement of stems made from precipitation-hardened martensitic stainless steel. From the failure analysis report on the bypass valve stem, it appears that stress corrosion cracking was the failure mode. The stress required for this cracking mechanism is much less than yield, so any amount of back seating can result in enough stress to drive stress corrosion cracking. Operators did not have practical training on the plant backseat policy or that certain valve stem materials were susceptible to failure as a result of this action. The practice of back seating valves should only be considered in appropriate circumstances.

The third event involved a leak in the potable water system of a plant, leading to internal flooding on various levels of the electrical building. The water caused an insulation fault on the 48 V direct current power supply and distribution switchboard for train 'A', as well as malfunctions on sensors that were part of the reactor protection system. The leak was caused by the failure of a copper coupling that was screwed onto a PVC pipe that had been added to the potable water system, without a bracket, to provide water supply to a basin and water heater. No risk analysis had been carried out on this fragile configuration to assess the internal flooding risk, as such an analysis was not required at the time by regulations for systems that were not involved in reactor operation. Despite various events of similar or identical origin in 2011, the licensee did not reassess the risk of internal flooding at the facility.

The fourth event involved continuing issues of degradation of the neutron-absorbing material in spent fuel storage racks. One plant incorrectly used a combined areal density of panels to determine compliance with the technical specifications. On examination, the areal density of a single panel was found to be below the value assumed in the licensing basis, which meant that the plant was not in compliance with its technical specifications. Similar issues have occurred at other plants, again resulting in non-compliance with the technical specifications.

The last event involved a reactor shutdown caused by the gas locking of primary heat transport feed pumps. During unit operation at full power, a failure occurred in the yoke sleeve of an isolation valve in the instrument air pipe connected to a bleed filter, and the valve opened, resulting in air ingress into the primary heat transport system through the purification circuit. This air ingress resulted in the gas locking of both primary heat transport feed pumps. Several attempts were made to restart the feed pumps, but because of the continuing ingress of air supply to the suction header, the pumps remained gas locked. The event investigation revealed that the plant design did not require a permanent physical connection between instrument air and the bleed filter. From past

experience, it was well understood by plant personnel that the bellow seal globe valves had an inherent design weakness that could result in yoke sleeve failure when excessive force was applied during the closure of these valves. With this in mind, an instruction at the plant stated that the use of torque wrenches should be considered for the operation of these valves, but the investigation showed that this recommendation was not put into practice. Also, the available operating experience with these types of valves was not being used to assess the risk of using these valves in various systems.

2.12.1. Safety significance

Two of these events were considered to have minor safety significance. However, the event involving the potential draining of the spent fuel pool, under a different set of initial conditions, could have caused uncontrolled drainage that could have led to the uncovering of the stored fuel assemblies. Also, the event involving internal flooding illustrated that safety risks can be caused by a non-safety system, unrelated to reactor operation but installed in an industrial area of the plant. In fact, that non-safety system rendered reactor protection system train 'A' unavailable and could have caused the loss of electrical power to other train 'A' equipment. Lastly, the event involving the degradation of neutron-absorbing material highlighted that unidentified and unmitigated degradation may challenge the subcritical margin for the spent fuel pool.

2.12.2. Lessons learned

The effective use of operating experience can be a valuable tool; using lessons from past mistakes can help prevent recurrence. The repetition of events demonstrates weakness not just in the affected programme but also in other organizational factors that can lead to long delays between receiving operating experience and implementing the identified recommendations. The main lessons learned were the need for timely review, the assessment of external operating experience, and the follow-through of identified recommendations and commitments. An organization dedicated to safety takes advantage of any information available to prevent the occurrence or reoccurrence of potentially significant issues that can affect safety.

3. SPECIAL ISSUES AND OPERATING EXPERIENCE STUDIES AND WORKSHOPS

3.1. SPECIAL ISSUES

3.1.1. Experience with reactor pressure vessels

Particular issues concerning flaw indications were found in a pressurized water RPV while it was shut down for refuelling. The reactor vessel internals were removed to perform the third 10 yearly in-service inspections of the RPV circumferential welds. Following operational feedback from similar nuclear power plants, the in-service inspection included the ultrasonic test inspection of the RPV for potential underclad cracks, which were limited to the beltline region of the RPV shell, up to a 30 mm depth from the cladding interface. This was the first time since the RPV was manufactured that an extensive inspection of the lower and upper core shell forgings had been carried out.

No underclad defects were detected, but numerous nearly laminar indications were identified. A decision was taken to perform further ultrasonic test examinations that were aimed at detecting nearly laminar indications in the whole thickness of the two core shell forgings, the transition ring forging, the nozzle shell forging and the RPV flange forging. Approximately 8000 flaws were detected. The flaws were concentrated in the two core shell forgings and in the lower core shell, located within a region 120 mm from the inner surface and remote from the circumferential welds. The detected flaw indications were nearly circular in shape, had a typical size of 10 mm and were essentially laminar in orientation.

A decision was taken by the licensee to perform the same inspection at another plant with an RPV of identical design and construction. Once again the inspection did not detect any underclad cracks but identified many laminar flaw indications, though less numerous (approximately 2000) than those found at the first plant. Both plants were in cold shutdown with the core unloaded.

Both these RPVs were constructed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The hemispherical cap of the bottom head and the elliptically shaped dome of the RPV head were made from low alloy steel plate, and the other parts of the RPV were low alloy steel forgings.

Vacuum degassing, which minimizes hydrogen in the environment during the manufacturing of forgings, has been a standard practice applied by all heavy section steel forging manufacturers. Dehydrogenation heat treatments and cooling practices to minimize hydrogen are also commonly applied by manufacturers of

nuclear RPVs. A forging is considered to be unacceptable if the ultrasonic test examination detects the presence of reflectors that produce indications resulting from discontinuities in the material, accompanied by a complete loss of back reflection from the far side of the structure.

3.1.1.1. Safety significance

The safety functions of the RPV provide a support structure for the internal reactor components, the retention of reactor coolant as part of the reactor coolant flow path, and a containment barrier for the release of fission products as part of the reactor coolant pressure boundary.

The flaws detected in the core shells of RPVs could challenge the safety performance and serviceability of the RPVs. The flaws were attributed to hydrogen flaking, which originated in the steel during the forging process.

The safety concern related to the identified flaws concerns not only the quantity of flaws but also their presence at all, as they occurred during the forging process.

The defect acceptance criteria for forgings specified in the 1974 edition of section III of the ASME Boiler and Pressure Vessel Code are found not to be stringent enough to cause a flaked component to be rejected. The evaluation of these defects may require extensive analysis and tests.

The acceptance standards for the ultrasonic examination of the forgings of Class 1 equipment should be reviewed and evaluated to verify that those standards adequately address the issue of hydrogen induced flaws.

The main potential consequences for the safe performance of the RPVs from the flaws attributed to hydrogen flaking are:

- (a) The initiation of RPV brittle failure from the hydrogen flakes under the postulated loading conditions and, especially, under the pressure and thermal transients;
- (b) The degradation of the expected mechanical behaviour of material that is required to be flaw tolerant with adequate plastic reserve strength to allow the structure to accommodate flaws.

3.1.1.2. Lessons learned

This event identified that safety concerns that may require a review of fabrication records can appear at any time after construction. Actions should be taken to ensure the availability of fabrication records, as well as the legibility and readability of the information they contain. Licensees are required to maintain all fabrication records, including non-destructive evaluation testing and acceptance

records for RPV forgings during fabrication. If recordable quasi-laminar indications were detected during the fabrication of any RPV, then the indications would need to be compared to the examination acceptance criteria. Unacceptable indications would require repair in accordance with the applicable codes.

In addition, degradations may be detected for which standardized evaluation procedures do not exist. Detection during the in-service inspection of flaw indications in the base metal of RPVs, even if they are assessed as fabrication defects, could bring into question the adequacy of an existing in-service ultrasonic test inspection programme that is more limited to the welds.

It was also found that the defect acceptance criteria used for forgings were not stringent enough to assess the acceptability of the hydrogen induced flaws detected in the core shells of the RPVs. As the defect acceptance criteria for forgings set forth in section III of the ASME Boiler and Pressure Vessel Code have not changed significantly since 1974, their adequacy to address the issue of hydrogen flaking in forged components should be re-evaluated.

There is no evidence of any factors unique to the forging practices used by the manufacturer of these RPVs that suggests an increase in the likelihood of developing quasi-laminar indications during the fabrication process in comparison with other forging manufacturers.

3.1.2. Experience with fuel damage or baffle jetting

A review of events reported during the 2012–2014 period identified recent occurrences of a previously well known fuel failure mechanism that was first identified in 1971. This phenomenon is called baffle jetting, and this fuel failure mechanism can occur in pressurized water reactors that use core bypass flow to cool the baffle area during operation. In reactors with a downward flow scheme, a portion of the coolant from the reactor inlet flows through a hole in the upper portion of the core barrel into the space between the outside of the baffle plates and the core barrel (this flow is part of what is known as bypass flow). Reactor coolant that is not bypass flow goes down through the annulus and then changes direction to go up through the core support and lower plate and then through the core. The different flow paths and the associated pressure drops between the two can create a differential pressure between these fluids. If enough of an area or gap exists between adjacent baffle plates, and the pressure difference between the core flow and bypass flow is high enough, a high pressure water jet can form between gaps in the baffle plates. This phenomenon is called baffle jetting, and the water jet that is created can impinge on nearby fuel assemblies and result in cladding perforation and a potential for the loss of fuel pellets and other fuel rod components into the reactor coolant, spent fuel pool and other associated systems (see Fig. 9).



FIG. 9. Fuel damage / baffle jetting.

Three units reported fuel damage related to baffle jetting during the 2012–2014 period.

The operator of a two unit pressurized water reactor site noticed a slight increase in the radioactivity level in the Unit 2 reactor coolant approximately nine months into the operating cycle ($<0.1\%$ technical specifications limit for dose equivalent ^{131}I). On the basis of these indications, the plant operator expected to find minor fuel cladding damage during the subsequent outage. During the outage fuel assembly inspection, the plant operator identified two visibly split fuel pins. The plant operator then attempted to identify and recover the fuel pellets from the damaged fuel pins. Material from approximately eight fuel pellets was recovered, but about seven fuel pellets were broken up into smaller pieces and were unable to be retrieved or located during the outage. These remaining fuel pellet pieces were likely eroded into fine particles that will remain contained in the reactor coolant system and likely have been or will be removed by the normal purification processes via reactor coolant system purification filters and ion exchangers.

Two similar events related to baffle jetting were also reported during this period. During a refuelling outage, one plant operator identified six fuel assemblies that were found to be leaking. During the transfer of fuel assemblies from the reactor to the spent fuel pool, a 50 cm long unknown object was found at the bottom of the fuel transfer channel. It was later determined that this object was a segment of a fuel rod. Visual inspection showed open cladding defects in eight fuel rods from three fuel assemblies. A second plant operator identified evidence of fuel damage from increased ^{133}Xe activity in the primary coolant.

Fuel sipping conducted during the subsequent refuelling outage identified two leaking fuel elements. These elements were situated at the outer core rim, in positions that were adjacent to the baffle plates. Each element had one side facing the baffle plates. One of the failed fuel elements contained at least two leaking rods, which were located next to each other and along the baffle plate. A visual inspection of this rod was performed by the fuel supplier. Both of the end caps, the plenum spring and several fuel pellets were missing from the rod. One end cap, the plenum spring and a part of the missing pellets were recovered during the search. The plant operator was already aware that the most damaged rod was located in a position where the core baffle contained an increased opening (maximum of 0.9 mm) between the baffle plates.

3.1.2.1. Safety significance

All three reported events posed no threat to the public. Low level, insignificant increases in reactor coolant activity level (<0.1% and <0.3% of the allowable technical specification limit, where reported) occurred as a result of the fuel damage. The actual damage to the core in all events was insignificant in terms of the safety analysis limits. Although one traditional defence in depth fission product barrier (the cladding) was breached in all cases, the two remaining fission product barriers (primary coolant piping and containment) were not affected in either case. A safety analysis was conducted by the nuclear steam supply system and fuel supplier to two of the three units that reported baffle jetting induced failures. The supplier evaluated the potential impacts of baffle jetting on departure from the nucleate boiling ratio, peak clad temperature, water logging (water–fuel pellet interaction), control rod operation and coolant activity. The supplier concluded that baffle jetting, even if it resulted in the worst possible fuel rod damage, did not constitute a safety concern.

Several minor concerns were reported by plant operators. One concern related to the impact of the irretrievable fuel pellet material on the reactor coolant system. Fuel assemblies include debris filters, and reactor coolant system cleanup filters are proven to minimize the potential for debris from fretting failures. The potential exists for operational challenges associated with localized areas of increased radioactivity and surface contamination in and around the primary coolant system and, therefore, an increased radiation dose to plant personnel.

3.1.2.2. Lessons learned

The plant operator at the reporting dual unit site had experienced Unit 1 baffle jetting events during the mid-1980s. To correct this problem, the plant operator ultimately performed a plant modification (called an upflow conversion)

in 1996. However, the plant operator did not perform a similar modification on Unit 2 because it already had what was considered to be an improved baffle design (it had full length edge bolting on all the centre injection baffle plate joints to minimize the gap between the plates). The plant had no previous history of failures or indications of baffle jetting developing on Unit 2, according to earlier fuel inspections. The same can be said for another plant that reported fuel failure because of baffle jetting.

Although baffle jetting is still a rare occurrence, two of these plants were once considered to have a low risk for baffle jetting because of their improved baffle bolting configuration. That both have recently experienced fuel failure because of baffle jetting may indicate that ageing mechanisms can result in gap widening at the baffle joint. One plant operator identified that the material properties of the baffle plates and bolting due to ageing mechanisms (stress, temperature and irradiation) since initial plant startup have resulted in relaxation and creep (void swelling). The relaxation and creep may have resulted in the gap widening at the baffle joint and set up conditions for baffle jetting to occur. It is important to consider the impact of plant and material ageing on systems that may currently be considered unlikely to experience generic failures.

3.1.3. Experience with the qualification and commercial grade dedication of safety related components

An information notice was issued by the Nuclear Regulatory Commission to provide information identified during vendor inspections concerning aspects of vendors' qualification and commercial grade dedication programmes. The information notice provided examples of situations in which vendors had not implemented sufficient controls to verify that safety related equipment supplied for use in nuclear power plants was qualified to meet its design requirements. The information notice also identified issues with the implementation of processes used by vendors to qualify components to perform their safety functions. The information notice highlighted examples in which previous qualification testing and analysis had not been established and had been improperly applied to the currently supplied component. This is of particular concern for commercial grade items, as changes made by commercial equipment manufacturers could affect the component's qualification yet go undetected. The Nuclear Regulatory Commission referenced specific guidance for the implementation of acceptable processes in the qualification of components to perform their safety function.

3.2. THE EUROPEAN CLEARINGHOUSE

Since 2008, a centralized regional clearinghouse, operated by the European Commission's Joint Research Centre in Petten, the Netherlands, has promoted collaboration on European nuclear power plant operating experience feedback. The European Clearinghouse performs in-depth analyses about certain topics to identify the main recurring causes, contributing factors and lessons that can be learned from reported events. These topical studies are primarily based on the IRS database, as well as on operating experience databases from France, Germany and the United States of America.

Since the last edition of this publication, studies have been completed on the following topics:

- Decommissioning;
- Digital instrumentation and control;
- Emergency diesel generators;
- Cracks and leaks in the reactor coolant pressure boundary;
- Organization and management.

The completed studies are available on the IRS web site. In addition to the topical studies, the European Clearinghouse also issues quarterly reports on safety significant nuclear power plant events and organizes workshops and technical meetings on operating experience.

3.3. INTERNATIONAL NEA OPERATING EXPERIENCE WORKSHOP

The OECD/NEA Committee on Nuclear Regulatory Activities Working Group on Operating Experience sponsored an international workshop on operating experience programme effectiveness measures from 8 to 10 September 2014. The following three topics were explored by breakout groups responsible for developing measures and tools: (a) identifying and gathering operating experience, (b) processing operating experience information and (c) assessing outcomes of operating experience programmes. During the workshop, conclusions and practices were identified to help improve operating experience feedback and assessment methods.

3.4. NATIONAL OPERATING EXPERIENCE FEEDBACK PROGRAMMES

The importance of operating experience feedback in enhancing nuclear safety is well recognized. A general goal of national and international operating experience feedback processes is to help prevent the recurrence of events involving serious potential hazards. National operating experience feedback is the basis for international operating experience feedback, so without high quality national operating experience feedback it is not possible to have effective international operating experience feedback.

The OECD has published the report *Update on the Use of International Operating Experience Feedback for Improving Nuclear Safety*, which looks at the current state of international operating experience systems, the role of regulators in operating experience systems and the positive and negative aspects of existing operating experience systems.

4. CONCLUSIONS

By signing the Convention on Nuclear Safety, each Contracting Party has committed, under Article 19, to taking the appropriate steps to ensure that “incidents significant to safety are reported in a timely manner by the holder of the relevant licence to the regulatory body” and that:

“programmes to collect and analyse operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies”.

All Contracting Parties have indicated in the review meetings that they have such programmes in place, and these programmes have indeed been valuable in sharing experiences.

In developing the operating experience feedback programme and the network for its implementation, it is important to keep in mind the main idea: writing reports and collecting data is meaningful only when there is a direct link to the reduction of risk and the enhancement of operational safety. Therefore, event reporting needs to be coupled with programmes that transform the lessons learned into risk reducing measures, such as improvements in design, operator training, operating procedures and safety culture. To achieve this goal, the

operating experience feedback process would need to provide greater focus on corrective actions, good practices, their implementation, and information sharing among all parties that contribute to the safe operation of nuclear facilities.

Operators and regulators are the providers of information. No operating experience feedback network can function effectively without the active participation of the nuclear operators and national organizations that work in the cooperative spirit demonstrated by signing the Convention on Nuclear Safety. The national organizations that need to be involved in the operating experience process are the operators of the facilities and the regulatory bodies. It is expected that the main responsibility of collecting and using relevant national operating experience and reporting it to the international network will be clearly assigned to appropriate organizations in each country. The other duty of the national organizations is to assess the operating experience information received from the international network and initiate proper measures to reduce risks at nuclear facilities.

Activities within the IRS extend beyond the exchange and feedback of event information. Both the NEA and the IAEA have assigned working groups of experts who meet frequently and discuss the safety relevance of such events. These groups take part in in-depth discussions of important events. The conclusions and recommendations from the in-depth discussions are extremely valuable to the nuclear community in helping to enhance the safety of plants in the design, development and operation stages. To strengthen the Member States' operating experience programmes:

- (a) All countries operating nuclear facilities need to act as part of an international network for mutual learning, as indicated in the Convention on Nuclear Safety, and freely share information of importance to operational nuclear safety as well as allocate the needed resources to make the international operating experience feedback effective and efficient.
- (b) The operating experience feedback process needs to contain all the components that close the feedback loop: collection, review and quality control, analysis, conclusions, dissemination, follow-up and feedback.
- (c) A knowledge management component needs to be built into the system with efficient search functions, like semantic searches, to ensure the maintenance of relevant nuclear safety information and its transfer to future generations of experts.

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The first edition of this publication covered safety related events reported between 1996 and 1999. This sixth edition covers the 2012–2014 period and highlights important lessons learned from a review of the 258 event reports received from participating States during those years.

The IRS is jointly operated and managed by the OECD Nuclear Energy Agency (OECD/NEA) and the IAEA.