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# WORKING MATERIAL

## HIGHLIGHTS FROM THE INTERNATIONAL REPORTING SYSTEM FOR OPERATING EXPERIENCE (IRS) FOR EVENTS IN 2010 – 2011

**REPORT OF A CONSULTANTS MEETING
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN VIENNA, AUSTRIA,
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## CONTENTS

# 1. INTRODUCTION

The International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) of the Organization for Economic Co-operation and Development (OECD) jointly operate the International Reporting System for Operating Experience (IRS) by collecting and distributing detailed information of member countries' NPP events considered important to safety, lessons learned and accident prevention.

The sharing of international experience from unusual events at the NPPs is the main impetus behind the IRS. Collection, analysis and distribution of information on unusual events form the basis for continuous learning to improve nuclear safety. IRS Reports are collected according to internationally accepted reporting criteria so the IRS reporting process is inevitably selective. The number of events reported to the national systems in member countries is far in excess of the number of reports reported to the IRS since the national systems have, as a rule, a lower threshold for event reporting. Since the IRS contains only selected incidents from member countries, it should not be used as a source for statistical purpose or for component reliability studies.

What is reported to the IRS is decided by the national regulatory body, since the IRS is the information system of regulators. Each member country is represented in the IRS by an IRS national co-ordinator who is a staff member of the respective national regulatory body.

The IRS was primarily established as a tool for the exchange of information among regulators who do not have access to information systems of the nuclear industry; it has a restricted character. Nevertheless, it is also intended for the use of nuclear power plants, utilities, technical support organizations, designers, vendors, research organizations and technical universities working in the nuclear field.

In the period 2010 – 2011, the member states submitted 166 IRS reports to the IAEA.

The IAEA convened a Consultants Meeting to review the IRS reports received from the Members States in the period of 2010 – 2011. The meeting was held in Vienna, 27 February – 2 March 2012.

The participants reviewed the IRS events reported during 2010 – 2011, identified and selected events with significant lessons and interesting information in the equipment area and management of safety area.

The reports selected for the highlights were sub-divided into groups:

- Events with Important Lessons
- Equipment and Systems Issues
- Management of Safety

The five selected "Events with Important Lessons" (chapter 2), were those viewed as especially significant because they involved multiple human errors, severe consequences, potential for common cause failure or long standing safety management problems. [redacted]

Thirty-six events were categorised as Equipment and System Issues (chapter 3), with a subsequent categorization of:

- Flooding
- Cracks and Corrosion
- Leakage

- Diesel Generators
- Control Rods and CRDM
- External Hazards
- Degraded Cooling Conditions
- Electrical Equipment Events
- Containment

Thirty-eight events fell under the category of Management of Safety (chapter 4), and were further grouped into the following categories:

- Human Performance
- Procedure Issues
- Control of Modifications
- Foreign Material Exclusion
- Design and Installation
- Criticality (reactivity) Events
- Ineffective Use of OE, Recurring Events
- Radiological Protection, Exposure, Contamination
- Procurement and Spares

It should be mentioned that many events involved several different aspects and could have been assigned to more than one category. For example, an equipment failure is very often combined with management deficiency or human error. Nevertheless the participants decided to assign each event to only the category viewed as most representative of the issue, to avoid repetition.

The complete list of IRS reports from 2010 – 2011 which were the subject of this review is provided in Annex 1.

The list of contributors to this IRS Highlights is provided in Annex 2.

The ten Highlights Reports covering the period from 1994 to 2009 are available on the WB IRS.

# 2. EVENTS WITH IMPORTANT LESSONS

For the period of 2010 – 2011 five events were selected to be highlighted in this chapter. All these events could have been included in either the Equipment section or the Management of Safety section as these events involved failures on multiple levels. All of these events involved several of the following factors:

- design and equipment deficiencies
- weakness in the qualification and training of personnel
- human errors
- deficiencies in the preparation and implementation of modifications
- deficiencies in procedures
- non-compliances with procedures
- inadequate application of operational experience feedback
- deficiencies in surveillance testing and in-service inspection
- insufficient analysis of technical problems
- deficiencies in the use of human error prevention tools (pre-job brief, self-check, peer check, STAR, etc.)
- ineffective or inadequate maintenance

In several cases described here, relatively minor failures became more significant due to several other contributing causal factors involving both human performance and equipment failures. Early capture and correction of the causes of minor events or near misses is important to avoid more significant events or repeat events.

In several reports known problems and sub-standard conditions were not considered important, and so were not addressed until a failure had occurred.

The failure to ensure that maintenance programs, system tests and procedures are robust and correctly implemented has led to several of the events mentioned in this section.

[redacted]

This section provides a brief summary of the event reports with significant learning points.

IRS 8079 – RECURRENT UNAVAILABILITY OF SG TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP

The report described repeated events involving the turbine driven auxiliary feed water pumps resulting in unavailability of the auxiliary feed water system (AFWS). The first two events were attributed to a high accumulation of water (steam condensate) in the turbine of the pump. When the water was expelled through the exhaust, the speed regulation system could not compensate for the prompt acceleration of the turbine, and the over speed protection threshold was reached.

The root causes of the first problems (three pump trips due to over speed protection) were not clearly identified by the operator. As a result, further periods of unavailability occurred when the pump failed to start due to the non-opening of the steam valve inlet to the turbine. Again, the root cause analysis focused primarily on the direct cause of the AFWS unavailability.

Many units of the operator were affected by similar AFWS unavailability, although less frequently. As a result, the operator launched a multidisciplinary working group to examine all AFWS unavailability encountered at the NPPs in order to develop a global action plan. The analysis was extended to all main components of the AFWS (i.e. motor-driven pumps). The proposed areas identified for improvement were:

- Implementation of procedures and modifications to avoid the accumulation of condensate water in the turbine. This included adding insulation, modifying the automatic purge system, increasing the turbine steam conditioning flow rate, and adding surveillance requirements for local and external temperatures, all areas that were suspected to have a strong impact on condensation accumulation.
- Reassessment of the maintenance plan of all equipment (mechanical, electrical, I&C) essential to ensuring the availability and reliability of the AFWS, in order to detect possible weaknesses and gaps with qualification requirements and vendor recommendations.
- Reassessment of the periodic test program of AFW-components in order to identify early degradations and to anticipate failures.

Lessons learned from the incidents include that addressing only the apparent causes may not be the solution in recurrent events and that a deep analysis of technical problems affecting complex systems or equipment like the turbine-driven AFW pump requires a multidisciplinary approach. Further, the actions defined in the framework of experience feedback process from events at one unit should be implemented at other units systems with the same design to avoid recurrent events.

## IRS 8088 – RADIOACTIVE RESIN FOUND IN VENTILATION SYSTEM

A combination of poor design, ineffective oversight, equipment deficiencies, and poor procedures resulted in the undetected flushing of radioactive resin into the plant ventilation system on multiple occasions.

Workers performing unrelated maintenance discovered fluid leaking from a temporary ventilation system in a radiologically controlled area. Analysis of the liquid found it to be slightly radioactive. The fluid was traced back to a flush of a resin tank that had been performed in the newly constructed solid waste treatment plant. The intent of the procedure had been to flush the resin tank and its overflow lines with clean water into a larger waste holding tank. As the level indication of the holding tank was known to have been inaccurate in the past, a video camera had been installed to assist in determining level. This camera was also not functioning properly. Calculations based on radiation levels around the tank gave an estimate that the tank was about 50 % full, when in fact it was actually 75 % full. In addition, the tank overflow line had been isolated, since the waste pond accessed by the overflow line was difficult to clean. When the resin was flushed into the nearly full tank, the tank overflowed into the tank’s gas removal pipe, and from there into the solidification plant ventilation system. The ventilation system of the solidification plant is shared with the auxiliary building, and feeds from there to the main ventilation stack.

During cleanup activities, dry resin was found in portions of the ventilation system that had not been affected by this event, and it was determined that this was not the first time radioactive resin had been flushed into the ventilation system. A total activity of 2 GBq may have been present in the ventilation system, though due to the low aerosol nature of the resin, it does not appear likely that much activity was released outside the plant. Neither the continuous monitoring system nor analysis of air filters showed any indication of increased activity. Assessment of the grounds outside the plant buildings found no radioactive particles. Traces of Co-60 were found in sand used in the parking lot during winter conditions, though analysis of the origin of the Co-60 is still pending.

Review of the incident determined that the root cause was a weakness in the design of the solidification plant that allowed direct access from the waste tank to the ventilation system. During construction of the solidification plant there was a lack of coordination between the designer and the end user, and different groups associated with the development of the solidification plant did not communicate with each other. The problem with the level indication had not been appropriately addressed despite being a recurring issue, and procedures associated with waste transfer did not mention the impact that a resin/water mixture could have on tank level measurements. The tank overflow line had been left isolated, preventing it from fulfilling its passive safety function, and there was no procedure directing operator actions in the event that tank overflow should occur. Corrective actions addressed process modifications for radioactive waste operations and re-evaluation of the operability of level measurement systems.

## IRS 8127 – UNDUE ACTUATION OF ONE EMERGENCY DIESEL GENERATOR DURING OUTAGE DUE TO ARCING SHORT

When the unit was in cold shutdown during a refuelling outage, maintenance personnel were performing dust removal activities in a reserve power supply cubicle of a 6 kV non-vital unit switchboard. When one of the two workers standing on a double ladder placed the cleaning spray can into the cubicle, he felt muscle jerking in his right forearm and his colleague felt the same in his lower leg, which led them to the conclusion of having received an electrical shock. Therefore, both of them came off the ladder, suspended work and notified the lead worker.

The lead worker wanted to test the voltage of the cubicle by using a 0.4 kV manual voltage tester and touched two phases of 6 kV busses with 0.4 kV tester contacts. An electric arc and loud noise followed. The lead worker conducting voltage testing suffered burns on his face and hands, and he was taken to a hospital for treatment. The two workers in the proximity were not injured.

Induced by the fault, differential protection of the 6 kV reserve bus actuated and the 6 kV normal breaker was de-energized. Emergency diesel generator started and the load-sequencing program actuated. The event involved no fire.

The injury and the de-energization of the 6 kV switchboard were caused by repeated violations of procedures by the personnel involved. This included: failure to perform the pre-job briefing required when assigning a work task, failure to perform the checks and briefing required during transfer of the work area, failure of the installation supervisor to perform independent verification, inadequate equipment labels and indications, and failure to apply the standards for work conduct. Proper adherence to any of these work practices could have prevented the event. Repeated violation of rules occurred despite the fact that both the personnel transferring the work area and the installation supervisors had completed regular checks and pre-outage training on the importance of maintaining updated information.

As a corrective action for the event, the personnel involved will undergo special classroom and on-the-job training.

## IRS 8144 – SUBMERGED ELECTRICAL CABLES

An information notice was issued by NRC (IN 2010-26) to inform licensees of recent operating experience regarding the significant number of events related to issues of protracted cable submergence in water. Inaccessible or underground power cable failures could disable accident mitigation systems or cause plant transients. This information notice is the second IN related to submerged electrical cables (see IRS 7527 “IN 2002-12, Submerged Safety-Related Electrical Cables”).

Cable failures have a variety of causes, including manufacturing defects, damage caused by shipping and installation, and exposure to electrical transients or abnormal environmental conditions during operation. Latent shield or insulation damage could result from errors during cable installation, which could be caused by cable jamming, cable sidewall bearing pressure, pulling cables through conduits and flexible conduit, or computerized cable routing system software routing cables through the wrong raceway. The likelihood of failure from any of these factors increases over time as the cable insulation degrades and/or is exposed to water.

A detailed review was performed by NRC inspectors of underground electrical power cables after moisture-induced cable failures were identified at some plants. The cables were exposed to submergence in water, condensation, wetting, and other environmental stresses. Because these cables are not designed or qualified for submerged or moist environments, the possibility that more than one cable could fail has increased; these failures could disable safety-related accident mitigation systems. As a result of these inspections, NRC inspectors identified numerous inspection findings that indicated that many licensees are not maintaining cables important to safety in an environment for which they were designed.

Eleven events were highlighted in this report identifying issues that were found during the above inspections. All of these events concerned safety related cables related to systems and equipment such as residual heat removal, containment spray, essential service water, safety related motorised controls, emergency diesels etc.

Six event reports were specifically attributed to ineffective or inadequate maintenance and testing programs and condition monitoring of submerged cables which resulted in the inability of the licensees to adequately demonstrate that cables subject to submergence in water would perform satisfactorily in service. Two of these reports concerned events with actual consequences. In the first report a transformer failed resulting in the loss of off-site power and in the second another transformer failed which caused the loss of a safety bus. In both these examples the maintenance and testing programs had not given any indication of imminent failure of the cables.

In the remaining five events, whilst no consequence was realised, the NRC inspectors identified numerous examples where the licensee had failed to demonstrate that the safety related cables were designed or qualified for continuous submergence or that they would remain operable.

Cables are not typically designed or qualified for submergence unless they are procured as submarine cables.

Demonstration that a cable is designed or qualified for long-term submergence (i.e., submerged in water continuously or for extended periods of time) requires a qualification test report or certification from the cable vendor. The industry’s previously conducted post-loss-of-coolant accident cable submergence tests do not demonstrate qualification for long-term cable submergence, and the use of the Arrhenius methodology by some licensees to demonstrate qualification for long-term cable submergence is invalid. For areas in which cables could be submerged, the licensee should identify and demonstrate that these cables are designed or qualified by documented testing for the required duration.

In many of the events above the licensee did not implement timely corrective actions dealing with longstanding issues of submerged cables or did not apply effective operational experience from events concerned with submerged cables. In many cases the licensee lives with known degraded conditions where the ingress of water is not prevented, subjecting cables in these areas to adverse conditions.

Cables not designed or qualified for, but exposed to, wet or submerged environments have the potential to degrade. Cable degradation increases the probability that more than one cable will fail on demand because of a cable fault, lightning surge, or a switching transient. Although a single failure is within the plant design basis, multiple failures of this kind would be challenging for plant operators. Also, an increased potential exists for a common-mode failure of accident mitigating system cables if they are subjected to the same environment and degradation mechanism for which they are not designed or qualified for. Some licensees have attempted to periodically drain the accumulated water from the cable surroundings to avoid cable failures. In some cases, the water quickly refilled the cavity in areas in which the water table was above the base level of a cable trench or underground vault. In other cases, water accumulated seasonally (e.g., because of snowfall or rain), filling the conduit or raceways. In both cases, periodic draining could slow the rate of insulation degradation, but it may not prevent cable degradation. Licensees should ensure that cables that could become submerged are adequately monitored.

IRS 8149 – UNAVAILABILITY OF RPS AUTOMATIC CONTAINMENT ISOLATION FUNCTION

The reactor protection system (RPS) containment isolation function for internal pipe breaks (I-function) is intended to monitor the containment and, in case of leaks from systems belonging to the reactor coolant pressure boundary, to initiate signals for the actuation of safety systems and functions.

The main functions include: activation of an automatic reactor trip, closure of containment isolation valves, initiation of the containment spray for cooling of and pressure reduction, starting the emergency diesel generators, and alignment of the emergency core cooling system.

At the beginning of a refuelling outage, interlocking of the reactor protection system containment isolation function for internal pipe breaks was planned. However it was discovered that the function was already interlocked.

Investigation concluded that the RPS I-function had been interlocked since the previous extended refuelling and modernization outage. This meant that automatic activation of the RPS I-function had been disabled during more than two months of power operation. RPS I-function activation signals to the main control room, and unit log and alarm computer remained enabled during this period.

Subsequent to the event, the licensee has strengthened its routines associated with operational readiness control of interlocks in instrumentation and control (I&C) cabinets. The licensee has also clarified the responsibility between the operations and I&C departments in routines of I&C cabinets, and modified operating procedures accordingly. A new steering document was created to describe what barriers have to be established, why they are established and when to establish them during the course of tasks associated with the RPS.

The licensee has made hardware modifications to the RPS I&C cabinets with the introduction of plug-in units, replacing the use of jumpers. These units have lamp annunciation on the cabinet doors. The licensee intends to install alarm annunciations in the main control room for interlocked automatic RPS isolation functions.

The event is potentially relevant for other nuclear plants worldwide. This event may be of particular interest to nuclear units in extended modernisation outages, with heavy workloads across the organisation and in the organisation of contractors.

The full adherence to written procedures, reporting procedures, and proper documentation is of prime importance for ensuring the successful performance of any task. Adherence to a strong safety culture and good individual work practices at all levels of the organisation are critical to ensuring safety.

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# 3. EQUIPMENT ISSUES

The events discussed in this chapter were mainly caused by failure or degradation of different type of equipment. The direct cause of these events is usually an equipment failure due to design deficiency, material and manufacturing deficiencies or some type of physical or chemical phenomena.

It should be noted that the plant management or personnel has influence on the equipment behaviour and these equipment failures can often be attributed, in part, to deficiencies in the maintenance and management system. Many of the events involved some management issues but as the major learning points are related to the behaviour of the equipment, they are discussed in this chapter. It should also be highlighted that some of the event reports are generic, integrating a significant number of lessons learned, and demonstrating the importance for safety improvement of taking advantage of the global international experience feedback.

Aging of the plant SSCs is a particular issue, especially in areas that may not be readily accessible, such as buried piping and cabling and portions of containment. Recent events highlight the need to review procedures and programmes to consider inaccessible areas and penetrations where poor environmental conditions can impact the integrity of the piping, equipment, and structures.

Once an unexpected or abnormal plant condition has been identified, such as the presence of corrosion, leakage, or a new source of path for flooding, prompt corrective action must be taken to prevent a further degradation or recurrence.

Although inadequate acceptance tests and manufacturing faults were attributed in the included reports to specific diesel generator designs, the analysis should be extended to other designs or equipment to prevent similar failures. Acceptance tests should be a significant part of the procurement process and identify, record, and report areas of concern to prevent occurrence of similar failures.

Some events related to external hazards could be prevented by improved communication and coordination with local organisations and authorities concerned with the operation and safety of the plant.

During the present biennual period a series of events related to the blockage of water inlet facilities were identified. Typically this blockage was caused by a combination of bad weather and poor design. Opening of upstream dams were also a potential source of debris. As a result of this event the plant has been working in conjunction with the local government to establish a safety separation area along the outgoing lines and to perform regular pruning of vegetation in the area.

During this review period, events associated with electrical equipment were reported in sufficient numbers to be highlighted as a separate reporting area. Compared to the previous period the number of reports with important lessons learned related to control rods and control rod drive mechanisms (CRDM) increased significantly. As such, this area merits increased attention. However, significant events related to fire and LOOP were not reported in 2010 – 2011. Plants should continue to put emphasis on the prevention of these types of events. [redacted]

## 3.1. Flooding

Safety related equipment must be protected from both internal and external flooding hazards. From 2004 – 2007, several events were reported each year. In the most recent biennial period of 2008 – 2009, a few design-related flooding events occurred but none of significance had been reported. In this period of 2010 – 2011 the most significant flooding event was, of course, the tsunami of 11 March

2011, [redacted] Additionally, three other events were selected to highlight the continued need for diligence in this area. One lesson to be reinforced is the importance of fully assessing the proximity of potential flooding sources to safety related equipment and the impact flooding could have on other equipment. In particular, these events highlight the vulnerability of electrical equipment to flooding. The second important lesson to be highlighted is that once a new flood path or source has been identified, prompt corrective action must be taken to prevent a recurrence and treated with a consideration for increased severity. These reports, though focused on flooding, are similar to the types of events highlighted in the section discussing the ineffective use of operating experience / recurring events.

IRS 8090 – INTERNAL FLOODING OF SAFETY RELATED 6 KV ELECTRICAL BUSBAR

During normal operation, water from the fire extinguishing system in the proximity of cabling overflowed into the safety related 6 kV electrical busbar one floor below. The flooding was caused by valves in the fire extinguishing system that were left partially opened following maintenance.

On discovery, personnel disconnected the safety related 6 kV electrical busbar from the service transformer and the associated EDG causing the inoperability of a safety related system. The event revealed that there existed a potential risk of internal flooding throughout the premises in the vicinity of fire extinguishing systems; that there were weak controls on infrequently performed preventive maintenance in areas that could affect safety systems at lower elevations; and that the design of the drain header was insufficient to handle potential flooding.

Included in the suggested corrective actions were to: perform an analysis of drain headers capacity, especially in the vicinity of electrical cabling; review maintenance procedures and related training for piping manipulations near cabling; install roofs above electrical busbars; and perform a probabilistic safety analysis (PSA) of internal flooding and fire for safety related electrical busbars.

IRS 8181 – REACTOR TRIPS RESULTING FROM WATER INTRUSION INTO ELECTRICAL EQUIPMENT

An information notice was issued by the NRC (IN 2011-12) to inform the industry of three recent events involving water intrusion into electrical equipment that resulted in reactor trips. In each event, an electrical fault occurred as a result of water intrusion into electrical equipment. Additionally, in each case, the licensee had previously recognized the source of the water but had not corrected it. In one case, the licensee had previously recognized the degraded condition of the building roof but left it unresolved for approximately seven years before the flooding event resulted in a dual-unit trip. In another event, the flooding was identified but the potential maximum impact of the flooding was underestimated. As with the events in ineffective use of operating experience / recurring events, the timely corrective actions are necessary to prevent recurrence and potentially more significant events.

IRS 8190 – LOSS OF PRIMARY COOLANT INVENTORY DURING REFUELING OUTAGE

The plant was in mid-loop operation during a refuelling outage, when around 25 $m^3$ of primary coolant was drained from the Reactor Coolant System to the containment sumps through the Residual Heat Removal (RHR) System, due to a misalignment produced during a surveillance test. The cause of the event, provoked by a human error, was the unscheduled opening of a valve located at the suction of the containment sump.

The water poured to the containment overflowed the sumps and reached a flooding level of about 5 cm. The event resulted in a reduction in the safety margins of the Critical Safety Functions at shutdown: Primary Inventory Control and RHR. The plant was working with reduced inventory with only one RHR train operable and the resulting conditions could have caused the inoperability of the other RHR pump. Additionally, the event demonstrated that the licensee did not maintain adequate configuration control during the refueling outage. Some of the key contributors of the event included: 1) The operations shift did not adequately follow the failure procedure, performing some actions not included in the procedure and not taking others that were included in it; 2) There was no training at the simulator for similar situations, as the simulator did not have the capability to simulate situations with the plant in a refueling outage with the vessel open; 3) All organisations involved or affected by a procedure should take part in the pre-job briefing. In this case, only maintenance and instrumentation staff attended this briefing, but other affected groups were unaware of what would be happening.

Some of the generic corrective actions for this event include: analysis of the configuration control of SSCs that can initiate a loss inventory while shutdown; review and implementation of the document INPO SOER 2010-02 "Engaged, thinking organizations." Include in the Abnormal Operating Procedures the key instructions for the start-up of important equipment; and include training for the control room personnel on scenarios with simultaneous malfunctions that require choosing and prioritizing among different Abnormal and Emergency Operating Procedures.

### 3.2. Cracks and Corrosion

The number of events relating to defects in pipes and equipment which result in corrosion leading to through wall cracks continues to be high. The failure to maintain high energy piping and components within specified thickness values can adversely affect the operability, availability, reliability or function of systems required for safe shutdown and accident mitigation. There is also the possibility of a significant threat to the safety of workers in the case of a pipe rupture as has happened in the past. Aging of the plant makes this an issue, especially for seawater systems. However, if inspection programmes were performed more rigorously and more frequently, many issues could be identified and resolved before they become significant. Some of the events demonstrate the need to ensure inspections are performed in areas that may not be readily accessible such as trenches and penetrations and where poor environmental conditions can impact on the integrity of the piping and equipment. Cavitation has also been found to cause degradation of the lining of pipes leading to exposure of the base metal and then corrosion, which demonstrates the need to adequately assess the engineering design of such systems where there are severe flow conditions. The presence of chlorides and contaminants, and particularly their effect on stainless steel piping is another area of concern, as is the relatively new issue of materials containing fluorides which decay when subject to high temperatures and radiation, particularly in locations where there are stress concentrations due to residual welding stresses.

IRS 8098 – THINNING OF RESIDUAL HEAT REMOVAL SEAWATER SYSTEM PIPING

During repair work of the inner lining of the carbon steel piping in the residual heat removal seawater system-A, marks of corrosion were found on the outer surface of the inlet piping for the heat exchanger. A measurement found the wall thickness of the area concerned to be 6.7 mm. This was less than the required minimum wall thickness of 7.08 mm as per the Technical Standards. Other sections of the piping were examined and no further thinning of the piping was found.

The piping of the RHRS system-A is a part of the piping from the intake pump to the heat exchanger and is used in the penetration to the waste treatment building. The outer side of the building penetration is formed by an underground trench with a ceiling hatch. The building penetration has a sleeve structure: the wall is initially opened by coring, a pipe passes through the opening, and the gap between the opening and piping is filled with mortar.

Several issues contributed to this event. Rainwater had entered through a hatch in the trench and collected on the anchor support of the pipe before seeping between the mortar and the piping, causing corrosion to occur. Because of the difficulty of accessing these areas, visual inspections were not performed on these penetrations. Treatment to prevent corrosion was not adequate for areas with high humidity.

Corrective actions included the addition of rainwater guttering and the improvement of visual inspection programmes for penetrations.

### IRS 8102 – THINNING OF SEAWATER PIPING OF HIGH PRESSURE CORE INJECTION DIESEL COOLING SYSTEM

During a periodic inspection, measurement of the wall thickness was conducted for the seawater piping of the high pressure core injection system (HPCI) diesel cooling system. On inspection it was found that a minimum wall thickness of 2.8 mm existed in an area of the diesel cooling water cooler inlet pipe. This minimum thickness was below the minimum required wall thickness (3.4 mm) calculated in accordance with the relevant technical standard. Wall thickness less than the minimum required wall thickness was not found at any other location.

According to the long term maintenance management policy associated with an evaluation on plant aging, visual inspection was conducted for the positions where seawater drifts. During this inspection, swelling and detachment of lining (coating) on some parts of the inner surface were observed.

The investigation confirmed the adherence of marine creatures on the inner surface of the affected seawater pipe. Over time this contributed to the damage of the tar epoxy resin lining on the inner surface of the pipe, allowing seawater to contact the base metal of the pipe, causing corrosion.

The seawater pipe had been checked for leakage at every periodic inspection since the initial start of plant operation. In addition, visual inspections had been conducted for the interfacing (connecting) elbows of the diesel cooling water cooler as representative positions to check the corrosion condition of the whole seawater piping system. In addition to the interfacing elbows, there are other positions that are under severe corrosive environmental conditions and therefore the interfacing elbows are not a sufficiently representatives of the whole seawater piping system. Additionally, the specific content and frequency of the inspections were not clearly defined.

In response to this event, parts of the pipe and elbows were replaced with new pipes containing a polyethylene liner that is less exfoliative than the tar epoxy resin liner. Other areas were checked for thinning of the wall and inspection programmes were improved in relation to method and frequency.

### IRS 8145 – LEAKAGE OF SEAWATER FROM SEAWATER INLET PIPING OF REACTOR COMPONENT COOLING WATER HEAT EXCHANGER

A maintenance worker found seawater leakage from the seawater supply pipe of the reactor component cooling water system cooler A installed on the 1st basement floor of the reactor auxiliary building. The unit was in an outage for the 27th periodic inspection. The affected pipe was isolated for inspection.

A visual inspection of the outer surface of the affected pipe identified no anomaly such as cracking or corrosion. However, after removing the paint on the outer surface near the leak, a hole of 2-3 mm in diameter was discovered. Corrosion in the base metal of the pipe had developed from the inner surface to the outer surface and a partial loss of the rubber lining was discovered on the inner surface of the pipe. In addition, wear marks presumably caused by cavitation were also observed in the vicinity of the lost lining. Due to the lost lining the base metal was exposed to seawater which caused the corrosion.

Examination of the operating conditions of the pipe showed that the lining could wear out due to cavitation resulting from severe flow conditions induced by the flow control valve installed upstream of where the flaw was found. The affected pipe was not considered to subject to special damage modes such as cavitation / erosion, even though it was under severe flow condition. As a result, the inspection frequency was not appropriate for such severe conditions.

The pipe was replaced and the inspection method and frequency for small and medium calibre pipes, particularly those subject to cavitation whose inner surfaces cannot be inspected directly from the flange openings, is to be improved and increased.

## IRS 8158 – CONTAMINANTS AND STAGNANT CONDITIONS AFFECTING STRESS CORROSION CRACKING IN STAINLESS STEEL PIPING IN PRESSURIZED WATER REACTORS

An information notice was issued by the NRC (IN 2011-04) to inform the industry of recent operating experience from several NPP's about the effects of contaminants and stagnant conditions on the potential for stress corrosion cracking (SCC) in stainless steel piping. Four examples are presented in this information notice identifying issues where stress corrosion cracking was a degradation problem for aging PWR plants particularly in environments that contain chlorides or stagnant flow conditions.

Austenitic stainless steel piping is susceptible to TGSCC when tensile stresses are applied in a chloride environment where local temperatures exceed approximately 140 °F. IGSCC can occur in austenitic stainless steels exposed for a sufficient time to temperatures between about 800 and 1,500 °F and subsequently exposed to tensile stress and water containing sufficient levels of oxygen at elevated temperatures. SCC can be initiated from the outside and inside surfaces of the pipe and can occur at the location of stress concentration regions (such as at welds for pipe restraint lugs) or susceptible regions for corrosion such as at the interface between the pipe and support clamp.

The operating experience described shows that, as nuclear plants age, SCC can potentially become an emergent degradation mechanism in PWRs for environments that contain chlorides or stagnant flow conditions. Licensees should be aware of the potential for SCC to occur in stainless steel in PWR applications.

Research indicates that even very low levels of chloride can have a detrimental influence on crack growth rates. Nuclear power plants located close to oceans are susceptible to chloride-induced ODSCC because of the salty air. However inland plants are also susceptible to chloride-induced degradation from other chloride sources.

Stresses that contribute to SCC are due to operational and/or residual stresses. Higher stress increases the susceptibility for SCC. Pipe cracking can also be initiated at surface discontinuities (e.g., welded pipe support lugs, pits, rough ground areas, and crevices created by mechanical or welded joints). These areas can have higher residual stresses and altered microstructures that are susceptible to SCC (particularly in the case of the welded materials). However, these areas can also be occluded areas where the local environment can evolve into a corrosive environment and become different from the bulk environment. For pitting corrosion, such as under pipe support clamps, the stress component of ODSCC may come from stress concentration points in pits in combination with the operational stresses such as pressure and temperature.

SCC can be managed effectively to minimize the potential for catastrophic pipe failure through stainless steel piping cleanliness control and limiting contact with fluids or condensation. Water chemistry control can be used to minimize the adverse effect of oxygen and chloride on SCC. Periodic inspections of the susceptible piping systems should be conducted or included as part of routine walkdowns.

IRS 8193 – CRACK INDICATIONS AT THE REACTOR WATER CLEAN-UP PUMPS

Transgranular crack indications were identified in the sealing area of an O-ring seal of the brackets (thermal barrier) of the reactor water clean-up pump. Six months earlier a previous report had been issued from another plant identifying similar crack indications in the sealing areas of the O-ring seals of the brackets (heat barriers) of reactor water clean-up pumps.

The pumps affected are an integral part of the reactor water clean-up system and have the function of transferring reactor coolant into the reactor water clean-up system. The affected pumps were single-stage centrifugal pumps of the same type from the same manufacturer that have flange gaskets that are designed as double seal with an inner (primary) graphite flat gasket and an outer (secondary) O-ring seal made of PTFE (Teflon®), and with suction grooves between the inner and outer seals. The pumps are supplied with hot reactor coolant of approximately 280 °C at a pressure of around 80 bars at the primary gasket. The brackets are cooled with water at about 30 °C to protect the electric drives against high temperatures.

Circumferential linear indications were detected on the reactor water clean-up pumps in the seal area of the bracket, mainly in the groove base (seat of the O-ring) and on the groove flanks. The cracks were particularly evident in the area of the welded-on paw supports. In addition to the linear indications, there were also indications of local shallow pitting corrosion and circumferential pitting corrosion on the O-ring support surface. The cracks started at the groove flanks and in the groove base of the O-ring seating and developed into nearly axial direction (in relation to the pump axis), the maximum crack depth was 13 mm at the bracket of pump.

In both nuclear power plants, the investigations showed no indications of chloride, but there was a high concentration of fluoride in the crack areas. Transgranular crack formation at the reactor water clean-up pumps can be attributed to the influence of fluorides generated by the decomposition of the seal material of the O-ring seal. A prerequisite for the damage mechanism is that the area of the O-ring seal was wetted with an aqueous medium.

In both cases there was evidence of the influence of coolant. The leakage suction pipes are located at the bracket at the lowest point and lead about 1.5 m upwards in the valve area. Due to condensation processes, the leakage suction pipes could fill with water and thus reach the space between the inner and outer seal.

In it is believed that the O-ring seals made of the material Teflon® (PTFE) decomposed under the influence of the temperature and, in particular, the radiation.

Due to the temperature and radiation the aqueous medium accumulated in the area of the O-ring grooves which then caused local corrosion attacks on the austenitic steel with increasing fluoride concentration. Starting from local pitting corrosion or corrosion or groove surfaces, cracks could develop in particular at locations where there were stress concentrations due to residual welding stresses on the support paws.

As a result of these events, other areas which contained components and in particular seals made of fluorine containing materials which could be subjected to high temperatures, radiation, humidity and aqueous medium were to be identified and examined and where necessary changed with a more suitable replacement.

## 3.3. Leakage

Prevention of leakage is one of the performance objectives of the nuclear power plants in order to maintain a high standard of safety, performance, maintenance quality, and radiation protection. Lack of a mechanism for identification of manufacturing defects in steam generators tubes can lead to leakage from the primary to secondary circuit.

The important issues identified in other events are deficiencies in aging management programmes, deficient procedures to address issues on aging components, incorrect maintenance practices, and inadequate corrective actions for precursors for the leakage such as inspection of components of similar design with known deficiencies, boron deposits, etc.

Compared with the previous highlights, the number of leakage related events with important lessons has considerably decreased; however, there is a need to continue to pay attention to minimize leakage events through aging management of components, strengthening preventive maintenance practices, and through training of plant personnel.

IRS 8081 – UNIT SHUTDOWN DUE TO STEAM GENERATOR TUBE LEAKAGE

Steam generator (SG) radiation monitoring system alarms actuated in the control room during plant operation showing increased gamma dose rate on SG secondary sides. The central radiation monitoring panel indicated volumetric activity increase in the SG secondary side blowdown water upstream of the filters. Based on the results of spectrometric analysis of the blowdown water, the primary-to-secondary leak was estimated to be 4.5 kg/h while the Iodine 131 specific activity was below the minimum detectable level. The unit was brought to cold shutdown to eliminate SG leak.

The root cause of the leak in the heat exchange tube was attributed to a manufacturing defect in the form of a burn which resulted in metal damage during operation through exposure to primary circuit coolant and consequent removal of the metal.

The corrective actions taken were aimed at: plugging the leaky tube from both sides, leak-tightness monitoring of the welds attaching the plugs to SG headers during primary system hydro-testing, performing SG chemical flushing, and conducting external TV examination and eddy-current testing of SG tubing.

Plants should closely examine the manufacturing defects of the SG tubes, preferably at the early stages of operation to avoid any functional degradation of SG tubes during further plant operation.

IRS 8141 – LEAKING OF SEAWATER FROM SEAWATER PIPING FOR EMERGENCY DIESEL GENERATOR COOLING

During power operation, leakage was found in a seawater pipe of the emergency diesel generator-B cooling system. Afterward, one of the two trains of the emergency diesel generators (EDG) was put out of service and the affected pipe was replaced with a new one. Investigation revealed that flaw found in the EDG cooling seawater pipe and the cross-sectional observation of the defective pipe identified corrosion in the base metal of the pipe, which developed from the inner surface to the outer surface, and a small through-wall crack in the lining material (polyethylene) on the inner surface of the pipe.

The examination of the maintenance records for the affected pipe suggested that the initial crack in the lining was generated by an impact load when the lining was hit by a tool used to remove marine creatures attached to the inner surface of the pipe. This impact load triggered the initial crack between the lining surface and the air bubble. The crack developed due to residual tensile stress and penetrated the lining layer. Then seawater infiltrated through the crack in the lining and corroded the carbon-steel pipe. The corrosion developed from the inner surface and finally penetrated the pipe wall.

Precautions were included in the work manual to ensure that impact load would not to be applied to the lining on the inner surfaces of pipes during inspection work. In addition, the integrity of the lining will be confirmed after inspection. The work manual was improved to draw up an inspection report which clearly describes the subjects, focuses and results of lining inspections.

Power plants should clearly define the precautions to be taken during the maintenance work so that the equipment integrity may not be affected having their effects on long term operation of the equipment.

## IRS 8172 – LEAKING OF PRIMARY COOLANT THROUGH NOZZLES OF UPPER UNIT OF THE ENERGY MEASUREMENT CHANNELS

Two incidents of leakage of primary coolant from the nozzles of the upper unit energy measurement channels (EMC) occurred. The first incident occurred when the plant was at full power while the second occurred at minimum controllable power level.

The leakage was observed during inspection of the flange of the channels which revealed a leak (steaming of the primary coolant) on the flange of one channel while no problem was noted on the other channel.

During the investigation it was established that the primary coolant had leaked through the gap between the dummy plug of the neutron measurement channel and the clamp nut. The leakage of primary coolant (at a diminishing rate as the pressure decreased) lasted about 24 hours. An inspection of the space under the cap in the upper unit of the reactor was performed. The inspection revealed traces of crystallized boron (leakage of primary coolant) on the nozzles of EMC-1 and -3. It was established that the leakage of working medium was due to multiple micro displacements of the sealing components relative to each other.

Both incidents were caused by plant maintenance service management’s inadequate monitoring of maintenance processes, the failure to take into account ageing of components of normal operation systems, deficiencies in technical maintenance procedures, and inadequate quality control of maintenance.

Detailed maintenance procedures and their implementation may prevent this leakage of primary coolant.

## IRS 8173 – PRIMARY COOLANT LEAK FROM COOLANT PURIFICATION SYSTEM VALVES

During power operation, area radiation monitors identified some leakage; however, there was no change in the parameters of the primary and secondary circuits. The plant staff conducted all the necessary inspections to locate the leak and found that there was steaming from the primary circuit water bleed valve from the RCP pressurized line. Two drainage valves of the 2SVO-1 facility were dripping the drain in area A-013/2 was opened and all the water drained off. Load reduction of the unit was initiated.

The leak in the primary circuit water bleed valve from the RCP pressurized line was caused by layering of the metal in the valve body which, upon temperature displacement of the layers of metal, led to escape of coolant in the steam phase. The cause of the layering of the metal was repeated overheating (to melting point) when welding the joint during maintenance of the valve body. The primary circuit water bleed valve from the RCP pressurized line was made and installed in 1979, i.e. it has been operating for over 30 years.

The incident was attributed to inadequate monitoring on the part of plant maintenance service management of maintenance processes, failure to take into account ageing of components of normal operation systems, deficiencies in technical maintenance procedures and inadequate quality control of maintenance. The failures could have been prevented with stricter control of the organization of maintenance, including the process for managing the ageing of components of normal operation systems.

### 3.4. Diesel Generators

Reliability of diesel generators (DG) as an onsite emergency or SBO (Station Black Out) source of electricity is considered vital for fulfillment of safety functions. This safety factor is also being considered as one of the most important safety improvements that should be adressed worldwide in the post Fukushima scenario. The previous biennial Highlights report focused on issues related to voltage regulator problems in emergency diesel generators, common cause failure in emergency diesel starter motors, complete loss of offsite power in combination with the unavailability of both diesel generators resulting in a significant reduction of safety margins and the problems related to the conventional petro diesel specifications.

The present highlights report mainly focuses on the deficiencies in design, inadequate factory acceptance tests, and manufacturing faults attributed to some specific designs and specific manufacturers of the diesel generators.

Operating experience feedback from a series of plants revealed five cases of damage to diesel engines requiring standard replacement. Given the safety-related role of emergency and SBO generators and the importance for safety of the diesel engines used in these generators, the manufacturer must focus its full attention on complying with the process for dealing with any modifications to equipment and on the in-factory inspections (end-of-manufacture inspections and tests).

With regard to maintenance for the specific manufacturer’s diesel engines, a number of points need to be followed up including: quality of maintenance at the factory and at the NPP sites, integration of operating experience feedback into preventive maintenance programs, and NPP attention needed for analysing any change of manufacturer or production process, especially when related to equipment used to ensure facility safety.

IRS 8106 – DAMAGE OF THE TURBOCHARGER OF ONE EMERGENCY DIESEL GENERATOR

During the surveillance test of the emergency diesel generator (EDG), a failure of the turbocharger was found and operation of the EDG was stopped. During subsequent overhauling of the EDG, damage inside the turbocharger was found resulting in the equipment inoperability.

Investigation into the event revealed that some of the nozzle retainers fixing bolts were not tightened with the prescribed torque during the turbocharger manufacturing process, because the bolt tightening method was not clearly described in the work procedure manual. Due to vibration during commissioning and load operation, the heads of the loosened bolts of the turbocharger contacted the rotor shaft and the turbine blades were damaged. As a result, the rotor shaft was de-centered and many inner parts of the turbocharger were damaged.

As a corrective action, the damaged turbocharger was replaced with specific consideration that the bolts be tightened with adequate torque, and the two turbochargers of the other EDGs were checked for proper tightening of the bolts with adequate torque. Further, the torque controls will be clearly specified to the supplier and the instruction/procedure manual will be revised to ensure the bolts are tightened with the necessary torque.

## IRS 8147 – FAILURE OF AN EMERGENCY DIESEL GENERATOR AFTER MANUFACTURER MAINTENANCE

In a test run of an Emergency Diesel Generator (EDG), the diesel switched off after 1.5 hours running time due to low oil pressure. The diesel engine had just been reinstalled in the plant after maintenance by the manufacturer. The affected diesel was manufactured in 1978 by the same company.

The inspection of the diesel showed that one master piston rod bearing had been destroyed and the associated slave rod bushing was broken and bent. Before the damage occurred, the diesel had been in operation 21 hours with 12 starts. Similar damage to same type of diesels is known from other countries. Here, the operating times were between 10 hours and 100 hours. The damages were investigated by designer and owners jointly. Accordingly, the damage mechanism was determined to be from damages to the master rod bearing bushings causing seizure of this bearing and twisting of the bushings. Due to the strong bearing abrasion, pistons hit the cylinder head, resulting in a piston deformation and seizure of the slave piston, and then the slave rod is torn off. In consequence of the broken slave rod, lubricant oil nozzles and the lower part of piston liner have broken.

The cause of the damage was probably due to the surface conditions (hardness, roughness) of the affected bearing bushings and thus a failure to adjust and smooth the running surface during running-in. The failure of other diesels with the same damage symptoms in other countries and the root cause analyses show that the failures occurred due to a systematic cause. Thus, several redundancies might be affected in case of demand at the same time if all available diesel generators would have been equipped with the unsuitable piston bearings mentioned. Due to the performance of maintenance and inspection of redundant emergency power diesels at different intervals and the fast damage progression during test runs with the piston rod bearings affected, a simultaneous failure of several diesels in case of demand is unlikely.

EDGs of the same manufacturer are also operated in other nuclear power plants. To ensure the functional reliability of these diesels, the piston rod bearing bushings with same reference numbers are to be replaced in the short term according to the manufacturer's recommendation.

## IRS 8164 – NON-COMPLIANCE OF ROD BIG END BEARINGS IN EMERGENCY DIESEL GENERATOR ENGINES

Failures occurred in diesel engines of emergency generators of the same manufacturer in various countries. Such damages to the diesel engine, causing engine seizure, are due to rapid degradation of a rod big end bearing made by a specific manufacturer. Upon expert appraisal of the bearing, it was found that the damage was due to the anti-friction coat being slightly too thick, possibly causing localized melting and deterioration. Sixteen engines in the same series of plants were fitted with bearings identical to the bearing in question.

Given the importance of the role played by these parts to ensure the smooth running of the diesel generator engines, the regulator declared a significant safety-related event classified Level 1 on the

INES scale and, as a preventive measure, decided to replace these “first generation” bearings with “second generation” bearings with the same geometry as the original bearings fitted in these engines. The defective “first generation” bearings were supplied by the diesel engine manufacturer, without undergoing prior qualification tests for use under conditions representative of their use in the diesel engines of the emergency generators used in nuclear power plants.

“Second generation” bearings had undergone in-factory qualification tests simulating 10 years’ of operation in a diesel engine identical to those installed in a series of NPPs. The inspections performed on several diesel engines at plants in this series revealed premature wear to some of these bearings. At two units in a plant in this series, the emergency generators and the station blackout (SBO) generator were fitted with these parts; this potential for common cause failure puts the on-site electrical power supply to the reactors at greater risk.

Taking this increased risk into account, in February 2011, the regulator declared a significant safety-related event classified Level 2 on the INES scale for these two units and Level 1 for the other units at which the emergency generators and the SBO generator are not both fitted with these bearings. The causes of premature wearing of the “second-generation” bearings have not, to date, been explained. The consequences of this phenomenon are to render the diesel engines unreliable thereby increasing the risk of a total loss of electrical power supply due to a common mode failure in the case where defective bearings are fitted in more than one engine.

### 3.5. Control Rods and CRDM

There were six event reports and one generic report related to degradation of systems required to control reactivity. Some of these events were also related to deficiencies in design, deficiencies in safety management / quality assurance system, deficiencies in safety evaluation, deficiencies in operation (including maintenance and surveillance) as well as generic problems of safety interest.

These events occurred mainly at boiling-water reactors and pressurized-water reactors. One event occurred at pressurized heavy-water reactor.

These events occurred with the reactor at full allowable power, in cold shutdown mode, and during refuelling/open vessel evolutions with all or some fuel inside the reactor. Plant status during some reports was not applicable.

Most of the events had no significant effect on operation or were not relevant. Several events caused manual load reduction. Some events led to outage extension.

Compared to the previous period, the number of control rods and CRDM related reports with important lessons learned increased, showing a negative trend.

IRS 8105 – OVER-INSERTION OF A CONTROL ROD DURING PERIODIC INSPECTION

With the boiling-water reactor in cold shutdown and under periodic inspection, during the restoration work after the inspection of the hydraulic control unit for the control rod drive hydraulic system, an alarm indicating unintentional control rod movement was actuated for one control rod out of 137 control rods. It was determined that the control rod was further inserted beyond the normal full-insertion position (over-insertion). Then the control rod immediately returned to the normal full-insertion position. This event had no radiological impact on the environment.

Estimated causes:

- Leak test of the hydraulic control unit of the control rod drive hydraulic system confirmed a small seat leak in a directional control valve.
- As a minute flaw was found on the seat surface of the valve by overhauling the concerned directional control valve, it was estimated that the small seat leak was caused by intrusion of foreign material.
- As a result of a mock-up test (a verification test), it was confirmed that a control rod could be over-inserted when a foreign material with a diameter of about 0.26 mm deposits onto the seat surface.
- The investigation on the possibility of foreign material intrusion showed that the filter on the control rod drive water inlet side of the concerned directional control valve was replaced with new one, unlike other filters, which are reused in rotation. Therefore, as the filter for that position goes through more work steps than other filters, from ultrasonic cleaning in the factory to installation on site, a foreign material may have come into the valve during those steps.
- Consequently, the foreign material may have deposited on the valve seat of the concerned directional control valve, when the valve was opened (and then closed) for scram functional test. Because of seat leak due to the foreign material, drive water to insert the control rod flowed for a short time during the restoration work of the system and thus the control rod was over-inserted (beyond the normal full-insertion position). After that, the control rod may have returned to the normal full-insertion position by its own weight because the foreign material was removed by drive water.

The corrective actions implemented by the plant included replacement of the directional control valve with new one and cleaning of the filters. After that, functionality of the valve was confirmed by operating the control rod drive mechanism. As a preventive measure against foreign material intrusion, ultrasonic cleaning of new filters as well as reused filters will be carried out.

IRS 8109 – ADJUSTER ROD SPURIOUS OUT-DRIVE

While the unit, a pressurized heavy-water reactor, was operating at high power, one adjuster rod spuriously drove out of the reactor core. There was a very short duration flux tilt alarm at the beginning of the transient. The operator noted the alarm and executed the procedure, which required shutting down the reactor by setting the reactor regulation system alternate mode power set point to -3 decades. The event was terminated approximately 7 minutes after its initiation. The unit was placed in the low power hot state.

The event was classified as an unrequested power increase. The scope of the event investigation report primarily focused on procurement issues and was not broad enough to include the actual root cause of the event, which was a fault in the design of the logic module that may inhibit the module from ‘failing safe’ under certain circumstances. The replacement of the logic module power supply reduces the likelihood of reoccurrence; however, it does not eliminate the unsafe failure mode.

As a corrective action, the logic module will be redesigned to eliminate its unsafe failure mechanism in the rod control circuit. In addition, an inspection on the procurement process will be performed.

IRS 8111 – INADVERTENT INSERTION OF CONTROL ROD DURING RESTORATION OF HYDRAULIC CONTROL UNIT OF CONTROL ROD DRIVE HYDRAULIC SYSTEM (EVENT 1)

With the boiling-water reactor in cold shutdown and under periodic inspection, when a stop valve for the control rod drive water was opened during the post-inspection restoration of the hydraulic control unit of the control rod drive hydraulic system, a control rod drift alarm was actuated for one control rod. It was reported that the control rod was over-inserted beyond the normal full insertion position. This event, related to degradation of systems required to control reactivity had no radiological impact on the environment.

The cause of the control rod drift alarm was unintended over-insertion of the concerned control rod due to a seat leak in the scram inlet valve generated during the post-inspection restoration work of hydraulic control unit of the control rod drive hydraulic system. The concerned valve was overhauled during the current periodic inspection. When the stem stroke was adjusted during re-assembling, the valve disk slightly shifted from the fully closed position to open direction. Consequently, seat leakage occurred since the valve disk did not seat completely on the valve seat even when brought to the fully closed position.

As a corrective action, during re-assembling of the valve, a marking will be made on the valve stem in order to correctly position the stem by confirming the mark when adjusting the stroke. This procedure will be reflected in the maintenance instruction manual of the valve. For the concerned valve, stroke adjustment will be carried out again in accordance with the revised maintenance instruction manual. In addition, the same type of valves as those overhauled during the current periodic inspection will be overhauled again to conduct stroke adjustment as a precautionary measure.

IRS 8112 - INADVERTENT INSERTION OF CONTROL ROD DURING RESTORATION OF HYDRAULIC CONTROL UNIT OF CONTROL ROD DRIVE HYDRAULIC SYSTEM (EVENT 2)

With the boiling-water reactor in cold shutdown and under periodic inspection, a control rod drift alarm was actuated for one control rod when the stop valve of the control rod drive water inlet line was opened for the restoration work after the inspection of the hydraulic control unit for the control rod drive hydraulic system. The operators judged it to be over-insertion of a control rod beyond the normal full-insertion position. This repeat event, related to degradation of systems required to control reactivity had no radiological impact on the environment.

The cause of the actuation of the control rod drift alarm was over-insertion of the control rod due to additional insertion pressure generated by drive water from a seat leak of the scram inlet valve, which was restored after the inspection of the hydraulic control unit for the control rod drive hydraulic system. The overhaul of the concerned valve identified partial losses of the Teflon-made valve seat. In addition, it was confirmed that in both this event and the previous one, old- and new-type stud bolt sets were simultaneously used for assembling the valve seat assembly. The investigation of mixed use of different types of stud bolt sets showed that assembling the seat with a mixture of different type of bolt sets causes non-uniform inward protrusion of the Teflon seat, since the axial clamping force differs between the different types of stud bolt sets.

The cause of the current event was estimated as follows. When stud bolts were tightened, the axial force of two old-type stud bolt sets on the 180-degree side tended to become relatively larger than the axial force of two new-type stud bolt sets on the 0-degree side. This imbalanced axial force resulted in a slanting gap between the valve body and the tail piece and, to compensate this, the two bolts on 0-degree side were further tightened with larger torque, which led to larger protrusion of the Teflon seat.

As a corrective action, for the valves where both old- and new-type stud bolt sets are used, all the stud bolt sets will be replaced with new-type ones. In addition, to assure the disk-seat contact, the procedure for stem stroke adjustment after re-assembling the valve will be improved to secure stroke margin by pushing down the stem before coupling it to the drive shaft, instead of the current procedure for securing the stroke margin by rotating the valve stem after coupling.

## IRS 8177 – CONTROL ROD SWELLING

Excessive wear and swelling phenomena have been observed on rod cluster control assemblies for a long time, leading power plants with pressurized-water reactors to improve their maintenance programs and increase monitoring of this equipment.

Incomplete insertion of rod cluster control assemblies into the lower part of the fuel assembly guide tubes has been observed since 2006. Analysis of these anomalies has established that they originate from a swelling phenomenon resulting from radiation. Since the kinetics appears more significant than expected, the maintenance strategy criteria were reassessed. To overcome excessive swelling, the operator defined a new maintenance strategy consisting of limiting the operational life of rod cluster control assemblies in the nuclear fleet.

Lessons learned from these weaknesses reveal a need to pay attention to control bank rod behaviour, with particular attention paid to rod cluster control assemblies most subject to neutron flux. In particular, mandatory criteria shall be established to limit the operating life of rod cluster control assemblies and there shall be specific monitoring to readjust criteria for replacing rod cluster control assemblies in case of faster than expected swelling.

## IRS 8201 – CONTROL ROD BLADE CRACKING RESULTING IN REDUCED DESIGN LIFETIME

An information notice was issued by the NRC (IN 2011-13) to inform the industry that in 2010 severe cracking in control rod blades near the end of their nuclear lifetime limits was discovered in international boiling-water reactors. The cracks were much more numerous and had more material distortion than those observed in previous inspections of control rod blades. The cracks were also more severe, resulting in missing boron-carbide capsule tube fragments from some of the inspected control rod blades. (A lost parts analysis determined that the missing fragments caused no negative effect on plant performance.) Additionally, these cracks occurred at locations of lower reported local boron-10 depletion than previously documented.

The identified cracking was attributed to irradiation-assisted stress-corrosion cracking that results when a material that is susceptible to irradiation is in an aggressive environment from oxidizing boiling-water reactor water and experiences excessive stress because of boron-carbide swelling. It was determined that a significant contributor to the extensive cracking was a rapid thermal transient that occurred when the automatic depressurization system actuated and injected cold water.

As a result of investigations into the cracking, it was determined that the design life of certain control rod blades may be less than previously stated and the manufacturer is revising the end-of-life depletion limits of these control rod blades.

## IRS 8213 – ERRONEOUS INSERTION OF A CONTROL ROD DURING PERIODIC INSPECTION OUTAGE

With the boiling-water reactor in cold shutdown and under periodic inspection, a control rod drift alarm was activated in the main control room when a valve on one control rod drive water line was operated as a part of restoration work for the control rod drive hydraulic control unit. A check was made as to whether the control rod actually moved, but no anomaly was found in the control rod position indicator system. Therefore, it was presumed that the control rod was temporarily inserted by about 15 cm into the core from the fully withdrawn position, but came back to the initial position in a short time. During investigation, no defect was found during overhaul of the valves in the drive system (for the concerned control rod) of the control rod drive hydraulic control unit.

The causes of the event were estimated as follows. During the periodic inspection, a small amount of air was entrained in the water charging line and was pressurized by the addition of the nitrogen gas that leaked from the accumulator into the water charging line. When the reactor protection system interlock function test was implemented, the scram inlet valve was opened and the pressurized air/nitrogen mixture moved into the insert line of the control rod drive mechanism. When the stop valve on the insert line was opened, the line pressure was relieved and the volume of the pressurized air/nitrogen mixture expanded, which caused insertion of the control rod. The control rod was brought back to the initial fully withdrawn position in a short time by the weight of the control rod and control rod drive mechanism.

As a corrective action, before filling the accumulator of the control rod drive hydraulic control unit with water, the accumulator drain valve will be opened to prevent pressurization of the water charging line. After filling the accumulator with water, pressure in the pressurized parts will be relieved prior to restoration of the control rod drive hydraulic control unit. In addition, the pressure relief procedure for this countermeasure will be reflected in the operation manual.

### 3.6. External Hazards

During the present biennual period a series of events related to the blockage of water inlet facilities were identified. Typically this blockage was caused by a combination of bad weather and poor design. Opening of upstream dams were also a potential source of debris.

These events illustrate that the amount of debris in inlet water can increase rapidly. Various blocking agents include: algae, seaweed, silt, and frazil ice. These events may represent a common cause failure because degraded heat transfer capabilities can occur at multiple trains or in several units. Though only a few of the most significant events are discussed below, more are discussed in the section on degraded cooling conditions and several other events not included here showed similar characteristics.

All these events illustrate the susceptibility of the safety significant service water system. Blockage of water inlet facilities may compromise cooling capability, especially for the removal of residual heat. Maintaining the ability to cool the core with the Ultimate Heat Sink is critical for ensuring the safety of the plant. Depending on the severity of the event, additional emergency means may need to be rapidly mobilized. The implementation of a predictive blockage risk indicator system could be beneficial for a prompt proactive response.

In another event a fire near outgoing transmission lines led to the tripping of a reactor. The area on the hill where the fire occurred was overgrown. The operator did not have the authority to clear the vegetation as the hill was under the jurisdiction of local government. However, the operator failed to recognize the significance of this hill fire hazard.

At another plant valve operation was blocked due to icing when the area experienced low temperatures. The minimum temperature established in the FSR was inadequate, as temperatures lower than this has been experienced in the past. This event illustrates that known weather extremes need to be considered in the design and operation of the plant.

IRS 8068 – TOTAL LOSS OF THE HEAT SINK FURTHER TO CLOGGING OF THE ROTATING DRUM SCREENS BY A VERY SIGNIFICANT ARRIVAL OF VEGATABLE MATERIALS

A total loss of heat sink at one unit, and partial loss at two additional units, occurred due to clogging of the trash rack, prefiltration system and rotating screens by plant material. The particular plant, “Canadian Pondweed” (Elodea canadensis), had never been observed at the station prior to this event. The massive arrival of Elodea occurred due to the opening of a dam upriver.

Difficulties were encountered in managing the situation of multiple heat sink losses due to the occurrence of this unexpected external hazard. The monitoring and protection measures currently in place at the pumping station were unable to detect the rapid clogging of the trash racks and filter screens that caused the event. As the fixed measures for cleaning this system were insufficient, additional mobile equipment had to be brought onsite.

This event represents simultaneous failure of both redundant trains of the essential service water. This event demonstrated that simultaneous partial losses of heat sink on two plants, and total loss of heat sink at one plant is possible given rapid occurrence of external hazards.

Effective management of similar events requires detection of the phenomena, diagnosis of the situation, and implementation of fast and reliable means of mitigation. Depending on the severity of the event additional means may need to be rapidly mobilized. The investigation of this event found that heat sink monitoring does not necessarily ensure fast and reliable detection of its loss in the event of massive arrival of clogging agents at the trash racks.

IRS 8179 – 500 KV OUTGOING LINE TRIPPED ON ELECTRICAL PROTECTION CAUSED BY HILL FIRE

While the plant was in full power operation, a fire was observed on a hill forest near the 500 kV outgoing transmission line. A couple of hours later, this line tripped due to differential protection. Smoke from the fire had reduced the air insulation of the line and allowed an electrical short from phase to ground. Unit power was maintained on house load and class 3 power busses were manually transferred to two diesel generators.

The next day the fire was extinguished and the transmission line was inspected. No damage was found on the transmission line and it was reenergized.

Because of inadequate estimation of the significance of this hill fire hazard, no safety separation area had been established along the outgoing lines, and the hill was administered by the local government. The plant had no right to cut or prune vegetation that encroached upon the lines.

As a result of this event, which resulted from a hill fire, the affected plant has been working in conjunction with the local government to establish a safety separation area along the outgoing lines and to perform regular pruning of vegetation in the area.

IRS 8186 – POWER REDUCTION DUE TO TRIP OF TURBINE-DRIVEN FEED WATER PUMP AS A RESULT OF ALGAE AND SILT INGRESS INTO THE CIRCULATION WATER SUPPLY SYSTEM

High winds resulted in ingress of slime, bottom sediments, and algae into the rotating drums of the unit pump station. Emergency teams were engaged to clean the drums and their screens. A significant amount of debris remained on the drums despite these efforts, resulting in reduced cooling water flow in the condenser of one turbine driven feed water pump. The pump tripped and reactor power was reduced to 50 %.

The design of the intake channel lacked adequate protection against silt and algae ingress into the drums of the pump house. Also monitoring of the cooling pond was inadequate. Debris was not regularly removed and the design depth of the pond was not maintained.

Subsequent to the event the drums were re-enforced, design changes to the protective structures in the cooling channel were performed, and the depth of the cooling pond and intake channel were increased.

IRS 8200 – INOPERABILITY OF BOTH INDEPENDENT CIRCUITS WITHIN THE ESSENTIAL SERVICE WATER SYSTEM DUE TO SAFEGUARD COOLING TOWER VALVES BLOCKED BECAUSE OF COLD WEATHER

Both independent circuits of the essential service water system became inoperable due to blockage of the valves feeding the safeguard cooling towers' wells from the make-up water supply. Low temperatures had resulted in freezing of stagnant water in the upper portion of the valves near the valve plug.

The minimum temperature established in the FSR was inadequate (0 °C). Temperatures lower than this has been experienced in the past, even for long periods. It is possible that similar impairment of valve operation may have occurred in the past and gone unnoticed.

The procedures against freezing were incomplete. The affected valves were not included in the scope of the procedure. No safety analysis was performed in the development of this procedure.

The station did not make use of available Operating Experience as the period established for reviewing external events was small. As a result relevant events were missed. Also extreme temperatures were not considered as initiating events.

As a result of this event several actions were taken. The procedure against freezing was expanded to take into consideration all equipment that could be affected by low temperatures. A comprehensive process to review procedures and identify systems, components or equipment whose operability could be impacted by severe weather conditions was implemented.

Some design changes were performed as well. Canvas and heating was provided to the valves and pipes of safety related systems. The minimum design temperature in the FSR will be reviewed to take into account temperatures historically encountered at the NPP location.

### 3.7. Degraded Cooling Conditions

During the last biennial period, one of the events involved a total loss of spent fuel elements normal coolant system for a significant period. Another event involved failure to properly maintain water chemistry in the cooling water spray ponds leading to common cause failure due to extensive fouling/scaling of the interior surfaces of heat exchanger tubes. The extensive fouling seen in these events represents a potential for common cause failure because degraded heat transfer capabilities occurred on all trains in all units.

Loss of the ESW system may be a significant contributor to the potential for a core damage accident. Another event involved frazil ice in cooling water channels leading to reactor scram, because procedures were not adequate for very fast decreases in sea temperature. Frazil ice was formed at the screens and partly blocked the cooling water screening system. A similar event occurred within the current biennial period.

All of the events in the current biennial period involve the blocking of cooling water inlet. In the majority of cases the blockage was caused by the ingress of plants, silt, or other material. One event occurred due to the formation of frazil ice. The generation of the blocking material tends to be associated with incoming tides, weather events, or with the operation of upstream dams. It would be beneficial to identify these initiating events in advance, so that measures can be taken to prevent blocking of the inlet.

In one event a series of factors contributed to the generation of debris. Heavy rain caused debris to wash into the inlet pond from tributaries. This debris was pushed to the inlet by high winds. Then a mudslide at the discharge channel caused additional debris to flow unexpectedly back to the inlet. The overflow of the discharge channel had not been anticipated, since extreme weather conditions like those in the preceding hours had not been anticipated.

The design of protection measures at these plants was insufficient to deal with a rapid accumulation of silt, debris and ice. This indicates that the worst case scenario for the generation of debris was underestimated.

### IRS 8075 – SEAWEED INGRESS LEADING TO PARTIAL LOSS OF SAFETY RELATED REACTOR SEAWATER COOLING

After a relative calm weather period winds got stronger and together with an incoming tide led to massive seaweed ingress to the cooling water inlet. This resulted in a temporary, partial loss of the heat sink.

Changing weather conditions together combined with the incoming tide had been identified in the past as a risk for seaweed ingress and deterioration of the heat sink. However, no early warning system was installed to cope with such situations. Additionally an automatic trip of the main cooling water pumps, intended to preserve cooling water for the safety related systems, did not occur.

This event highlighted the need for an early warning system for impending potential influx of seaweed and that operators need to be clear on the actions to be taken should a large ingress of seaweed occur.

Training was provided to all operators, and procedures were modified to emphasise the importance of maintaining the safety significant cooling in preference to other loads.

This event led to implementation of a seaweed risk indicator system which demands proactive actions to be taken based on tidal, wind direction, and wind speed indicators.

### IRS 8085 – OUTAGE DUE TO EXCESS SUSPENDED MATERIAL IN PLANT COOLING WATER (LAKE WATER)

A pressure decrease in the service water system was caused by fouling of the filters after the bypass gates of the rotating grids in the pump house opened due to the retention of suspended material.

This incident began when tributaries introduced debris near the water intake area after an unusually heavy rainfall. Moreover, a mudslide downstream of the discharge channel introduced sediment and debris into the discharge channel. This mudslide caused the discharge channel to overflow allowing additional debris to reach the lake near the intake of the pump house. In addition to the two sources of sediment and debris, a strong wind was also present that pushed debris toward the intake. The debris overloaded the rotating grids and caused the bypass gates to open. This led to a sudden fouling of the filters in the common water intake for service water and circulating water.

The self-cleaning rotating grids have a cleaning system that is automatically activated in response to an upstream/downstream difference in water level. At the time of the incident the system was operating in manual due to deterioration of the grids and a desire to preserve them.

The overflow of the discharge channel that caused the water's route to be bypassed had not been anticipated, since extreme weather conditions like those in the hours preceding the incident had not been anticipated. As evidenced by this event, the weather conditions used as the basis for the plants design and operation were not always the worst possible.

## IRS 8087 – MASSIVE INGRESS OF PLANT DEBRIS INTO RAW WATER PUMPING STATION

This report describes a period in which station operation was severely disrupted by frequent circulating water pump trips due to fouling of drum screens by plant debris and sediment. In three cases this ingress also led to reactor trips. The reactor trip sequences took place correctly and essential service water pump operability was maintained.

Flooding earlier in the year had dislodged sediment that had accumulated over several years in the estuary. Dredging operations were not performed following the flood because of new environmental regulations aimed at preserving certain animal species in the estuary (allowing migration and hatching). These regulations banned dredging in January and February.

Because of the frequent number of reactor trips as a result of drum clogging, it was decided to implement new surveillance and operating measures. These measures included: new criteria for detecting debris likely to clog the drum screens, a procedure to anticipate plant shutdown before circulating water pump trip, and modification to the annual dredging profile. In March an additional dredging operation was performed around the intake to reduce the amount of plant debris there.

This incident highlighted the difficulty in predicting the impact of all environmental factors while emphasising the need to monitor them.

## IRS 8095 – ICEING OF PROTECTIVE GRID AT CHOOZ B PUMPING STATION

Very low winter temperatures led to the formation of frazil ice and, for the first time at the site, icing of the protective grid located at the water intake, causing a difference of two metres between upstream and downstream water height. Blockage of this grid by ice lowered the water level at the entrance to the pumping station below the safe lowest water level, though it did not cause the essential service water pumps to malfunction. However, a slightly lower water level in the intake canal could have had significant safety-related consequences by affecting the performance of these pumps.

As soon as the icing of the grid was detected, the operator took action to break up the ice. This resulted in a quick return to the normal conditions at the pumping station.

Recirculation in winter was included as part of the design to eliminate the risk of icing inside the pumping station. However, since the risk of rapid icing of the protective grid from frazil ice was not considered, no protective measures against this particular risk were taken.

This event is comparable to an event (IRS 7921), which was discussed in the previous IRS Highlights.

The event recalls the importance not only of observing safety requirements for climatic hazards, including frazil ice, but also of regularly examining the adequacy of protective measures against these risks.

## 3.8. Electrical Equipment Events

Failures in electrical equipment can lead to unanticipated plant transients and the failure or unavailability of safety related equipment. They can also affect equipment important to safety or can potentially challenge safety related equipment. In addition there is the possibility for severe injury to personnel. The events identified below are result from many different causal factors including management of safety, poor working practices, use and adherence to procedures, failure to follow vendor recommendations and self-checking or verifications not being performed and as such could have been mentioned in other sections within this IRS Highlights report. However it is felt that the increase in recent years of problems associated with electrical equipment has now become significant enough that it needs to be highlighted separately.

IRS 8143 – INADEQUATE ELECTRICAL CONNECTIONS

An information notice was issued by the NRC (IN 2010-25) to inform the industry of recent operating experience regarding four events that highlight the issue of inadequate electrical connections.

In the first event, an electrical fault occurred on a 6.9 kilovolt (kV) non-segregated bus while the plant was operating at 100 per cent power. The fault caused a main generator differential lockout, which resulted in a main turbine trip and subsequent actuation of the automatic reactor protection system. The non-segregated bus experienced a catastrophic failure and fire.

In another event workers replaced a cell in a safety related battery bank after which an undetected loose electrical connection rendered the Train B battery inoperable.

In this third example, plant personnel were performing TS surveillance requirements of a Class 1E battery when they discovered that its voltage was below the required value. Subsequently signs were discovered of a loose connection on the breaker that provides charging current for the battery in its normal configuration.

In the last example station personnel identified that the open indication light for the normally open primary containment isolation valve was flickering. Investigation led to the discovery of an intermittent high-resistance electrical connection for control power to the valve. The intermittent electrical connection caused the indicator light to flicker. The intermittent connection would have prevented valve closure from the control room.

In all the above examples the root causes were identified as poor maintenance procedures and working practices which led to electrical connections not torqued as required or recommended by the vendor, verifications not being performed for critical tasks, quality control requirements not being applied and technical specifications not being met.

IRS 8185 – IMPACT OF ELECTRIC BREAKDOWN UPON ELECTRICIAN OF ELECTRIC DEPARTMENT FOLLOWING SHORT CIRCUIT IN 6 KV SWITCH-YARD DUE TO OCCUPATIONAL SAFETY VIOLATIONS DURING SWITCHOVERS

The duty electrician was given a verbal order by the shift supervisor to perform part of the operations indicated in the switching sheet (e.g. to roll out to the repair position the circuit-breakers for the connections and to connect the circuit-breaker cells to the grounding blades for bus 2RV) in order to affect repairs of bus 2RV and disconnector R-GSR-2. The electrician arrived at the location of bus 2RA in the 6/0.4 kV switching station, to prepare the work location for the authorized repairs but by mistake, instead of cell Vs 2RA-2RV, he started working in cell Vr-2RA, which is opposite cell Vs 2RA-2RV.

The electrician disconnected the stub of circuit-breaker Vr-2RA, rolled the circuit-breaker into the service corridor of bus 2RA, removed the protective hood with the board and disabled the operational interlock of the grounding blade without checking that there was no voltage in the cell. He connected the grounding blade which then caused a short circuit with formation of an arc and release of a flame from beneath the protective housing of the cell's cable compartment. The on-duty electrician suffered burns and was sent to the hospital for examination and treatment.

Many factors contributed to this event. The configuration of the equipment was not in accordance with the switching sheet. When performing the work, the duty electrician did not have the switching sheet with him as prescribed in the "Instructions for the performance of operational switching in electrical installations". The electrician failed to perform voltage checks to ensure the equipment was not live. The appropriate personal protective equipment was not worn. There was no pre-job brief prior to commencement of the work or adequate task supervision.

While there are some inherent problems associated with the design and layout of the equipment it is clear that many of the issues are associated with management of safety including training, supervision, self-checking and management expectations. The corrective actions applied looked at addressing deficiencies in these areas.

IRS 8224 – UNAVAILABILITY OF EMERGENCY FEEDWATER PUMP (EFP) DUE TO TRIP FROM OVERCURRENT OF ITS ELECTRICAL CIRCUIT BREAKER

A periodical test of the Emergency Feedwater System (EFS) was taking place when during test execution Emergency Feedwater Pump (EFP) N°2 tripped on its third start up. An investigation was immediately started and the breaker of EFP N°2 was isolated for inspection and the unit entered a Limiting Condition of Operation (LCO). Electrical operations staff performed a preliminary inspection of the electrical breaker but the cause of breaker trip could not be immediately determined and since no failure indication on electrical breaker protection actuation was found, a latent breaker failure was assumed.

The breaker was then inspected in the Electrical Maintenance Section (EMS) workshop and the periodic test of EFS was interrupted. The inspection determined that the breaker trip was caused by a long delay overcurrent protection (overload protection). The electrical breaker was replaced in the switchboard and was tested under test conditions and subsequently reconnected. A 10 minute test run of EFP N°2 was carried out with the electrical breaker performing as expected. The testing of EFP N°2 was resumed as per the requirements of the periodical test and was completed successfully.

Repeated start-ups/shutdowns of EFPs typically occur during the pump testing only. Three interlocks have to be checked which requires the start-up of the pump three times in succession. These start-ups are performed with energized motors in order to avoid unnecessary unavailability of the safety related equipment.

The testing procedure identifies that the OFF command, or OFF protection signal after start-up of the pump should not be generated sooner than 1 minute after a steady-state value of current is reached, so that interruption of the high run-up current is avoided. Furthermore the procedure also states "In the case of 3 subsequent starts of the motor from a cold state, a delay of 5 minutes is required before the 2nd and 3rd start; and in the case of 2 subsequent starts of the motor from a warm state, a delay of 5 minutes is required before the 2nd start. When this number of permitted start-ups is reached, a break of 20 minutes shall be made."

During investigation of this event it was determined from a test log that time delays between the first and second start-up of the EFP N°1 and N°2 were 2´52" (two minutes, fifty-two seconds) and 2´5", respectively. Time delays between the second and third start-up were 3´4" and 1´56", respectively. When the third interlock - pressurization of surge chamber was tested, four seconds after the third EFP N°2 start-up the actuation of overload protection tripped the EFP's electrical circuit breaker. However when EFP No: 1 was started for the third time, its electrical breaker did not trip on overload protection, and the fourth start-up occurred after a time delay11´30".

It is clear that for the testing of both EFP pumps that both the senior turbine operator in charge of test execution and an experienced unit shift supervisor did not observe the safety requirements for EFP test and the requirements included in the plant procedure. The senior turbine operator conducted a briefing prior to test commencement and informed control room operators of the administrative and safety measures which were to be followed during the test execution. However plant personnel involved in the test were not fully aware of the fact that a specific time interval between several consecutive EFP start-ups is required.

Corrective actions included the improvement of training to include lessons learnt from this event, the effective use of pre-job briefs, and the review and improvement of operating practices with regard to testing of the EFS power supply breakers to ensure the fulfilment of EFS' safety function for any incident situations.

## 3.9. Containment

The reactor containment is required to be operable to limit the leakage of fission product radioactivity from the containment to the environment. Recent operating experience highlights containment liner corrosion and pitting of PWR and BWR containments due to long term exposure to water and moisture, including that in inaccessible areas. Containment coatings applied to the steel containment serve to prevent or minimize loss of material thickness due to corrosion. However, once the coating material begins to degrade, the steel is susceptible to moisture which can lead to corrosion. Issues involving the degradation of coatings and corrosion of steel containments and liner plates due to the presence of water in inaccessible areas continue to be of concern for aging management. Implementation of the in-service inspection requirements for steel components of steel and concrete containments is necessary in order to ensure the containment will be able to perform its intended functions through the period of extended operation. Corrosion that originates between the liner plate and concrete is a greater concern because visual examinations typically identify the corrosion only after it has significantly degraded the liner. Several events were traced back to the time of original construction.

IRS 8117 – CONTAINMENT LINER CORROSION

An information notice has been issued by the NRC (IN 2010-12) to provide information on three incidents involving corrosion of the steel reactor containment building liner. In one case, during a planned visual examination of the interior containment building steel liner, a paint blister was identified. After cleaning, the corroded section was discovered to have penetrated through the entire liner plate thickness. The pitting-type corrosion (rust) was due to a piece of wood that was left behind as a result of inadequate housekeeping and quality assurance practices during the original construction of the containment wall in the early 1970s. In another incident, corrosion was caused by the installed felt that wrapped the outside of the containment penetration sleeve, which had become wet during the original construction. In the third incident, heavy corrosion on the containment liner just above the concrete floor was found during an inspection of the containment moisture barrier seal between the concrete floor and containment liner. Although this area of the containment liner was considered inaccessible, upon further review, examinations should have identified evidence of corrosion (rust on floor) and prompted removal of lagging to determine the source of the corrosion products. The source of the moisture that caused the corrosion was service water leakage from the containment fan coil units and associated piping.

These incidents provide examples of containment liner degradation caused by corrosion. Concrete reactor containments are typically lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. In-service inspections, including periodic visual examinations and limited volumetric examinations using ultrasonic thickness measurements, as well as, leak rate testing of the containment pressure-retaining components and isolation valves are required. This operating experience highlights the importance of good quality assurance, housekeeping and high quality construction practices during construction operations. A review design documents to identify locations where organic material was intentionally installed between the liner or penetration sleeve and schedule additional examinations of these areas to monitor for liner material loss is recommended.

## IRS 8206 – STEEL CONTAINMENT DEGRADATION AND ASSOCIATED LICENSE RENEWAL AGING MANAGEMENT ISSUES

An information notice has been issued by the NRC (IN 2011-15) to provide information on recent issues identifying degradation of nuclear power plant steel containments that could impact aging management of the containment structures during the period of extended operation. In two cases, degraded torus coatings resulted in corrosion of the steel substrate, which in some instances involved localized galvanic corrosion (pitting). The pitting corrosion rate can be 5 to 15 times faster than that of generalized (uniform) corrosion and is less predictable. In the cases discussed, the corrosion did not reduce the steel thickness below its design thickness limits, and the aging of the coating was being managing by repairing areas of coating degradation. In other plants, a complete recoating of the torus was planned.

This IRS report also addresses the discovery of water in the drywell air gap region in two other plants involving corrosion due to the presence of water in the inaccessible area of the sand-bed region of the drywell shell. The corrosion was likely caused by water in the gap between the drywell and the concrete shield. The source of water was leakage through the seal between the drywell and the refuelling cavity. Ultrasonic testing was used to detect wall thinning and determine corrosion rates. Routine inspection of the drainage capability of air gap drains in BWR containments (with or without sand pockets) decreases the likelihood of water being trapped. BWR Mark I containments should augment the monitoring and trending requirements of their aging management program to address inaccessible areas of the drywell.

## IRS 8207 – CONCRETE DEGRADATION BY ALKALI-SILICA REACTION

An information notice has been issued by the NRC (IN 2011-20) to provide information about recent issues of the occurrence of alkali-silica reaction (ASR) -induced concrete degradation of a seismic Category 1 structure. ASR can be identified as a likely cause of degradation during visual inspection by the unique “craze,” “map” or “patterned” cracking and the presence of alkali-silica gel.

Petrographic examinations were performed after concrete cracking patterns typical of ASR were observed. Concrete core samples demonstrated a substantial reduction in compressive strength and a modulus of elasticity substantially lower than the expected value. A prompt operability determination demonstrated sufficient margins to the code design limits, and therefore, the structural integrity of the control building remained. It is believed that the waterproof membrane was damaged during original installation or backfill activities causing water intrusion that resulted in the ASR problems. Water intrusion was exacerbated by the fact that dewatering channels were abandoned. If ASR-induced degradation is identified, condition monitoring should include determining the extent and rate of the degradation.

# 4. MANAGEMENT OF SAFETY

The events discussed in the management of safety section highlight the consequence of failing to maintain a strong organizational safety culture. Management oversight and timely communication of management expectations could have been instrumental in preventing many of the events presented in this section.

Communication is an important tool both within the plant and with external organizations such as contractors, manufacturers, vendors, and local governments for maintaining safety during all phases of nuclear power plant operation.

Procedures need to be accurate and provide sufficient detail so that activities can be performed safely by trained, knowledgeable personnel. Particular attention should be taken to make sure that procedures are maintained and periodically reviewed to incorporate changes in plant status and reflect lessons learned Poor procedures provide a disincentive for procedural adherence and can lead to further human performance issues.

Effective training with regards to changes in plant status is essential for successful implementation of modifications to the plant and procedures and applying lessons learned from operating experience feedback.

Based on an increase in the number of event reports with regard to deficiencies in the procurement process a new section was added to highlight the importance of a robust quality assurance program. Although fuel handling was a new section in the previous version of IRS Highlights no significant events in this area were reported during the current review period. However, there was a significant increase in radiation protection events including spread of contamination and excessive personnel exposure. Many of these events could have been prevented with stronger focus on organizational safety culture.

## 4.1. Human Performance

The focus in the human performance section is on events where personnel mistakenly operated the incorrect controls, resulting in plant transients. In all of these events, the incorrect action was accompanied and enabled by other factors which, taken together, can have a significant impact on the likelihood of human errors. These included deficiencies in training, a lack of clear and complete communications, and in three cases, inadequate procedures. These events show that human errors can lead to significant transients, and that these incidents can be significantly reduced when sufficient emphasis is placed on training, procedure development, proper communication, and adherence to procedures.

IRS 8096 – SLOW ELECTRICAL TRANSFER CAUSES REACTOR TRIP

A latent defect in a relay resulted in a temporary loss of offsite power, resulting in what should have been an uncomplicated plant trip. Actions that had been initiated earlier to work around known problems with a turbine trip feature placed excessive burden on unsupervised, non-licensed operators. Under the pressure of a stressful situation an operator failed to follow the procedure.

Following a scheduled test of an emergency generator, a relay which was known to be potentially defective but which had not been replaced, failed to open, resulting in a slow transfer to backup power and a temporary loss of offsite power to some safety systems. This caused a reactor trip. By design, the turbine would be allowed to run down for about 90 seconds before being tripped offline by reverse power protection relays in order to prevent motoring of the generator. However this protection feature had proven unreliable in the past, and instead of addressing the problem of the defective turbine trip mechanism, the utility changed the procedure to require that the turbine be manually tripped 60-90 seconds following a reactor trip. Due to control room staffing deficiencies, this task was assigned by procedure to a non-licensed operator. In this event, the non-licensed operator attempted to perform the procedure from memory and tripped the turbine early, 12 seconds after the reactor trip, resulting in an unanticipated plant response.

Several deficiencies were noted in this event, including a failure to replace relays known for several years to be potentially defective, inadequate control room staffing, and working around known problems. In this case, rather than fixing a known problem with the turbine trip protective function, the burden was placed on an unsupervised, non-licensed operator to take actions quickly following a reactor trip. The limited time available increased the likelihood that the operator would not follow the written procedure, and led directly to the situation where the turbine was tripped too soon. The utility was required to take action to address long-standing equipment deficiencies, improve control room staffing and supervision, and review the effectiveness of the root cause analysis program.

IRS 8125 – POWER REDUCTION ON TG-3 TRIP BY THE PROTECTION ON LO-LO LEVEL IN THE VOLUME CONTROL TANK OF GENERATOR SHAFT SEALING SYSTEM DUE TO PERSONNEL ERRORS

During normal full power operations while conducting a routine evolution to fill a generator lube oil filter, a series of operator errors resulted in a loss of lube oil flow to the generator shaft sealing system, resulting in a generator trip and power reduction to 50 %.

Operators conducting the procedure to fill the oil filter misinterpreted the non-specific direction of the procedure, causing oil pressure to fall, resulting in the start of the automatic standby pump (MNG-32). Once the original pump, MNG-31, was operating, the machinist attempted to disconnect MNG-32, but mistakenly shut the discharge valve for pump MNG-31 instead, causing it to become inoperable. Not realizing the error, the machinist reported that MNG-32 was disconnected, and with no indication otherwise, the control engineer shut down MNG-32. A lack of understanding of the behaviour of the emergency standby pump caused the engineer to also disable the interlock that would have caused that pump to start. With the discharge valve closed for MNG-31, the oil tank level indication fell to the level where the generator tripped off, resulting in a power reduction to 50 %.

The cause of the event was the machinist shutting the discharge valve on the wrong pump, although there were several deficiencies revealed here. These included inadequate training on operation of the generator lube oil system, an underestimation of the risk to the plant of potential errors when operating the system, a lack of clear communications between the machinist and the engineer to take action when unexpected indications were received, the absence of any indication on the control panel for generator lube oil parameters, and a maintenance procedure that relied heavily on operator knowledge of the system. After the event steps were taken to improve both the procedures for operating the generator lube oil system and staff understanding of the system.

IRS 8126 – REACTOR SCRAM DURING REACTOR PROTECTION SYSTEM TRIP TEST DUE TO INAPPROPRIATE OPERATOR ACTION

While operating at full power, technicians performing a routine test of the reactor protection system caused a reactor scram by pressing the buttons for the wrong channels.

While conducting the test for Channel C, the technician inadvertently pressed the button for Channel A. Recognizing the mistake, the technician attempted to reset Channel A to restore it to its previous position, but unknowingly reset Channel B, which was already in its normal position. This left Channel A in a "trip" position. The technician then resumed the testing on Channel C, and when Channel C was tripped, the two-of-three logic was satisfied to result in a reactor scram.

In this instance, two consecutive failures of the technician to appropriately verify which channel was affected resulted in reactor scram. This resulted from a failure to follow the procedure and the failure to use human error prevention techniques both before selecting the button to press and afterwards when verifying which channel had been affected by the action. Poor communications between the technician and the supervisor also contributed to the problem. Training after the incident focused on the effective use of human error prevention techniques.

IRS 8152 – INADVERTENT CONTAINMENT SPRAY DURING THE COMMISSIONING OPERATION

While heating up the plant following a shutdown during commissioning tests, a series of operator errors led to containment spray actuation resulting in a reactor coolant system level and pressure transient, manual initiation of safety injection, and the discharge of 423 tons of slightly contaminated borated water into the containment.

Operators attempting to restart the plant for further testing during plant commissioning had left one of two containment spray isolation valves out-of-position (open) because the procedure, carried over from plants that were already operating, did not account for design changes in the new plant. An operator in the main control room had earlier inadvertently placed the hand switch for the remaining containment spray isolation valve in the open position. Because of the work that had just been done though, this did not cause the valve to open. A test engineer at a local panel noted that following testing the previous day, the reset button for the containment spray system had not been pushed. Without consulting the control room, the engineer pressed the button. With the hand switch in open, this completed the logic to open the remaining containment spray isolation valve and initiate containment spray flow into the containment.

This event resulted from a combination of factors, including inadequate procedures and training. However, it was the action by the operator in the main control room that mistakenly placed the hand switch to open, and by the engineer to press the reset button without obtaining permission from the control room (which might have prompted verification of the hand switch position) that combined to cause the event. Procedures were updated to ensure that differences in the new plant design were accurately reflected, and management emphasized the need for staff to approach operations and nuclear safety at the new plant in the same way they would on an operating plant.

### 4.2. Procedures

A common theme seen in events involving procedural deficiencies was the failure to fully understand the potential risk of the evolution being performed. In these events, the activities had been performed before without problems, leading to a sense of complacency that since nothing had happened before, then nothing serious would happen. In two of the three events, the procedures assumed a level of experience with the evolution and did not adequately specify all the information necessary to perform the task safely. With incomplete analysis of the potential consequences of actions that could be taken while still within the scope of the procedure, slight variations in operator response, or performance of multiple procedures simultaneously, led to unexpected situations with consequences for both reactor and radiation safety.

IRS 8006 – DROP OUT OF A SELF-POWERED NEUTRON DETECTOR FROM TRANSPORT CONTAINER IN THE REACTOR HALL DURING REFUELLING OUTAGE

While lifting and moving a highly activated Self-Power Neutron Detector (SPND) tube and container during a refuelling outage, longstanding deficiencies resulting from an inadequate procedure caused the rope holding the SPND container to break, resulting in significantly higher dose rates in the reactor hall.

The SPND was being moved as scheduled from a shielded pit to the high activity waste storage pit using a lifting procedure which had been in place for years, but which depended on the skill and knowledge of the crane operator for successful completion. Longstanding deficiencies with the tube used to transport the SPND and with the complex procedure which allowed little room for error, had come to be accepted as normal by plant staff, and so no action had been taken to correct these deficiencies. The lift procedure required that two separate hoists operate in close coordination to avoid tilting the container, and did not designate a maximum speed for lifting the container, even though too high a speed could result in excessive swing of the container as the lateral crane movement stopped, placing excessive load on the rope. Furthermore, a risk analysis of the procedure was not required to be performed for operations that did not involve transporting fuel. Risk analysis might have noted the weaknesses in this procedure.

When the rope broke, the check-lock which should have prevented the SPND from falling out of the container was found to not have been inserted because of deformations that prevented easy installation. The SPND dropped out of the shielded container resulting in a sharp increase in exposure levels in the reactor hall. The reactor hall was evacuated and work ceased for several days while provisions were made for re-entry to return the SPND to a shielded container and return exposure levels to normal.

Procedural deficiencies had likely led to overloading the holding rope on several previous occasions, but had never been addressed as the rope had never actually broken before. Because the evolution had always been performed successfully in the past, neither these procedural deficiencies nor the degrading condition of the check-lock and the container itself were corrected. Past successful completions led to underestimation of the potential risk involved in carrying out this complex procedure, resulting in missed opportunities to improve the process.

IRS 8097 – LEAKAGE OF RADIOACTIVE LIQUID WASTE IN CONTROLLED AREA ON 2$^{ND}$ BASEMENT OF AUXILIARY BUILDING

While transferring liquid waste storage from a holding tank to a concentrated waste storage tank, operators failed to recognize initial indications of leakage until after sump alarms alerted them that blocked drains were causing liquid waste to leak out of the drain pit into the sump. Further review found contaminated liquid waste at three other points around the liquid waste pump rooms, with a total contamination activity of 1.2 GBq.

In 2006 operators facing an increase in liquid waste levels drained the liquid waste from the holding tanks through the drainage system without consideration for the effects that suspended solids in the waste could have on the drainage system. Unknown to plant operators at the time, the accumulation of suspended solids partially blocked the drainage system piping. When preparing the work plan for the 2009 waste transfer, the potential impact of suspended solids both from the previous transfer in 2006 and from the planned transfer was not appropriately considered. In addition, past operating experience indicating that the transfer should be stopped if the liquid observed turned brown or became more viscous, was not incorporated into the procedure. Operators observing the evolution noted the change in colour of the waste, but did not stop the evolution until they received the alarm indicating leakage. Investigation revealed that the suspended solids in the liquid waste had completely blocked the drainage system, causing liquid waste to overflow and leak into the waste pump rooms, spreading contamination.

The procedure used for the liquid waste transfer had not been verified against technical standards, and depended on the experience of the operators conducting the evolution to know when to stop. It also failed to give full consideration to the potential problems posed by suspended solids. Actions were taken to ensure that the drainage system would be used only for clear rinse water to prevent blockage by solids, and that procedures for transferring liquid waste with suspended solids would clearly evaluate the effects of these solids and use other methods for waste transfer.

## IRS 8116 – POTENTIAL FOR STEAM VOIDING CAUSING RESIDUAL HEAT REMOVAL SYSTEM INOPERABILITY

An information notice was issued by the NRC to provide information about an issue at three Pressurized Water Reactor (PWR) plants where on multiple occasions, their Residual Heat Removal (RHR) systems were inoperable because of the potential for steam voids at the RHR pump suction piping.

The water temperature in the RHR system could be as high as 350 º F when the system is being used for cooling of Reactor Coolant System (RCS). At this juncture, if the suction source is switched from the RCS hot leg to the refuelling water storage tank (as required to be done following a loss-of-coolant accident) or to the containment sump (as required to be done during extended response to a loss-of-coolant accident), conditions at the suction of the RHR pump would result in steam voiding due to the water temperature being above the saturation temperature at this location. Steam voiding can result in binding of an RHR pump/ refuelling water storage tank discharge check valve, system flow interruptions, and water hammer; potentially inhibiting the capability of the RHR system to fulfill its safety functions. Some PWR plants occasionally use multiple RHR trains to perform plant cooldown. In such cases, multiple RHR trains can become simultaneously inoperable.

The pressure and temperature at the suction of the RHR pump depends on the RHR system lineup and the as-built system configuration. For example, when the RHR system is aligned for shutdown cooling, the RHR pump suction pressure is the same as RCS pressure; during safety injection and containment sump recirculation operations, the pressure at the suction of an RHR pump is equal to the static head pressure created by the refuelling water storage tank and the containment sump, respectively. In all system lineups, the as-built configuration also determines the head loss associated with different system configurations. The range of possible pump suction pressures makes the RHR system susceptible to steam voiding and water hammer during system lineup changes with suction temperatures above certain values.

Since the pressure and corresponding saturation temperature at the suction of an RHR pump depends on RHR system design and as-built configuration, it is important that each licensee ensure that the RHR operating procedures are tailored to their specific systems and include parameters validated as plant-specific to ensure availability of the RHR system in accordance with technical specifications.

## IRS 8216 – SIMULTANEOUS RUPTURE OF OVER PRESSURE RELIEF DEVICES OF THE CALANDRIA DUE TO AN EXPLOSION WITHIN THE MODERATOR COVERGAS

A series of decisions made independently to support maintenance resulted in an undetected build-up of hydrogen within the calandria, resulting in an explosion which ruptured the overpressure relief devices.

In a heavy water reactor, radiolysis of the heavy water results in formation of deuterium and oxygen. Normally, catalytic recombiners in a helium purification circuit minimize the build-up of deuterium, and regular sampling of the helium covergas ensures that deuterium levels do not rise to explosive levels. With the reactor recently shutdown, the recombiners were removed from service to allow for maintenance, resulting in a build-up of stagnant gases. Gadolinium added to the moderator to maintain a safe shutdown state increased the conductivity of the moderator and the rate of radiolysis. The procedure for sampling the covergas did not allow for sampling while the system was depressurized, so no sampling was performed. A maintenance procedure was approved for work which opened up the system, exposing a pocket of deuterium to oxygen from the atmosphere. This resulted in an explosion which was sufficient to rupture the over pressure relief devices of the calandria.

In this event, shutdown procedures did not adequately provide for monitoring deuterium levels, or for taking action to reduce deuterium levels in anticipation of removing the recombiners from service. The maintenance procedure requiring the system to be opened to atmosphere did not adequately analyse the potential impact of exposing the system to atmosphere, or detail precautions to be taken to ensure potentially explosive levels of deuterium were not present. The covergas sampling procedure did not provide a method for sampling when the system was depressurized. These procedures have since been revised to emphasize the importance of maintaining awareness of deuterium concentrations in the covergas at all times and of the potential risks involved from operations which can increase the concentrations of deuterium and oxygen.

### 4.3. Control of Modifications

The number of events caused by deficiencies in the modification process continues to be relatively high. The management of technical modifications should have the same systematic and rigorous review and as the initial design and installation phase of a nuclear facility. In several cases, the problem is observed with the selection of a component type when a previous component type is being replaced.

In one of the events involving degradation of electrical wire insulation, it was found that following a modification a fire protection wall was built in order to separate the area containing electrical and I&C systems from the machinery hall. This modification had significantly reduced the heat removal from the above mentioned area. As a result, the area temperature was remaining higher than the original analysed design temperature. The increase in area temperature caused damage to the insulation of connecting electrical wire.

In another event, three pipe supports of the emergency feed water line were found damaged during a refuelling outage. Investigations revealed that in an earlier design review of all safety related supports the plant operator had noted that the pipe supports were under-sized for their seismic function and hence were modified. The modification was insufficient in that the new supports did not immobilize the piping adequately during an earthquake. The event highlighted the importance of making sure that any modification in an installation does not degrade its safety and that it complies with the requirements for which it has been designed.

In another modification, the handle of an installed air-vent valve was replaced with a heavier one. As a result, the natural vibration frequency (22.0 Hz) of the affected piping changed and was in the range of the vibration frequency (21.8 Hz) generated by operation of the charging pump. Consequently, the affected piping resonated with the vibration generated. The maintenance staff did not conduct vibration evaluation after replacing the valve handle, and therefore did not consider its influence on the natural vibration frequency of the system.

In another event, overhaul maintenance of all pressurizer safety relief valves was conducted and some of their parts were replaced with the new ones manufactured by another company. Investigations done following the event revealed that the replacement parts were inconsistent (in terms of tolerances and material) with the earlier ones. These inconsistencies led to the sticking of the safety relief valve of the primary system pressurizer in open position.

In another event it was found that existing solenoid distributors of fast acting isolation valves were replaced with new ones. The reason for this replacement was the improvement of qualification of valves for conditions after the loss of coolant accident. Following an en-masse replacement, two fast acting valves in unit-1 were found inoperable during post installation testing. Also, approximately a year after the replacement, three containment isolation valves were found inoperable. Following this, the problem was identified as a common cause failure due to the use of fixative screws with a larger diameter head. A second common cause was also identified due to the galling of the spring and the valve guiding peg. The event report highlighted that despite the fact that the manufacturer was aware of the problems and necessary corrective measures, they did not communicate this information to their customers to prevent the potential deficiencies.

During a periodic test, two electrically operated pilot valves failed to close due to jamming of their pilot valves. In addition, a third valve was found to be stuck in the closed position. These affected valves were of a new type and had failed due to prolonged exposure in the operating environment conditions. These valves had shown slightly prolonged opening time during previous periodic tests but these early indications were ignored as the opening time was within the acceptance criteria. Following this event, plant procedures concerning design modifications were improved for better follow-up of long-term behaviour of new components in their actual operating conditions.

It is very important that any changes performed in a nuclear power plant are carried out with careful attention, identifying and analyzing the potential safety impacts as well as controlling and reviewing the affected design, including testing procedures.

IRS 8082 – LACK OF SEISMIC RESISTANCE IN THE SUPPORTS FOR EMERGENCY FEED WATER SYSTEM MOTOR- DRIVEN PUMP SUCTION LINE

The emergency feed water system supplies water to steam generators under certain accident conditions (such as loss of offsite power supply, feed water line break, steam line break, loss of heat sink) so as to evacuate the residual heat from reactor core. Three pipe fixing supports of the emergency feed water line (between the storage tank and the suction of motor driven emergency feed water pumps) were found damaged during a refuelling outage in an even numbered unit [the NPP comprise pairs of twinned units]. Following this event, the plant initiated actions to understand the reasons for damage to the pipe supports In parallel, it was decided to check the design adequacy of these pipe supports.

Investigations revealed that under certain operating conditions, the emergency feed water pumps operate with partial flow rate and this causes pressure fluctuations in the system piping. These hydraulic disturbances generated stress in the system piping which in turn resulted in damage to the pipe supports. These hydraulic disturbances were also encountered in all other even numbered units where the piping lay out was different from those of the odd numbered units (1&3). It was brought out in the report that it is tricky to anticipate stress due to hydraulic disturbances and therefore this was not considered at the design stage.

Investigations also revealed that a design review of all safety related supports had been done a long time before. During this review, the plant operator had noted that the pipe supports in the suction line of the emergency feed water pumps were undersized for their seismic requirements, and hence were modified. During this modification, the existing guide bearings were replaced with the supports fixed to the wall by tie-plate. These new supports included a metal section resting on the upper part of the piping which did not immobilize the piping in the horizontal plane even though the forces are expected to be highest in this plane during an earthquake. From this it was understood a Safe Shutdown Earthquake (SSE) could have caused a loss of integrity of the suction piping of the emergency feed water pumps which would trigger the emptying of the storage tank and its total unavailability. Dealing with such an accident would have involved resort to ‘feed and bleed’ type operation to cool the core.

After the event, corrective actions were taken to modify the supports of similar reactors, taking into account the seismic considerations and the hydraulic disturbances in the line.

The event highlighted the importance of making sure that any modification in an installation does not degrade its safety and that it complies with the requirements for which it has been designed. Also, it shows the need for detailed analysis of operating feedback for issues like hydraulic disturbances (which cannot be accounted for easily during design) to learn all lessons in terms of operation and design.

### IRS 8110 – TWO COMMON-CAUSE FAILURES OF FAST-ACTING ISOLATION VALVES

In the NPP, pneumatically operated fast-acting isolation valves are provided on the pipelines penetrating through the containment building. These valves are operated through solenoid distributors. Each solenoid distributor has two identical controlling electromagnetic valves. One closes the isolation valve and the second opens it.

During unit outages in 2008, the existing solenoid distributors of the fast acting isolation valves were replaced with new ones in both the units. The reason for this replacement was the improved qualification of the valves for conditions after a loss of coolant accident. Following this en-masse replacement, two fast acting valves in unit-1 were found inoperable during post installation testing. It was noted that the head of the fixative screws used for fastening the solenoid distributors had damaged components of the electromagnet valves, rendering the fast acting valves inoperable. These defects were repaired. However, workers performing the job did not come to the conclusion that the screws with the large diameter head supplied by the manufacturer were unsuitable.

Approximately a year after the replacement, three containment isolation valves of unit-2 were found inoperable during routine inspections. The reason for these failures was also due to damage to the electromagnetic valve components from the fixative screws with large diameter heads. Following this, the problem was identified as a common cause failure and necessary inspections /repairs were taken for all related valves during the outages of both units in 2009. The corrective actions include replacement of the existing fixative screws with screws of reduced head diameter. However, early after the outage of unit 1, some of valves were impossible to open during operational manipulations and inspections. This inoperability was caused by the galling of the spring and the guiding peg inside the armature in the solenoid distributor. Systematic checks found the defect of this type in seven out of twenty three isolation valves in unit-1. This was the second common cause failure mode.

Inspection done during the unit-1 outage in 2009 also revealed that the controlling electromagnetic valves were unintentionally interchanged (the closing valve had been changed for the opening one and vice versa) in one of the fast-acting isolation valves. This deficiency was corrected.

The report brought out that the screws with large diameter head were included in the supplies from the manufacturer of the solenoid distributors and were in compliance with the manufacturing documentation. It means that a fault in design occurred at the manufacturer of the solenoid distributors. The manufacturer had supplied fourteen fast-acting isolation valves with the same type of solenoid distributors to other power plants in the country around the same time when the solenoid distributers were supplied for affected plant. The diameter of fixative screws supplied to the other power plant was reduced after the manufacturer realised the effect of the large diameter heads. However, the manufacturer did not send information about the necessity of the adjustment of the fixative screws to the affected plant. The workers performing the installation noticed problems but were not attentive enough to take comprehensive corrective actions.

## IRS 8114 – CRACK ON WELD JOINT IN AIR-VENT PIPING OF CHARGE LINE OF CHEMICAL AND VOLUME CONTROL SYSTEM

During reactor operation, an operator in the main control room observed dripping water in the regenerative heat exchanger room in the containment. The reactor was shut down to identify the leakage point and to conduct detailed inspection and investigation.

Visual inspection and liquid penetrant test revealed leaks from a weld joint in the air-vent piping connected to the charging line upstream of the regenerative heat exchanger. The observed flaws showed an indication mark of about 26 mm length at the center of the weld on the outer surface and an indication mark of about 15 mm length near the boundary between the weld and the base material of the nozzle stub on the inner surface. Observation of the fracture surface with a magnifier identified beach marks and structural patterns which were specific to fatigue cracking. The crack was generated on the inner surface of the pipe and developed toward the outer surface.

Investigation carried out towards establishing the cause for the event revealed that the handle of an air-vent valve installed in the affected piping had been replaced with a heavier one during a periodic inspection outage in 2005. As a result of this, the natural vibration frequency (22.0 Hz) of the affected piping changed and was in the range of the vibration frequency (21.8 Hz) generated by operation of the charging pump at the 100 % flow. Consequently the affected piping resonated with the vibration generated by operation of the charging pump at 100 % flow during the subsequent periodic inspections and high cycle fatigue cracking was generated on the inner surface of the affected piping due to repetitive stress. This crack developed during normal operation of the charging pump and finally ran through the wall to the outer surface.

The maintenance staff did not conduct vibration evaluation after replacing the air-vent valve handle, because they did not fully recognize the importance of vibration evaluation after changing the valve handle and therefore did not consider its influence on the natural vibration frequency of the system.

## IRS 8150 – FAILURE OF SSC (PRIMARY CIRCUITBLOWDOWN/OVERPRESSURE VALVES) DUE TO INADEQUATEQUALIFICATION OF SLIGHTLY MODIFIED REPLACEMENT PARTS

During a periodic test, two electrically operated pilot valves in the primary circuit blowdown / overpressure protection system failed to close due to jamming of their pilot valves. In addition, a third valve was found to be stuck in the closed position. These affected valves were of a new type and had been introduced during the previous refuelling / maintenance outage. These valves had shown slightly prolonged opening times during previous periodic tests but this early indication was ignored as the opening time was within the acceptance criteria.

The new pilot valve had a guide bushing made of a new material (martensitic steel) with a chromium coating. Investigation revealed that the prolonged exposure in the operating environment conditions (temperatures around 250 °C, saturated steam/water environment) resulted in local surface corrosion of the guide bush coating, which caused the jamming of the valve piston in its place.

The plant had replaced only 5 of the 14 electrically operated pilot valves so far and therefore, other valves remained unaffected. Since each blow down valve is additionally equipped with a spring governed pilot valve (in parallel to the electrically operated one), the overpressure protection had not been jeopardized. After the incident, all electrically operated pilot valves of the new type were replaced with ones of the old type, with the exception of two valves of a modified new type. These were kept to gain the operating experience to better facilitate the shift from the old valve type.

Following this event, plant procedures concerning design modifications were improved for better follow-up of long-term behaviour of new safety system components (SSC) in their actual operating conditions.

IRS 8163 – PRESSURIZER MAIN SAFETY RELIEF VALVE FAILED TO CLOSE DURING ROUTINE PRESSURE BUILDUP TESTING

The reactor was under startup after completion of the refuelling outage. The primary circuit pressure was 158 kgf /cm² and the temperature was 270 °C. The routine testing of the pressurizer Pilot Operated Safety Relief Valves (POSRV) was performed by increasing the primary circuit pressure up to 185 kgf /cm². During testing, a pilot valve & its associated safety relief valve opened and primary circuit pressure started dropping. However, the safety relief valve did not close following reduction of primary circuit pressure below 175 kgf /cm² although the interlock on its closure circuit was actuated on-time. Long term release from the pressurizer increased pressure and temperature of the relief tank resulting in rupture of its membrane at 12 kgf /cm². From this moment, primary circuit water started to leak to containment. Following this, the operator terminated the test, switched on ECCS high pressure pumps, and then actuated spray channels, which caused borated water to pour onto the diving bellows and reactor main flange studs. The event resulted in extension of the outage. There was no radioactive release to the NPP site or the environment. The event was rated at level 1 on the International Nuclear Event Scale (INES).

The original pressurizer safety relief valves from the manufacturer were in service since start of the reactor operation in 1986. During the refuelling outage in 2009, overhaul maintenance of all pressurizer safety relief valves was conducted and some of their parts were replaced with parts from another manufacturer. Investigations done following the event revealed that the replaced parts were inconsistent (in terms of tolerances and material) with the original ones and this had resulted in non-closure of the pilot valve due to the slide valve wedging in the open position in the guide bushing. The non-closure of the pilot valve led to sticking of the safety relief valve in the open position.

IRS 8205 – INOPERABILITY OF THE QUICK OPERATIVE ISOLATING VALVES CAUSED BY CABLING DEGRADATION

The report brings out an event wherein damaged insulation on the connecting electrical wire caused a short circuit and loss of voltage to the control board and switch board I&C for Quick Operative Isolating Valves (QOIV) supply. The event resulted in inoperability of four QOIVs on steam lines of the steam generators and two QOIVs on the interim cooling circuit of the reactor coolant pumps. Inoperability of QOIVs did not lead to complete degradation of the safety function as other standby QOIVs (mounted in series) were available.

Following the event, the plant operator initiated actions to establish the reason for degradation in the electrical wire insulation. During this review, it was noted that a fire protection wall was built (as part of a modification) in order to separate the area containing electrical and I&C systems from the machinery hall. This modification had significantly reduced the heat removal capacity from the above mentioned area where a high energy piping system (SGs main steam header and connecting pipes) with temperature 250 °C was also located. As a result, the area temperature was remaining higher (i.e. 50 °C) than the analysed design temperature of 35 °C. Increase in area temperature caused damage to the insulation of the connecting electrical wire.

The event highlighted that the licensee has not considered the impact of modification on aging of electrical and I&C components.

### 4.4. Foreign Material Exclusion

Foreign material remains a significant contributor to events at nuclear generating stations. The potential threat presented by foreign material is twofold. The first is the risk of clogging pipe systems and fuel cooling paths that can result in overheating equipment or fuel assemblies. The second is the risk of jamming rotating machines or control rod movement and result in degradation of reactor protection and control capabilities. In the past significant FME events have resulted in steam generator tube damage, failure of emergency diesel generators, service water system degradation, and control rod failure.

In the recent review period additional risks were identified. In one event FME caused the failure of a valve allowing a backflow of steam to enter into a normally cool section of piping. This led to a guillotine rupture of the pipe, as this pipe was not designed for the pressure and temperature experienced. This rupture also damaged nearby pipe anchors and supports. This event demonstrates that foreign material may represent a significant hazard to personnel and equipment. As the foreign material was introduced during construction, it also highlights the need for a rigorous FME program at all stages of a plants lifecycle.

In another event foreign material was introduced to a load limiter during maintenance. This resulted in an unexpected increase in generator power. Use of proper FME controls and barriers during maintenance, such as the use of an FME plug when the device was opened or performing the maintenance in a clean area, could have prevented this event.

The final event focused on the detection of foreign material using non-destructive testing. In this case, an eddy current inspection of steam generator tubes found an indication of mechanical wear caused by foreign material. Upon review of past maintenance records it was determined that a similar indication had been detected by automated testing equipment several times before during previous inspections. However, the human analyst had rejected these signals and no further investigation was performed. If these analysts been more thorough in their review of the data the foreign material could have been identified and corrected sooner. This event had no consequences but it is an example where a human performance issue allowed foreign material to remain in a system.

IRS 8073 – MANAGEMENT OF STEAM GENERATOR LOOSE PARTS AND AUTOMATED EDDY CURRENT DATA ANALYSIS

During an eddy current inspection of steam generator tubes an automated tester identified a possible distortion. Upon investigation the licensee concluded that mechanical wear between the tube and a foreign object caused the indication. The licensee stabilized and plugged the tube.

This tube had been inspected during prior outages. In reviewing historical data, it was found that an indication had existed at this location for several years. As a result, the licensee concluded that the tube should have been plugged earlier.

Two independent automated data analysis systems were used in the inspections. During these inspections, the primary automated data analysis system identified a distorted signal at the location where the flaw was observed. In previous inspections the human analyst rejected these signals, and therefore no further investigation into the nature of the signal was performed.

The cause of the event was determined to be historic human performance issue related to the technical rigor applied during the review of the distorted eddy current data identified by the automated data analysis system.

As a corrective action, the guidelines for the eddy current data analysis were revised to emphasize the requirements when deciding whether an indication requires additional testing and/or analysis. Lessons learned from this event were incorporated into the site training and testing program.

## IRS 8101 – FOREIGN MATERIAL INTRUSION RESULTING IN POWER CHANGE DURING GENERATOR POWER INCREASING OPERATION

The unit was increasing power when a rapid unexpected increase in the generator power was experienced. The unit was shutdown and an inspection of the load limiter switch and control circuit were performed. It was determined that foreign materials likely intruded in the sliding portion inside the load limiter main body. It is possible that the foreign material was a fragment of a liquid silicone gasket.

The governor equipment is dismounted during every periodic inspection in order to prevent intrusion of foreign material when flushing the oil in the hydraulic pressure system. Its opening is promptly plugged with cloth to prevent intrusion of foreign material through the opening. It is possible that a fragment of the liquid silicone gasket may have been introduced along with the cloth. It is also possible that a fragment of the liquid silicone gasket might have entered during maintenance work as the area was not designated as a clean area.

To prevent future intrusion of foreign material a special plug will be provided during maintenance work. Also, the maintenance work area will be designated as a clean area. The work instruction will be revised to reflect these changes. Additionally some maintenance activities will be performed at the factory instead of onsite.

## IRS 8128 – GUILOTINE RUPTURE OF FIRE WATER LINE TO A STEAM GENERATOR IN REACTOR BUILDING

This report covers an event of guillotine rupture of a fire water line connected to one of the Steam Generators. In this unit, the water to the secondary side of each Steam generator can be fed through the following three paths: the normal feed water line, the auxiliary boiler feedwater direct discharge line, or the fire water. As it is expected that these lines would be used in different operating states, they were designed to tolerate different temperatures and pressures.

A check valve in the feedwater line stuck open which led to backflow of hot, pressurized steam from the steam generator into the feedwater line. The collapse of steam inside the feedwater line introduced severe pressure transients in the system and caused the failure of the check valve in the fire water line. As a result of this, hot and pressurized steam (at about 200 °C and 29 kg/cm2 respectively) entered the fire water system from the steam generator. This line was designed for a temperature of 100 °C and a pressure of 50 kg/cm2. The rigidly supported piping of the fire water system could not accommodate the thermal expansion resulting from the entry of the high temperature steam. This resulted in a guillotine rupture of the pipe near a weld joint in fire water line to the steam generator. This also caused uprooting of a nearby pipe anchor and damage to some of pipe supports

Investigations revealed that presence of foreign material (a welding electrode piece) between the seat and the flapper of the check valve in the feedwater line that prevented it from fully closing during reactor operation. The presence of foreign material inside the feedwater system of the steam generator was attributed to deficiencies in Quality Assurance practices during construction.

As this event occurred at a newly constructed reactor, the presence of foreign material in the check valve indicated deficiencies in the quality assurance program and implementation of the foreign material exclusion policy. Following the event, the utility strengthened the quality assurance practices in all of its Nuclear Power Plants.

The fire water lines from the Steam Generators to the first isolation valve were modified to be the same as the secondary side of the Steam Generators. Thermal sensors were installed upstream of the check valve in fire water lines to detect passing of this valve during reactor operation. These modifications are also being implemented in other NPPs.

### 4.5. Design and Installation

The number of events caused by design and installation problems is still high. Deficiencies in both component and system design can typically present a potential for common cause failure, that may result in simultaneous long term availability of redundant safety systems. The reviewed events indicate that design deficiencies can exist for a long time period before suddenly causing significant problems. Deficiency in the inspection and preventive maintenance is also a contributor to the fact that these installation defects remain undetected.

IRS 8070 - CONSTRUCTION-RELATED EXPERIENCE WITH CABLES, CONNECTORS, AND JUNCTION BOXES

An information notice was issued by the NRC (IN 2010-02) to provide information about the construction related operating experience at domestic and international facilities associated with the rating, installation, and qualification of cables and junction boxes. The notice brings out the following relevant events.

1. An isolation fault was detected in the junction boxes of two containment isolation valves for the nuclear sampling system in a nuclear power plant. Electrical cables in these junction boxes were exposed because of “electrical cable insulation cuts” and “improperly shrunk thermal shrink-fit sheathing’ and has accessed external ground paths. Following this observation, an extensive inspection of cables and junction boxes (pertaining to safety systems) was performed in this and other nuclear plants in the country and similar damage to the safety electrical cables in the junction boxes was noticed. Improper installation practices combined with a degradation of the insulating sheaths over time was suspected as the most likely causes for these damages.

2. A cable fire occurred in a fire-resistant, electrical penetration carrying 6.6 kilovolt electrical cables (to the circulating pumps) and other safety-related cables between the electrical building and the turbine hall. The fire was detected by alarms indicating an insulation fault on the 6.6 kV electrical switchboards, followed by a fire alarm. The primary cause of the fire was attributed to undersized cables for the circulating water pumps. A contributing factor to this event was the closure of the cable penetration at both ends, which allowed a buildup of heat causing carbonization of the cable insulation. Following this event, modifications included (1) using two cables per phase to increase the allowed amperage per phase and (2) by opening one end of the cable penetration to eliminate the oven effect on the cables.

3. In another event, cables of a cable tray sustained damage to their cable jackets from the radiant heat effects of an uninsulated hot pipe that was in proximity to the cable tray. The cables were rated for 90 °C and were close to a steam generator blowdown pipe, which had a surface temperature of 248 °C. As part of the corrective measures, the damaged sections of all cables were replaced as these were not fit for further service and had the potential for cable-to-cable interaction.

IRS 8089 – BREACH OF SAFE OPERATING LIMITS, CAUSED BY SPONTANEOUS OPENING AND FAILURE TO CLOSE OF SG SAFETY VALVE DUE TO FRACTURE OF CONTROL SYSTEM TUBE

The plant was operating at full power. During a walk down, a steam leak was observed from the impulse tube of the Main Safety Valve (MSV) control circuit of a steam generator. Thirty two minutes later the MSV opened and failed to close. Attempts to close the MSV from the control room proved unsuccessful. The Reactor Coolant Pump (RCP) was tripped and the main steam isolation valve was closed from control room. Secondary circuit "rupture" protection and reactor scram actuated. In the course of the event, breach of the following safe operation limits occurred: level drop below 1100 mm of the nominal and steam pressure decrease below 45 kgf /cm² in the SG after the reactor scram.

During a maintenance outage, the diameter of the tube was reduced for sealing the tube connection to the MSV head. This generated stress in the tube connection and it fractured later under the impact of oscillating loads. The root cause for the event was attributed to deficiencies in the manufacturer's maintenance procedures that allow connection by squeezing the tube with greater force, resulting in stress concentration.

IRS 8119 – VALVE ACTUATOR HYDRAULIC OIL LEAK LEADING TO A FIRE AND MANUAL REACTOR SHUTDOWN

Following failure of a hydraulic control unit of a main steam valve, oil escaped and ignited due to falling on hot steam pipe work. The fire lasted for about 5 minutes before auto-extinguishing. As a conservative measure, the plant was shut down and inspections were done to know the extent of damage caused by the fire. During this, it was noted that the fire damaged a number of electrical systems and associated cabling, requiring the need for an extended shutdown to carry out remedial work.

The root cause of the event was inadequate installation of the actuator hydraulic control system, and failure of mounting arrangements due to over-tightening high tensile steel bolts into a soft aluminium plate. A number of corrective actions were identified following this event. These include:

- Revising maintenance instructions for installation of the hydraulic control mounting
- Modifying hydraulic mounting arrangements to an approved design
- Carrying out a design review of the steam and feed valve hydraulic actuation systems

The event highlighted that the plant design and maintenance practices did not reflect the original equipment manufacturer's (OEM) recommendations and this led to failure of a component through over tightening. Changes in material selection has also compromised the original design

A number of required changes to fire barrier / segregation arrangements were done in the light of lessons learnt from this event.

IRS 8210 – EARTHQUAKE INDUCED SUBSIDENCE BETWEEN BUILDINGS

In 1996, as part of the first stage programme of preventive maintenance of civil engineering structures, inspection of the terraced roof of a Nuclear Auxiliary Building (NAB) was carried out. A horizontal displacement of several centimetres was discovered between an NAB roof beam and the supporting wall. In addition, cracks were also observed in a number of beams and walls. Subsequent investigations linked the horizontal displacement of the beam to opening of an inter-building seal resulting from subsidence caused by differences between the foundations of the four structures comprising the NAB. The cracks in the roof were the result of a design in which some of the beams were supported by two different structures affected by differential movements. (This event was reported as IRS 7762 - Fissured roofing due to differential subsidence)

In 2003, the plant operator undertook a study, in light of the above mentioned event, to verify the behaviour of civil engineering structures and their response to a design basis earthquake. The study revealed that the structure and behavior of the buildings concerned are not in question. However, stresses induced due to an earthquake as well as earthquake induced subsidence would likely damage some safety related systems (such as emergency feed water system, component cooling system, essential service water system) which are required to remain functional during design basis earthquake conditions.

When the civil structures were designed, the inter-building differential subsidence was calculated and taken into account at the junctions between the buildings. No account was taken of potential differential subsidence within a single given building as each building is assumed to be monolithic. The NAB was, however, a special case as it was built on a monolithic foundation slab, but divided into four equally sized superstructures (A, B, C & D) by a seal in the form of a cross. The differential subsidence across NAB cannot be determined at the design stage. The actual settlement of such buildings at the end of their life is estimated by a simulation on the basis of historic subsidence records, and taking the specific characteristics of the building into account. The event brought out the need for re-evaluation of the forces exerted on the piping and its supporting structures, taking into account the differential subsidence at the end-of-life of the structures and the earthquake induced subsidence.

The incident did not any actual impact on the operation or safety of the plant. However, the potential impact could have been detrimental to safety, depending on the amplitude of a postulated earthquake.

## 4.6. Criticality Reactivity Events

There were three significant reports about criticality events, one of which is generic. The events were related to degradation of systems required to control reactivity, deficiencies in operation (including maintenance and surveillance), deficiencies in safety management / quality assurance system, and deficiencies in safety evaluation. These events occurred at all types of reactors in all modes, including at power, and during fuel handling, and low-level power operations, and when testing or maintenance was being performed. These events were considered as events of potential safety significance. During one of the events after the reactor became subcritical through xenon build-up and reactor coolant temperature increased, operators delayed inserting control rods for nearly 2 hours. One event was rated as INES level 2.

IRS 8104 – NON CONFORMANCE WITH CRITICALITY CONTROL ARRANGEMENTS THROUGH INAPPROPRIATE OPERATIONAL DECISION MAKING

Fuel load was carried out at a gas-cooled reactor using a combined fuel assembly made up of a plug unit and a stack of fuel elements. When a bump latch was carried out to join a new fuel stringer to a plug unit, it was noted through a viewing window that there was foreign material caught in the coupling. The fuel movement was stopped, leaving the fuel stringer suspended from the plug unit approximately 2.5 m above its original position. As the integrity of the coupling was not known, the decision was taken to inject expanding polyurethane foam into the new fuel carrier to act as a shock absorber if the coupling should fail and the fuel drop. Because the foam had the potential to act as a moderator, this was in breach of local arrangements for the prevention of accidental criticality. In addition, the foam was a flammable material. Further event recovery action led to an unapproved modification to the plant to fit a temporary clamp to restrain the fuel to prevent it from falling.

The root causes were attributed to inadequate knowledge on the part of the individuals involved in the event of potential criticality relating to ex-reactor activities. The potential for criticality from injecting the foam was not recognised. There was a failure to set up an effective event recovery organisation to deal with the suspended fuel stringer.

The key corrective actions focused on the following areas: closing the criticality knowledge gap through development / delivery of initial and refresher training; addressing decision making shortfalls at both management team and worker level; reviewing arrangements for visibility of criticality control arrangements at the point of work.

The event was rated as INES level 2.

## IRS 8157 – OPERATOR PERFORMANCE ISSUES INVOLVING REACTIVITY MANAGEMENT AT NPPS

An information notice has been issued by the NRC (IN 2011-02) to provide information about events involving deficiencies with reactivity management planning and implementation at several nuclear power stations.

A subsequent review of one plant shutdown found that control room operators did not effectively control reactivity to maintain the reactor in the desired condition during low-power operations by properly anticipating, controlling, and responding to changing plant parameters. Operators did not use control rods or boron concentration - two means that operators can use to directly control the amount and timing of reactivity changes - to adjust for reactivity changes resulting from xenon build-up and reactor coolant temperature changes.

During one of the events, after the reactor became subcritical through xenon build-up and a reactor coolant temperature increase, operators delayed inserting control rods for nearly 2 hours. NRC IN 1992-39 discusses an event in which, after the operators brought the reactor subcritical by inserting control rods, an inadvertent unplanned return to criticality occurred because operators delayed actions to continue inserting control rods while changing shifts.

Plants and utilities may consider revising procedures and training operators so that, after the reactor becomes subcritical, the operators will proceed immediately to insert control rods or add boron to ensure the reactor remains shut down.

## IRS 8215 – UNEXPECTED DECREASE OF THE MODERATOR GADOLINIUM CONCENTRATION DURING MODERATOR REFILL

During moderator refill at a pressurized heavy water reactor, a decreasing trend of the Moderator gadolinium concentration was observed through routine sampling. This decrease in concentration was both unexpected and unexplained. The licensee took the conservative decision to stop the Moderator refill and return the reactor to the safe drained Guaranteed Shutdown State. Throughout this event, the reactor was always subcritical by a significant margin; however, the event did result in the normal margin being reduced.

The event related to degradation of systems required to control reactivity and had a negligible effect on public safety. The direct cause was the introduction of oil into the Moderator circuit. The root cause was the failure to identify the fact that non-gaseous hydrocarbons could generate oxalate anions in the Moderator system. The contributing cause was determined to be an organizational insensitivity to contaminants in systems, structures and components, which affected equipment reliability. There have been three previous gadolinium oxalate events.

### 4.7. Procurement Technical Specifications and Spare Parts

This is a new category that was not included in previous biennial highlight reports. During the current biennial period, several events related to the procurement process and the use of spare parts were reported. The significance of these reports was sufficient to merit that this category be included to share the important lessons learned.

The events are related to deficiencies in purchasing specifications, showing the need for improvement of QA programs during the procurement phase, the need for an enhanced review of the quality and configuration control of data packages, and acceptance inspections of spare parts.

A generic analysis of events from recent operating experience alerted licensees to observations and findings in the area of commercial grade dedication (CGD). These observations and findings show shortcomings in the engineering justification of changes in critical characteristics of commercial dedicated parts, and deficiencies in the traceability of the sampling plan during the dedication process. The generic study of the event provides recommendations on how sampling is used effectively in the commercial dedication process.

Some of the events involve deficiencies that remained unnoticed as a latent problem for some time. In some cases, given the similarities of equipment between other NPPs of the same series, the deviations needed to be promptly analyzed, taking into account the extension of cause determination to identify the potential impacts in other installations of the same or similar design.

If an issue is identified with the quality of parts supplied by a subcontractor, it may be difficult to fully address the issue.  Items from previous orders from the same contractor should be inspected. However, it can be difficult to follow the procurement chain through a network of contractors and subcontractors. Material from the deficient subcontractor also may be introduced to plants indirectly through other suppliers that commission the subcontractor.

In some cases, the required non-destructive examinations performed prior to on-site installation were not designed to reveal all the potential important problems. This show the need to enhance attention during fabrication by using adequate inspection methods and appropriate procedures applied by qualified people.

IRS 8091 – DEFICIENCIES IN PURCHASING SPECIFICATIONS FOR THE REPLACEMENT OF EQUIPMENTS

During the replacement campaign of safety electrical supply boards, several problems or incidents happened in the plant linked mainly to the procurement phase, including deficiencies or absence of requirements in the purchasing technical specifications, and deficiencies in the factory and site acceptance testing.

One of the problems related to the overheating of a safety air compressor motor due to the replacement of an isolation device by another type with different characteristics. The maintenance worker noticed a strong smell of burning coming from the motor. It was concluded that the cause was that the replacement breakers and contactors were another type which were not designed for a high number of operations. The equipment specification did not mention explicitly the necessary requirements to make a correct choice for the new isolation device.

In another case, after a long investigation it was discovered that as result of the replacement of the supply boards, the rotating fields of the EDG were reversed. This deficiency remained unnoticed as a latent problem for six months.

Improvement of QA programs during the procurement phase by the establishment of an independent and systematic review of the quality and configuration control of data package and test results from the factory and onsite might be an important contributor in reducing the number of deficiencies, and efforts in this area should be further strengthened.

The above effort could be further enhanced by appointing a suitably qualified and experienced engineer (or independent reviewer) to perform a systematic review of all equipment design specifications and ensure the completeness of the functionalities checked during factory and on site commissioning tests.

IRS 8093 – CONFORMITY DEVIATION AFFECTING QUICK BLEED VALVE MEMBRANES IN THE STEAM ISOLATION VALVE CONTROL

The plant operator noted that the membranes of the quick bleed valves located in the quick closing control line for steam isolation valves (MSIV), replaced four months previously were not working as expected due to a premature degradation. The membranes affected were in a grade of elastomer that did not comply with the design drawings and were likely to degrade at a lower temperature. The procured material should have been EPDM, which withstands temperatures in the order of 120 ºC.

Given the similarities of this equipment among other NPPs of the same series, the plant operator, in conjunction with the common supplier, checked the characteristics of membranes installed in other plants of his fleet, and found that none the membranes delivered by the supplier in the last eighteen months complied with the design requirements. In one of the plants, during a test, four membranes stuck to their seat and prevented the normal operation of the MSIV control valve within the required closing time.

The cause analysis revealed that there were no manufacturing inspections carried out on the premises of the supplier or his subcontractors, because there was not a specific requirement to do so by the plant operator. The replacement membranes degraded at temperatures higher than 80 ºC, whereas the material required should have been capable of withstanding temperatures up to 120 ºC.

The incident illustrates the need to take into account all requirements resulting from normal and accident situations and to sustain them with properly procured and manufactured spared parts. As soon as deviations are noted, they must be promptly analyzed taking into account the extension of cause determination to identify the potential impacts in other installations. From now on, the plant operator will procure spare parts that have a potential impact on safety and availability through a new spare parts category covered by stipulations for the supplier and quality monitoring of manufacture.

IRS 8124 – MANUAL SHUTDOWN AND FORCED OUTAGE DUE TO FAILURES OF HIGH PRESSURE EMERGENCY CORE COOLING (HPECC) GAS ISOLATION VALVES

While performing logic test on High Pressure Emergency Cooling (HPECC) gas isolation valves, the valves failed to open, causing the unavailability of the system. Considering that the valves' function could not be recovered within the 8 hour Technical Specification requirements, a manual shutdown was carried out for a forced outage.

The inspection and analysis subsequently revealed some discrepancies in the size and shape of the curve of the valve seat when compared with the original valve seats. The deformation of the valve seats increased under high system pressure, causing enhanced friction between sealing rings and other parts of the valve, leading to significantly increased operating torque. As a result the valve could not open.

When investigating the event, it was found that some months before the valves had been disassembled for preventive maintenance, and on this occasion new spare valve seats were installed. Although the new valve seats were supplied by the original manufacturer, the spare valve seats did not meet the technical specifications. The reasons why some discrepancies existed between different batches of spare parts are pending further explanations.

The event also shows that the acceptance and inspection of the spare valve seats were inadequate. To reinforce the procurement process, and ensure adequate quality of the spare parts, the plant has developed acceptance procedures, including guiding steps to verify the material, size and function of spare valve seats to ensure quality and reliability.

IRS 8133 – DEFICIENCIES IN THE QUALIFICATION OF A MANUFACTURER, IN QUALITY ASSURANCE AND CONTROLS DURING MANUFACTURING OF A SAFETY RELATED VALVE

As part of a generic project related to the study of steam generator tube rupture accidents, it was decided to motorize the isolation valves of the main steam (MS) relief valves to the atmosphere in order to minimize their closure time and potential radiological release. The licensee decided to replace both the actuator and the valve. Important welding problems occurred on one of these new valves. Further analysis showed indications of flaws and metallurgical problems on the other ones. Previous pre-service tests and inspections by the manufacturer and by the authorized inspection agency had not reported these faults.

Further in-depth examination revealed qualification deficiencies by the manufacturer and quality assurance deficiencies in the manufacturing and inspections of the valves.  Shortcomings in the casting quality system of the foundry subcontractor did occur, due in part to poor supervision. The whole chain of controls, including non-destructive examination and tests, were unable to identify the metallurgical problems that occurred. Only one inspector monitored the performance of all manufacturing stages at the foundry, which resulted in insufficient adherence to the ASME code. The quality monitoring was defective and failed to identify deficiencies.

The following lessons should be highlighted:

The required non-destructive examinations (according to the ASME code) performed prior to on-site installation are not necessarily able or designed to reveal all the potential important problems. Some of them have to be captured during the fabrication by adequate methods and through appropriate procedures applied by qualified people.

Licensees have to perform adequate oversight of contractors (including sub-contractors). This oversight should be done according to applicable design codes and QA manuals, and should be graded according criteria such as safety significance, staffing, and experience of the (sub-) contractor.

Moreover, other orders were made to the subcontractor during the previous years, and similar problems could arise in the ordered elements. Likewise, material from the same subcontractor may indirectly be in the power plants of the licensee through other suppliers that would commission the same subcontractor.

Manufacturing and ordering procedures have been modified as a result of this event. Also, it has been necessary to identify the materials potentially impacted. A list of pending orders to the subcontractor has been established to assess the possible impact on the basis of the feedback. With the aid of the licensee’s suppliers that uses the same subcontractor, a list has been established incorporating other elements that might be likely to encounter the same problems.

IRS 8156 – COMMERCIAL-GRADE DEDICATION ISSUES IDENTIFIED DURING NRC INSPECTIONS

An information notice has been issued by the NRC (IN 2011-01) to provide information about recent operating experience and alert licensees to observations and findings in the area of commercial grade dedication (CGD). CGD is the acceptance process undertaken to provide reasonable assurance that a commercial grade item, when used as a basic component, will perform its intended safety function, and in this respect, is deemed equivalent to an item designed and manufactured under a quality assurance (QA) program.

The following summarizes the finding areas: lack of engineering justification during the CGD process, documentation, vendor audits versus commercial grade surveys and sampling plans.

In several instances it was found that that the vendor made design and engineering changes during the dedication process without an engineering justification to validate these changes. In one instance, the previous table of critical characteristics identified some items as having a safety related function, but the current version identified the part as non-safety related. The vendor failed to provide engineering justification for the downgrade.

In another instance, it was noted that all components for an item were procured as commercial grade. However, none of the commercial components had any documented technical evaluation or acceptance method bases.

During one vendor inspection it was found that the vendor survey failed to verify that the sub vendor's quality controls included specific processes, such as material traceability and lot or batch controls, relevant to the commercial components to support the sampling plan during the dedication process.

Cases were found where vendor procedures did not provide adequate guidance for the development of sampling plans, or adequate sampling criteria to include qualitative factors such as the safety significance of the item, complexity of the item, or performance history.

Sampling of items for dedication can be controlled by establishing traceability of metallic material or establishing lot/batch controls of the components. When neither can be established, documented sampling plans can be established on an individual item-specific basis. Guidance on how sampling is used in commercial dedication process is described in NRC Inspection Procedures IP 38703 and 38704 which refer to Programs and Inspection of Commercial Grade Dedication.

### 4.8. Ineffective use of OE, Recurring Events

Effective use of operating experience can provide a valuable tool, allowing lessons from past mistakes to prevent future recurrence. Repetition of mistakes demonstrates not just weakness in the affected program, but an organizational weakness in the areas of root cause analysis and the corrective action program. An organization truly dedicated to safety takes advantage of any information available to prevent the occurrence of potentially significant issues which can impact safety. A number of events reviewed from the IRS from the last two years demonstrate the impact that an ineffective use of operating experience can have on plant safety.

IRS 8074 – INADVERTENT CONTROL ROD WITHDRAWAL EVENT WHILE SHUTDOWN

Incomplete application of operating experience (OE) resulted in three rods drifting out of the core while shutdown due to valve manipulations by non-licensed operators. The mechanism for the rod drift was identical to that described in an industry OE report reviewed by the plant just the year before.

In 2007, an industry OE report notified utilities of the potential for rod drift in boiling water reactors while shutdown. This situation, experienced in other plants (IRS 7834) could occur if a sufficient number of hydraulic control units were isolated while the plant was shutdown. After reviewing the OE report, the licensee recognized that they were vulnerable to the same event, and instituted a keyword search to identify control rod procedures where this situation could be applicable. At the time the event occurred though, operators were using a reactor protection system procedure. The earlier search to revise relevant procedures had not looked at procedures for other systems. Rod drift alarms were received in the control room indicating that three rods were out of position, as accurately shown by the rod position indication. However, it was initially assumed that the indications were due to on-going testing of instrumentation. It was 20 minutes before it was recognized that the control rods had actually moved, and over an hour and a half before they were fully re-inserted. After the fact, one of the operators recalled training that had mentioned the industry OE report, but none of the control room operators monitoring indications at the time recognized the situation.

Nothing in the procedure used by the non-licensed operators manipulating the valves directed specific core conditions or a maximum time between operating each of the two valves associated with each control rod. Though in this case the core remained fully shutdown, had the moderator temperature been lower, the time from shutdown had been longer, or the time to cycle the valves been longer, this event could have resulted in an inadvertent criticality. The utility was aware of the potential for this event to occur, but had taken inadequate steps to ensure that applicable procedures were revised to prevent it. Training on the OE and on the procedure revisions was not sufficiently effective for operators in the control room to immediately understand what had happened. All plant procedures have now been reviewed and precautions have been noted in procedures directing manipulation of hydraulic control unit isolation valves that these valves have the potential to affect plant reactivity.

## IRS 8083 – NON-COMPLIANT LUBRICATION MIXTURE IN QUALIFIED ACTUATORS

During regularly schedule preventive maintenance, operators discovered a non-compliant mixture of lubricating grease in electric actuators. It was presumed that lubricating grease intended for nearby yoke nuts had been applied instead to the electric actuators.

Though the lubricating grease for the electric actuators and the yoke nuts was of a different colour, the lubricators for applying grease to the electric actuators and the yoke nuts were identical and in close proximity to each other. Though the lubricating greases were independently qualified for use on their respective components, the unintended mixture of the greases in the electric actuators presented the potential for common mode failure of the safety-related actuators as the mixture had not been verified to be compliant.

This has been an issue discussed in previous IRS reports (IRS 1260 and IRS 7320), but the recurrence of the issue demonstrates that corrective actions to more visibly differentiate the two separate systems, and to increase operator awareness of both the potential for mixing greases and the possible consequences of the error had not been effective in preventing recurrence.

## IRS 8131 – TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP REPETITIVE FAILURES

This report describes the events from IRS 8079 (discussed in the section on events with importance) along with similar events from two other plants where repetitive failures of the turbine-driven auxiliary feedwater (TDAFW) pump were not appropriately addressed by the plants. This resulted in multiple failures of this safety significant equipment over the course of two years.

At one plant, corrosion of the turbine governor control valve stem had been noted in 2005. At the time, it was also noted that the steam admission valves were leaking. The valve stem was replaced, but actions taken to address the leaking steam were not effective. In 2008 and twice in 2009, the pump tripped on overspeed during routine surveillance testing. Though plant personnel found and corrected deficiencies in the maintenance program that might have degraded pump performance, despite their previous experience with the corroded valve stem, they did not discover that as the cause until after the third pump trip in eight months. At another plant, the pump tripped during a surveillance test because a reset latch had been inadvertently bumped sometime prior to the test, causing the pump to trip on overspeed. The vulnerability of the latch to being bumped by personnel transiting the area had been noted at least five times over the course of seven years, but no actions had been taken despite the impact this could have on pump operation. Further problems with the pump demonstrated a reluctance on the part of the plant to incorporate well-established OE for the TDAFW system into their maintenance program.

The TDAFW system is a complex system that can be made inoperable by any number of different failures. Because of its safety significance though, it is important to make use of all available operating experience, both from within the plant and from the rest of the industry, from TDAFW systems and also from related turbine driven injection systems. In the event of failure of the TDAFW system, operating experience can be used to ensure that the root cause analysis truly identifies the cause of the failure. Corrective actions that address not only the failed component, but also the inadequate processes that allowed the component to fail are more likely to be effective in preventing recurrence.

IRS 8134 – FUEL ASSEMBLIES DAMAGED DURING REFUELLING OPERATIONS DUE TO MISALIGNMENT

Unable to couple a control rod drive mechanism to its associated control rod assembly during vessel reassembly following a refuelling outage, the utility removed the vessel head and discovered that an inadequate procedure had allowed several fuel assemblies to be damaged during the vessel reassembly, which was interfering with the control rod coupling.

Guidance from the fuel vendor, informed by past industry events (see IRS 8041 and IRS 1166), specified a maximum allowed gap value between a fuel assembly and the channel wall. The procedure used to verify that the vessel was ready for reassembly required only a qualitative examination of gaps, specifying a maximum number of gaps allowed in a given area, but not the size of the gaps. The fuel loaded in the reactor was in accordance with the plant procedure, but the gap between the fuel assembly and the channel wall for the most severely damaged assembly was larger than the maximum allowable gap specified by the fuel vendor. Because of this gap, the fuel assembly upper end fitting and reactor vessel head did not align properly, resulting in the weight of the plenum resting on the upper end fitting tabs of the assembly and causing severe deformation. Absent the problem with the control rod coupling, it is possible that the plant would have completed an operating cycle with a deformed fuel assembly with questionable cladding integrity. The deformed fuel assembly would also have been more likely to stick to the reactor vessel head, posing a risk for a dropped fuel assembly and potential for fission product release.

When reassembling the reactor vessel, operators appropriately followed plant procedures, but still place the reactor in a compromised position. Plant procedures did not adequately incorporate available operating experience and guidance from the fuel vendor, leading to a situation which could have been avoided. Plant procedures have since been updated.

### 4.9. Radiological protection, Exposure, Contamination

Review of IRS reports from the last two years showed a significant increase in the number of events involving contamination and personnel exposure. This increase was sufficient that it was judged appropriate to add a category concerning radiological protection in order to emphasize the importance of the events. Many of the events discussed below resulted from improper coordination of maintenance activities, a theme that was also noted in the section on procedures. In several instances, plant operators failed to recognize that activities could have a significant impact on exposure and contamination levels. More disturbing though, was a demonstration in many of the incidents of a lack of appreciation for the hazards posed by radiation and airborne activity at all levels of the plant organization. There are examples of workers not understanding the importance of protective equipment, of radiation protection personnel failing to take action in situations where exposure was likely, and of managers making non-conservative decisions with respect to radiation protection. Radiological safety should be a top priority of every radiation worker, but the volume of events from around the world related to exposure and contamination indicates that this message is in danger of being forgotten.

### IRS 8160 – EXPOSURE OF TWO WORKERS IN EXCESS OF STATUTORY ANNUAL DOSE LIMITS

Inadequate coordination of maintenance activities resulted in exposure of two workers above the statutory annual limit. Deficiencies in the plant programming of electronic personal dose (EPD) alarms could have resulted in even higher levels of exposure if circumstances had been slightly different.

During a maintenance outage, an electrician, accompanied by radiological protection staff, was installing lamps in the area directly under the reactor vessel. The background exposure levels in the room were 1.5 mSv per hour, above the EPD dose rate alarm set point of 1 mSv/hr. Unknown to either of the workers, because the dose rate alarm was already alarming, it was suppressing the total dose alarm which could alert them if their actual total dose went over the pre-set limit of 5 mSv.

While under the reactor vessel, a separate maintenance activity requiring the withdrawal of activated incore guide tubes from the reactor core resulted in a significant increase in the area dose rate to over 1000 mSv/hr. The two personnel were only alerted to the change by a radiation protection (RP) coordinator who heard about the planned guide tube withdrawal, and seeing that necessary precautions had not been taken to prevent personnel access to potential high radiation areas, found the two workers under the vessel and warned them to get out. Had the coordinator not heard that the guide tube withdrawal was occurring or not understood the radiological impact and necessary precautions, the workers could have been exposed to much higher levels of radiation because their EPD did not alarm at 5 mSv as they expected it would.

Two deficiencies resulted in two personnel being exposed in excess of statutory annual dose limits, and could easily have resulted in much higher levels of exposure with more significant consequences. Precautions were not taken to ensure all personnel were clear of areas that would be exposed to high radiation during withdrawal of the incore guide tube, and to ensure personnel could not access those areas. In addition, the lack of knowledge on the part of the workers as to the operation of the EPD alarms meant that it was only the fortuitous intervention of the RP coordinator that prevented even higher exposure levels. The plant has changed the programming of the EPD alarms, improved coordination of maintenance activities, and installed protective interlocks to prevent withdrawing the incore guide tube when the reactor base room door is open.

### IRS 8165 – AIRBORNE ALPHA RADIATION EMITTING CONTAMINATION HAZARD RESULTS IN WORKER UPTAKES DURING REFURBISHMENT ACTIVITIES

Plant programs assumed that actions taken to detect and protect against beta contamination would be sufficient to also protect against any alpha contamination. When airborne alpha-emitting particulates were detected, further analysis determined that several hundred workers may have received an uptake of alpha contamination, and that alpha-emitting airborne particulates may have been present, undetected, for several years.

As part of plant restart activities from a multi-year extended shutdown, the radiation protection group, using experience based solely on the restart of the another plant on the site, made non-conservative assumptions about the presence of alpha contamination relative to beta/gamma contamination. As a result, criteria for airborne particulate surveys were insufficient and work permits were issued without requirements for appropriate breathing protection. Once indications were received that alpha contamination might be present, plant oversight of maintenance activities failed to halt work or provide further protection for workers for over a month until further confirmatory evidence was received.

Poor oversight of the radiation protection group and a failure to apply operating experience from beyond the site led to non-conservative assumptions about the potential for uptake of alpha contamination. No measures were taken to provide for early detection of alpha contamination, and no measures were taken to protect against potential uptake by workers. Dose assessment determined that no workers were exposed beyond regulatory limits. Immediate actions were taken at all plants around the country to provide for improved detection and mitigation of alpha radiation hazards.

IRS 8167 – ELEVATED LEVELS OF AIRBORNE ACTIVITY IN CONTAINMENT BUILDING

Ineffective control of air being used to aid in steam generator tube inspection led to airborne activity in the containment and potential uptake by over a hundred workers. Elevated airborne activity levels had been detected by radiation protection personnel over the course of three days, but they did not have the preparation or training to deal with an abnormal airborne activity event.

RP personnel detected elevated airborne activity levels over the course of two days, but the results were not reviewed and no actions were taken until workers started to show positive whole body count results. Examination after the fact determined that air being used to push a probe through steam generator tubes for eddy current testing was escaping into the containment, carrying contamination from the primary circuit. One train of the containment ventilation system was taken out of service for maintenance while the steam generator testing was in progress without consideration for the potential effects on containment activity levels. During earlier testing, this had not been an issue because seals around the edges of the nozzle dams had been made water tight during previous outages, preventing the air being used from escaping into containment. Because the primary circuit had been drained below the nozzle level prior to this round of testing, the seals were not made tight, and were only installed for purposes of foreign material exclusion.

The decision to remove a train of containment ventilation from service at a time when the primary circuit was open to the containment atmosphere did not demonstrate an understanding of the potential risk of the steam generator testing evolution. Review of the event found significant weaknesses in the management and oversight of the RP program, allowing known elevated airborne activity levels to remain unaddressed for almost three days, in part because RP personnel had no guidance on how to proceed in the event of elevated airborne activity readings. An inaccuracy in the whole body count computer printouts masked the level of uptake by workers in containment, delaying appropriate response. The plant has instituted procedural updates and improved training for RP personnel for addressing airborne activity events, and increased the emphasis on ensuring proper ventilation in containment when the primary circuit is open.

## IRS 8170 – TRITIUM INTAKE BY WORKER DURING D2O SYSTEM MODIFICATION

A failure to follow procedures regarding the use of personal protective equipment and a lack of understanding of the hazards of tritium resulted in tritium uptake by a welder with a total dose of 14.5 mSv.

To install a line in a D2O supply system, machinists wearing full protective equipment cut a D2O supply line. The intent was to drain the line, collecting the D2O inside for processing, then to weld the new line into place. The radiation protection (RP) support personnel had left the job site before the work started, but the decision was made to start cutting the pipe anyway. Some D2O spilled to the floor with the initial cut. After collecting 11 litres of D2O from the pipe, a machinist wearing only a respirator, but no hood or protective clothing, started welding the new pipe in place. D2O was still dripping from the opening of the pipe at a rate of about a drop every three seconds when the welding started. The RP technician returned and found that the cloth on the floor had collected some of the D2O seeping from the pipe. The welder was warned to stop work, but the welder insisted on finishing. When the welder attempted to leave the Radiation Control Area, contamination was detected. It was determined that the welder had skin contamination and had also received an uptake from inhalation of the D2O vapour.

This event showed a lack of understanding on the part of the machinists of the danger posed by tritium and the uptake risk posed by even small amounts of D2O. The welders failed to follow procedures requiring the presence of an RP technician before commencing work, and procedures requiring that any spill of D2O be immediately reported and cleaned before further work took place. The RP technician did not follow procedures for ensuring the use of protective equipment and did not enforce the need for the welders to leave the area when the exposed D2O was discovered. Corrective actions emphasized the need for RP presence and a clear understanding of the hazards posed before any work is performed that poses a high risk for contamination.

## IRS 8183 – OVEREXPOSURE OF WORKERS BEYOND ANNUAL REGULATORY LIMIT

Inadequate review of procedure requirements and a failure to adhere to standard work practices resulted in four workers receiving a dose in excess of the annual regulatory limit, with the potential that they could have received a much higher dose.

A work permit was issued and approved in the control room for painting in the Spent Fuel Transfer Duct (SFTD). As part of the work package, control room staff were instructed to ensure that no fuel was transferred while personnel were in the SFTD because this would result in high exposure levels in the SFTD. However, no further precautions were taken to prevent spent fuel transfer. About two hours later, an operator who had not been in the control room when the work package for painting was issued, initiated a fuel transfer. The personnel in the SFTD heard an unusual noise and immediately left the area. Review by health physics personnel determined that four of the seven workers had been exposed above the annual regulatory limit, and that exposure could have been even higher if they had not left the room immediately or if the fuel had become stuck during the transfer.

A failure to take precautions to physically prevent fuel transfer while personnel were working in the SFTD resulted in four personnel exceeding their annual statutory limit and could potentially have had more catastrophic consequences.

## 5. CONCLUSIONS

The consultants reviewed the 166 event reports submitted to the IAEA in the period 2010 – 2011. About half of them were selected for more detailed review because they had important learning points. Five events and a brief summary of the events from [redacted] were discussed separately under the paragraph of "events with important lessons". The rest of the events were grouped into two main categories of "equipment" and "management of safety".

This categorisation however is rather misleading because only a very small number of events can be characterised by a single equipment failure or simple human error. Even though the events were discussed according to the categories they were divided in, there is significant overlap among these categories. There are some common issues which could have been a basis for different categorisations. Having reviewed the set of events with important lessons and the events related to equipment failure and management issues the following general conclusions were drawn:

COMMENTS

In the conclusions of previous biennial highlight reports it has been highlighted that IRS content is substantially improving: better quality of entries, more and better level of detail, lessons learned substantiated and explained, and corrective actions progressively introduced. Furthermore the quality and quantity of attachments included with the reports has also increased. This greatly aids the understanding of event reports. Overall, the IRS data base is growing positively and is becoming an important knowledge management database, based on actual experience. The scope of the IRS has expanded to include operating experience from construction, commissioning and decommissioning. These conclusions continue to be applicable and showing steady progress in a good direction.

- One report selected as an important lesson describes recurrent events attributed to inadequate root cause analysis and delay in the implementation of lessons learned from one event applicable to other units with similar design of systems and equipment involved. It is important to not only address the direct cause of events, but also to look at wider programmatic issues to prevent recurrence of similar events.

- Several events highlighted the need to ensure coordination of maintenance activities. This is particularly essential before approving the start of work to ensure that the impact of current conditions and concurrent activities is fully understood and analysed.

- It is imperative that plant personnel at all levels of the organization maintain an understanding that even routine activities on non-safety related systems can impact reactor and radiation safety.

- The significant increase in the number of events involving personnel exposure and contamination shows a disturbing trend in the area of radiation protection. The importance of radiation safety as a top priority should be emphasised at all levels of the organization.

- Common cause failures remain an important issue resulting in actual or potential failure of redundant components of safety systems. There is a need to identify vulnerabilities and enhance awareness on common cause issues based on lessons learnt from operating experience. This also highlights the need for effective and timely sharing of lessons learned from the common cause failures.

- The number of events caused by deficiencies in the modification process continues to be relatively high. Many of the events were caused by shortcomings in the procurement process, and deficiencies in the replacement of spare parts. Some of the events involved deficiencies that remained unnoticed as a latent problem for some time. Because of this, a new section on Procurement and Spare parts has been added to this edition of IRS Highlights. This is an area that may merit increased attention.

- Licensees and regulatory organizations of several member states are issuing more generic reports with comprehensive information assessment, giving perspective to the evolution of trends, as well as advice for corrective actions programs. This is a noteworthy good practice encompassing substantial added value that was already highlighted in the conclusion of the previous Highlights report. We positively encourage the submission of such generic reports from all member states, as this is an important source of OE.

- The nuclear industry regularly performs drills and exercises to assess their emergency response capability. Licensees are encouraged to report valuable lessons discovered during emergency preparedness and response exercises to assist others who may benefit.

- There has been a significant rise in the number of events associated with underground power cable failures that may affect accident mitigation systems or cause plant transients. Licensees should ensure that their monitoring and testing programmes are robust and effective so that underground safety related cables that are exposed to continuous submerged environments are identified and evaluated.

- The number of events involving the ageing of containment, submerged cables and piping has increased since the last period. The deficiencies were sometimes associated with the original design or construction and identified only recently because they are located in inaccessible areas. To cope with such ageing problems, an aging management programme with proper procedures and practices is needed that can identify deteriorated components prior to failure. It is recommended that in-service inspection programmes and ageing management programmes are reviewed for SCCs in inaccessible areas to include additional measures to identify indications of corrosion that may be visible and include the appropriate follow-up actions.

# ANNEX I. LIST OF SELECTED IRS REPORTS FOR EVENTS IN 2010 – 2011

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 01-03-06 | INOPERABILITY OF 22 OUT OF 61 CONTROL RODS OF THE REACTOR PROTECTION SYSTEM | M | 15-04-11 | 7773 | [redacted] | [redacted] | | |
| 09-04-07 | REACTOR UNIT SHUTDOWN DUE TO A FAILURE ON A 6.6 KV EMERGENCY SWITCHBOARD | F | 20-07-11 | 7871 | [redacted] | [redacted] | | |
| 04-09-07 | PARTIALLY FAILED SCRAM | M | 02-02-10 | 7920 | [redacted] | [redacted] | | |
| 08-01-05 | FRAZIL ICE IN COOLING WATER CHANNELS LEADING TO REACTOR SCRAM | M | 02-02-10 | 7921 | [redacted] | [redacted] | | |
| 04-05-09 | DROP OUT OF A SELF POWERED NEUTRON DETECTOR FROM TRANSPORT CONTAINER IN THE REACTOR HALL DURING REFUELLING OUTAGE | M | 28-11-11 | 8006 | [redacted] | [redacted] | Management Issues | Procedures |
| 16-04-08 | ERRORS IN TURBINE DISINTEGRATION AND FIRE SAFETY ANALYSIS | M | 15-02-10 | 8067 | [redacted] | [redacted] | | |
| 01-12-09 | TOTAL LOSS OF THE HEAT SINK FURTHER TO CLOGGING OF THE ROTATING DRUM SCREENS BY A VERY SIGNIFICANT ARRIVAL OF VEGETABLE MATERIALS | P | 03-03-10 | 8068 | [redacted] | [redacted] | Equipment Issues | External Hazards |
| 28-01-08 | LOSS OF CONTROL OVER SAFETY RELATED MAINTENANCE | MG | 04-03-10 | 8069 | [redacted] | | | |
| 17-02-07 | CONSTRUCTION-RELATED EXPERIENCE WITH CABLES, CONNECTORS, AND JUNCTION BOXES: NRC IN 2010-02 | M | 19-03-10 | 8070 | USA | PALISADES | Management Issues | Design & Installation |
| 12-03-09 | FAILURES OF MOTOR-OPERATED VALVES DUE TO DEGRADED STEM LUBRICANT: NRC IN 2010-03 | MG | 22-03-10 | 8071 | USA | | Management Issues | Procedures |
| 12-11-08 | DIESEL GENERATOR VOLTAGE REGULATION SYSTEM COMPONENT FAILURE DUE TO LATENT MANUFACTURING DEFECT: NRC IN 2010-04 | M | 22-03-10 | 8072 | USA | PALO VERDE-2 | | |
| 06-04-09 | MANAGEMENT OF STEAM GENERATOR LOOSE PARTS AND AUTOMATED EDDY CURRENT DATA ANALYSIS: NRC IN 2010-05 | M | 22-03-10 | 8073 | USA | BRAIDWOOD-1 | Management Issues | Foreign Material Exclusion |

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| 03-11-08 | INADVERTENT CONTROL ROD WITHDRAWAL EVENT WHILE SHUTDOWN: NRC IN 2010-06 | M | 22-03-10 | 8074 | USA | DRESDEN-3 | Management Issues | Ineffective Use of OE, Recurrent events |
| 02-08-06 | SEAWEED INGRESS LEADING TO PARTIAL LOSS OF SAFETY RELATED REACTOR SEAWATER COOLING | M | 26-03-10 | 8075 | [redacted] | [redacted] | Equipment Issues | Degraded Cooling Conditions |
| 28-03-10 | IMPORTANCE OF UNDERSTANDING CIRCUIT BREAKER CONTROL PANEL INDICATIONS: NRC IN 2010-09 | M | 22-04-10 | 8076 | USA | H.B. ROBINSON-2 | | |
| 28-03-10 | FIRES AFFECTING SAFETY RELATED EQUIPMENT: PRELIMINARY NOTIFICATION II-10-002A (H.B. ROBINSON-2) | P | 22-04-10 | 8077 | USA | H.B. ROBINSON-2 | | |
| 18-03-09 | WELDING DEFECTS IN REPLACEMENT STEAM GENERATORS | M | 26-04-10 | 8078 | USA | SAN ONOFRE-3 | | |
| 23-10-05 | RECURRENT UNAVAILABILITY OF SG TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP | M | 28-04-10 | 8079 | [redacted] | [redacted] | IMPORTANT LESSONS | IMPORTANT LESSONS |
| 01-10-08 | WELDING AND NON-DESTRUCTIVE EXAMINATION ISSUES: NRC IN 2010-08 | M | 30-04-10 | 8080 | USA | PRAIRIE ISLAND-1 | | |
| 10-01-10 | UNIT SHUTDOWN DUE TO STEAM GENERATOR TUBE LEAKAGE | M | 26-04-10 | 8081 | [redacted] | [redacted] | Equipment Issues | Leakage |
| 28-05-08 | LACK OF SEISMIC RESISTANCE IN THE SUPPORTS FOR THE EFWS MOTOR-DRIVEN PUMP SUCTION LINE | MG | 21-05-10 | 8082 | [redacted] | | Management Issues | Control of Modifications |
| 15-07-09 | NON-COMPLIANT LUBRICATION GREASE MIXTURE IN QUALIFIED ACTUATORS | MG | 21-05-10 | 8083 | [redacted] | | Management Issues | Ineffective Use of OE, Recurrent events |
| 04-07-06 | UNIT POWER REDUCTION DUE TO EXTERNAL IMPACTS | MG | 08-04-10 | 8084 | [redacted] | [redacted] | | |
| 24-11-09 | OUTAGE DUE TO EXCESS SUSPENDED MATERIAL IN PLANT COOLING WATER (LAKE WATER) | M | 23-04-10 | 8085 | [redacted] | [redacted] | Equipment Issues | Degraded Cooling Conditions |
| 13-02-10 | PRIMARY CIRCUIT LEAK VIA A BANK OF PRESSURIZER HEATERS DUE TO CRACKED WELD | M | 18-05-10 | 8086 | [redacted] | [redacted] | | |

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| 02-03-09 | MASSIVE INGRESS OF PLANT DEBRIS INTO RAW WATER PUMPING STATION | M | 26-05-10 | 8087 | [redacted] | [redacted] | Equipment Issues | Degraded Cooling Conditions |
| 30-03-10 | RADIOACTIVE RESIN FOUND IN VENTILATION SYSTEM | M | 10-10-11 | 8088 | [redacted] | [redacted] | IMPORTANT LESSONS | IMPORTANT LESSONS |
| 30-03-10 | RADIOACTIVE RESIN FOUND IN VENTILATION SYSTEM | P | 23-06-10 | 8088 | [redacted] | [redacted] | | |
| 17-01-10 | BREACH OF SAFE OPERATING LIMITS, CAUSED BY SPONTANEOUS OPENING AND FAILURE TO CLOSE OF SG-1 SAFETY VALVE DUE TO FRACTURE OF CONTROL SYSTEM TUBE | M | 01-04-10 | 8089 | [redacted] | [redacted] | Management Issues | Design & Installation |
| 18-03-10 | INTERNAL FLOODING OF SAFETY RELATED 6 KV ELECTRICAL BUSBAR | M | 14-06-10 | 8090 | [redacted] | [redacted] | Equipment issues | Flooding |
| 04-07-05 | DEFICIENCIES IN PURCHASING SPECIFICATIONS FOR THE REPLACEMENT OF EQUIPMENTS | M | 31-05-10 | 8091 | [redacted] | [redacted] | Management Issues | Procurement & Spares |
| 04-01-08 | EMERGENCY FEEDWATER PUMP UNAVAILABILITY DUE TO LUBRICATION PIPE LEAK | M | 01-07-10 | 8092 | [redacted] | [redacted] | | |
| 19-12-08 | CONFORMITY DEVIATION AFFECTING QUICK BLEED VALVE MEMBRANES IN THE STEAM ISOLATION VALVE CONTROL | MG | 01-07-10 | 8093 | [redacted] | | Management Issues | Procurement & Spares |
| 03-10-08 | FAILURE OF A MOBILE HOIST | M | 01-07-10 | 8094 | [redacted] | [redacted] | | |
| 05-01-09 | ICING OF PROTECTIVE GRID AT CHOOZ B PUMPING STATION | MG | 02-07-10 | 8095 | [redacted] | | Equipment Issues | Degraded Cooling Conditions |
| 27-10-09 | SLOW ELECTRICAL TRANSFER CAUSES REACTOR TRIP | M | 09-07-10 | 8096 | [redacted] | [redacted] | | |
| 27-10-09 | SLOW ELECTRICAL TRANSFER CAUSES REACTOR TRIP | F | 22-09-10 | 8096 | [redacted] | [redacted] | Management Issues | Human Performance |
| 01-12-09 | LEAKAGE OF RADIOACTIVE LIQUID WASTE IN CONTROLLED AREA ON 2ND BASEMENT OF AUXILIARY BUILDING | M | 16-07-10 | 8097 | [redacted] | [redacted] | Management Issues | Procedures |
| 13-01-10 | THINNING OF RESIDUAL HEAT REMOVAL SEAWATER SYSTEM PIPING | M | 16-07-10 | 8098 | [redacted] | [redacted] | Equipment Issues | Cracks & Corrosion |

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| 19-11-09 | ACTIVE VENTILATION FILTER DISPOSAL DURING DECOMMISSIONING RESULTING IN SPREAD OF CONTAMINATION | M | 20-07-10 | 8099 | [redacted] | [redacted] | | |
| 01-04-09 | CIRCUMFERENTIAL CRACKING IN ALLOY 600 MA STEAM GENERATOR TUBES WITH PIERCED TUBE SUPPORT PLATES | MG | 04-08-10 | 8100 | [redacted] | | | |
| 13-11-09 | FOREIGN MATERIAL INTRUSION RESULTING IN POWER CHANGE DURING GENERATOR POWER INCREASING OPERATION | M | 16-07-10 | 8101 | [redacted] | [redacted] | Management Issues | Foreign Material Exclusion |
| 14-10-09 | THINNING OF SEAWATER PIPING OF HIGH PRESSURE CORE INJECTION DIESEL COOLING SYSTEM | M | 16-07-10 | 8102 | [redacted] | [redacted] | Equipment Issues | Cracks & Corrosion |
| 01-12-09 | DEFICIENCY IN IMPLEMENTATION OF A DIGITAL CONTROL SYSTEM UNDER 10 CFR 50.59: NRC IN 2010-10 | M | 23-08-10 | 8103 | USA | LASALLE-2 | | |
| 29-06-09 | NON CONFORMANCE WITH CRITICALITY CONTROL ARRANGEMENTS THROUGH INAPPROPRIATE OPERATIONAL DECISION MAKING | M | 28-07-10 | 8104 | [redacted] | [redacted] | Management Issues | Criticality Reactivity |
| 28-05-09 | OVER-INSERTION OF A CONTROL ROD DURING PERIODIC INSPECTION | M | 16-07-10 | 8105 | [redacted] | [redacted] | Equipment Issues | Control Rods & CRDM |
| 21-08-09 | DAMAGE OF THE TURBOCHARGER OF ONE EMERGENCY DIESEL GENERATOR | M | 16-07-10 | 8106 | [redacted] | [redacted] | Equipment Issues | Diesel Generator |
| 26-03-09 | ERRONEOUS INSERTION OF CONTROL ROD | M | 16-07-10 | 8107 | [redacted] | [redacted] | | |
| 28-09-10 | MECHANICAL DEFECTS (CRACKS) AT THE UPPER PART OF CONTROL ROD DRIVES HOUSING | P | 20-10-10 | 8108 | [redacted] | [redacted] | | |
| 21-04-10 | ADJUSTER ROD SPURIOUS OUT-DRIVE | M | 16-08-10 | 8109 | [redacted] | [redacted] | Equipment Issues | Control Rods & CRDM |
| 16-04-09 | TWO COMMON-CAUSE FAILURES OF FAST-ACTING ISOLATION VALVES | M | 18-08-10 | 8110 | [redacted] | [redacted] | Management Issues | Control of Modifications |
| 26-03-09 | INADVERTENT INSERTION OF CONTROL ROD DURING RESTORATION OF HYDRAULIC CONTROL UNIT OF CONTROL ROD DRIVE HYDRAULIC SYSTEM ( EVENT 1) | M | 16-07-10 | 8111 | [redacted] | [redacted] | Equipment Issues | Control Rods & CRDM |
| 06-04-09 | INADVERTENT INSERTION OF CONTROL ROD DURING RESTORATION OF HYDRAULIC CONTROL UNIT OF CONTROL ROD DRIVE HYDRAULIC SYSTEM ( EVENT 2) | M | 16-07-10 | 8112 | [redacted] | [redacted] | Equipment Issues | Control Rods & CRDM |

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| 23-03-09 | FULL INSERTION OF A CONTROL ROD OCCURRED DURING INCREASING OF GENERATOR OUTPUT | M | 16-07-10 | 8113 | [redacted] | [redacted] | | |
| 23-03-10 | CRACK ON WELD JOINT IN AIR-VENT PIPING OF CHARGING LINE OF CHEMICAL AND VOLUME CONTROL SYSTEM | M | 03-08-10 | 8114 | [redacted] | [redacted] | Management Issues | Control of Modifications |
| 16-03-10 | DAMAGE IN STEAM GENERATOR TUBE DUE TO PRIMARY WATER STRESS CORROSION CRACKING | M | 03-08-10 | 8115 | [redacted] | [redacted] | | |
| 04-08-08 | POTENTIAL FOR STEAM VOIDING CAUSING RESIDUAL HEAT REMOVAL SYSTEM INOPERABILITY: NRC IN 2010-11 | MG | 23-08-10 | 8116 | USA | | Management Issues | Procedures |
| 01-03-08 | CONTAINMENT LINER CORROSION: NRC IN 2010-12 | MG | 23-08-10 | 8117 | USA | | Equipment issues | Containment |
| 15-06-07 | FAILURE TO ENSURE THAT POST-FIRE SHUTDOWN PROCEDURES CAN BE PERFORMED: NRC IN 2010-13 | MG | 23-08-10 | 8118 | USA | | | |
| 30-11-09 | VALVE ACTUATOR HYDRAULIC OIL LEAK LEADING TO A FIRE AND MANUAL REACTOR SHUTDOWN | M | 13-09-10 | 8119 | [redacted] | [redacted] | Management Issues | Design & Installation |
| 01-06-09 | SPONTANEOUS OPENING OF SG SAFETY VALVE DUE TO SPURIOUS SIGNAL GENERATION IN THE VALVE CONTROL UNIT | M | 16-09-10 | 8120 | [redacted] | [redacted] | | |
| 06-07-10 | LOSS OF OFFSITE POWER DUE TO INCOMPLETE LOGIC FOR POWER TRANSFER | M | 24-08-10 | 8121 | [redacted] | [redacted] | | |
| 18-12-09 | MAIN GENERATOR TRIP DUE TO INADVERTENT ACTUATION OF UNDER FREQUENCY PROTECTION | M | 21-09-10 | 8122 | [redacted] | [redacted] | | |
| 13-01-10 | AUTOMATIC REACTOR TRIP FOLLOWING TURBINE TRIP CAUSED BY SET-POINT RELAY FAILURE | M | 27-09-10 | 8123 | [redacted] | [redacted] | | |
| 18-02-10 | MANUAL SHUTDOWN AND FORCED OUTAGE DUE TO FAILURES OF HIGH PRESSURE EMERGENCY CORE COOLING (HPECC) GAS ISOLATION VALVES | M | 27-09-10 | 8124 | [redacted] | [redacted] | Management Issues | Procurement & Spares |
| 11-05-10 | POWER REDUCTION ON TG-3 TRIP BY THE PROTECTION ON LO-LO LEVEL IN THE VOLUME CONTROL TANK OF GENERATOR SHAFT SEALING SYSTEM DUE TO PERSONNEL ERRORS | M | 28-09-10 | 8125 | [redacted] | [redacted] | Management Issues | Human Performance |
| 08-03-10 | REACTOR SCRAM DURING REACTOR PROTECTION SYSTEM TRIP TEST DUE TO INAPPROPRIATE OPERATOR ACTION | M | 22-09-10 | 8126 | [redacted] | [redacted] | Management Issues | Human Performance |

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| 05-05-10 | UNDUE ACTUATION OF ONE EMERGENCY DIESEL GENERATOR DURING OUTAGE DUE TO ARCING SHORT | M | 10-10-10 | 8127 | [redacted] | [redacted] | IMPORTANT LESSONS | IMPORTANT LESSONS |
| 23-12-09 | GUILLOTINE RUPTURE OF FIRE WATER LINE TO A STEAM GENERATOR IN REACTOR BUILDING | M | 11-10-10 | 8128 | [redacted] | [redacted] | Management Issues | Foreign Material Exclusion |
| 28-05-10 | MAIN STEAM VALVE FAILURE TO FULLY OPEN DURING PREPARATION FOR PLANT RESTART DUE TO FRACTURE OF VALVE LOCK ASSEMBLY | M | 15-10-10 | 8129 | [redacted] | [redacted] | | |
| 24-11-09 | INTERNAL EXPOSURE TO PLANT PERSONNEL DUE TO CONSUMPTION OF TRITIUM CONTAMINATED DRINKING WATER | M | 15-10-10 | 8130 | [redacted] | [redacted] | | |
| 02-12-08 | TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP REPETITIVE FAILURES: NRC IN 2010-20 | MG | 20-10-10 | 8131 | USA | | Management Issues | Ineffective Use of OE, Recurrent events |
| 04-10-09 | CRACK-LIKE INDICATION IN THE U-BEND REGION OF A THERMALLY TREATED ALLOY 600 STEAM GENERATOR TUBE: NRC IN 2010-21 | M | 20-10-10 | 8132 | USA | VOGTLE-1 | | |
| 01-01-09 | DEFICIENCIES IN THE QUALIFICATION OF A MANUFACTURER, IN QUALITY ASSURANCE AND CONTROLS DURING MANUFACTURING OF A SAFETY RELATED VALVE | MG | 09-11-10 | 8133 | [redacted] | | Management Issues | Procurement & Spares |
| 02-11-09 | FUEL ASSEMBLIES DAMAGED DURING REFUELING OPERATIONS DUE TO MISALIGNMENT: NRC IN 2010-22 | M | 17-11-10 | 8134 | USA | OCONEE-1 | Management Issues | Ineffective Use of OE, Recurrent events |
| 04-08-10 | MANUAL SCRAM CAUSED BY THE TRIPPING OF A GROUP OF BUSBARS IN 220 KV OPEN SWITCHYARD AND LOSS OF STANDBY POWER DUE TO CURRENT TRANSFORMER DESTRUCTION AND FIRE | M | 06-12-10 | 8135 | [redacted] | [redacted] | | |
| 01-10-10 | MALWARE ON PROGRAMMABLE CONTROLLER UNDER SIMATIC WINCC AND SIMATIC PCST | MG | 25-11-10 | 8136 | [redacted] | | | |
| 26-06-09 | HIGH OXYGEN CONTENT IN TWO GASEOUS WASTE TREATMENT TANKS CONSTITUING FIRE RISK | M | 28-10-10 | 8137 | [redacted] | [redacted] | | |

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| 30-05-08 | COMMON CAUSE FAILURE OF BOILING WATER REACTOR RECIRCULATION PUMPS WITH VARIABLE SPEED DRIVES: NRC IN 2010-17 | MG | 17-12-10 | 8138 | USA | | | |
| 01-02-08 | GENERIC ISSUE 199, "IMPLICATIONS OF UPDATED PROBABILISTIC SEISMIC HAZARD ESTIMATES IN THE CENTRAL AND EASTERN UNITED STATES ON EXISTING PLANTS": NRC IN 2010-18 | MG | 17-12-10 | 8139 | USA | | | |
| 22-10-09 | MALFUNCTIONS OF EMERGENCY DIESEL GENERATOR SPEED SWITCH CIRCUITS: NRC IN 2010-23 | MG | 17-12-10 | 8140 | USA | | | |
| 27-04-10 | LEAKING OF SEAWATER FROM SEAWATER PIPING FOR EMERGENCY DIESEL GENERATOR COOLING | M | 20-12-10 | 8141 | [redacted] | [redacted] | Equipment Issues | Leakage |
| 02-06-10 | REACTOR MANUAL SHUTDOWN DUE TO FAILURE OF STEAM LINE ISOLATION VALVE INSIDE CONTAINMENT OF REACTOR CORE ISOLATION COOLING SYSTEM | M | 20-12-10 | 8142 | [redacted] | [redacted] | | |
| 25-03-08 | INADEQUATE ELECTRICAL CONNECTIONS: NRC IN 2010-25 | MG | 17-12-10 | 8143 | USA | | Equipment Issues | Electrical Equipment Events |
| 11-09-07 | SUBMERGED ELECTRICAL CABLES: NRC IN 2010-26 | MG | 17-12-10 | 8144 | USA | | IMPORTANT LESSONS | IMPORTANT LESSONS |
| 11-06-10 | LEAKAGE OF SEAWATER FROM SEAWATER INLET PIPING OF REACTOR COMPONENT COOLING WATER HEAT EXCHANGER | M | 20-12-10 | 8145 | [redacted] | [redacted] | Equipment Issues | Cracks & Corrosion |
| 25-06-10 | REACTOR MANUAL SHUTDOWN FOR INSPECTION OF RESIDUAL HEAT REMOVAL SEAWATER SYSTEM | M | 20-12-10 | 8146 | [redacted] | [redacted] | | |
| 07-08-09 | FAILURE OF AN EMERGENCY DIESEL GENERATOR AFTER MANUFACTURER MAINTENANCE | P | 04-02-11 | 8147 | [redacted] | [redacted] | Equipment Issues | Diesel Generator |
| 18-12-08 | FALSE ACTIVATION OF RPV FLOAT LEVEL SWITCHES DUE TO EXCESSIVE CORE COOLING RATE AND ECCS FLOW OSCILLATION DURING FLOW REGULATED MODE | M | 06-12-10 | 8148 | [redacted] | [redacted] | | |
| 16-03-09 | UNAVAILABILITY OF RPS AUTOMATIC CONTAINMENT ISOLATION FUNCTION | M | 06-12-10 | 8149 | [redacted] | [redacted] | IMPORTANT LESSONS | IMPORTANT LESSONS |
| 16-05-10 | FAILURE OF SAFETY SYSTEM COMPONENTS (PRIMARY CIRCUIT BLOWDOWN / OVERPRESSURE VALVES) DUE TO INADEQUATE QUALIFICATION OF SLIGHTLY | M | 22-12-10 | 8150 | [redacted] | [redacted] | Management Issues | Control of Modifications |

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| | MODIFIED REPLACEMENT PARTS | | | | | | | |
| 09-07-10 | PERSONAL CONTAMINATION EVENT FOLLOWING VALVE OPERATION ON REDUNDANT EQUIPMENT | M | 05-01-11 | 8151 | [redacted] | [redacted] | | |
| 17-09-10 | INADVERTENT CONTAINMENT SPRAY DURING THE COMMISSIONING OPERATION | M | 06-01-11 | 8152 | [redacted] | [redacted] | Management Issues | Human Performance |
| 26-08-10 | AUTOMATIC REACTOR TRIP FOLLOWING LOSS OF GRID SUPPLIES TO STATION TRANSFORMER | M | 07-01-11 | 8153 | [redacted] | [redacted] | | |
| 01-01-08 | EXCESSIVE VIBRATIONS ON THE MOTOR OF A PUMP OF THE CONTAINMENT SPRAY CIRCUIT OWING TO TEMPERATURE EVOLUTION DURING AN ASME TEST | M | 26-01-11 | 8154 | [redacted] | [redacted] | | |
| 14-01-09 | VENTILATION SYSTEM PREVENTIVE MAINTENANCE AND DESIGN ISSUES: NRC IN 2010-27 | M | 02-02-11 | 8155 | USA | GRAND GULF-1 | | |
| 15-02-09 | COMMERCIAL-GRADE DEDICATION ISSUES IDENTIFIED DURING NRC INSPECTIONS: NRC IN 2011-01 | MG | 22-03-11 | 8156 | USA | | Management Issues | Procurement & Spares |
| 20-10-03 | OPERATOR PERFORMANCE ISSUES INVOLVING REACTIVITY MANAGEMENT AT NPPS: NRC IN 2011-02 | MG | 22-03-11 | 8157 | USA | | Management Issues | Criticality Reactivity |
| 01-09-08 | CONTAMINANTS AND STAGNANT CONDITIONS AFFECTING STRESS CORROSION CRACKING IN STAINLESS STEEL PIPING IN PWRS: NRC IN 2011-04 | MG | 22-03-11 | 8158 | USA | | Equipment Issues | Cracks & Corrosion |
| 26-03-10 | REDUCTION OF TRIP COVERAGE FOR SHUTDOWN SYSTEM N° 2 | M | 25-03-11 | 8159 | [redacted] | [redacted] | | |
| 03-08-09 | EXPOSURE OF TWO WORKERS IN EXCESS OF STATUTORY ANNUAL DOSE LIMITS | M | 23-03-11 | 8160 | [redacted] | [redacted] | Management Issues | Radiological Protection, Exposure, Contamination |
| 14-12-10 | UNIT SHUTDOWN DUE TO LOCAL LEAK IN REACTOR MAIN FLANGE | M | 28-03-11 | 8161 | [redacted] | [redacted] | | |
| 07-02-11 | BREACH OF LIMITING CONDITIONS OF OPERATION CAUSED BY SPONTANEOUS OPENING AND FAILURE TO CLOSE OF SG-1 S | M | 28-03-11 | 8162 | [redacted] | [redacted] | | |

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| 22-09-09 | PRESSURIZER MAIN SAFETY RELIEF VALVE FAILED TO CLOSE DURING ROUTINE PRESSURE BUILDUP TESTING | M | 10-05-11 | 8163 | [redacted] | [redacted] | Management Issues | Control of Modifications |
| 01-07-08 | NON-COMPLIANCE OF ROD BIG END BEARINGS IN EMERGENCY DIESEL GENERATOR ENGINES | PG | 20-05-11 | 8164 | [redacted] | | Equipment Issues | Diesel Generator |
| 26-11-09 | AIRBORNE ALPHA RADIATION EMITTING CONTAMINATION HAZARD RESULTS IN WORKER UPTAKES DURING REFURBISHMENT ACTIVITIES | M | 03-06-11 | 8165 | [redacted] | [redacted] | Management Issues | Radiological Protection, Exposure, Contamination |
| 06-09-10 | TURBINE AND REACTOR COOLANT PUMP TRIP FOLLOWED BY EMERGENCY FEED WATER PUMP START ON SPURIOUS SIGNALS | M | 27-06-11 | 8166 | [redacted] | [redacted] | | |
| 12-09-10 | ELEVATED LEVELS OF AIRBORNE ACTIVITY IN [redacted] UNIT 1 CONTAINMENT BUILDING | M | 13-07-11 | 8167 | [redacted] | [redacted] | Management Issues | Radiological Protection, Exposure, Contamination |
| 12-06-09 | CLOSURE OF A MAIN STEAM ISOLATION VALVE [redacted] AT 100% RATED POWER RESULTING IN REACTOR TRIP | M | 20-07-11 | 8168 | [redacted] | [redacted] | | |
| 26-05-10 | OUTAGES DUE TO TURBINE TRIP CAUSED BY FAILURES IN MAIN GENERATOR EXCITER | MG | 13-07-11 | 8169 | [redacted] | [redacted] | | |
| 25-02-11 | TRITIUM INTAKE BY WORKER DURING D2O SYSTEM MODIFICATION | M | 04-08-11 | 8170 | [redacted] | [redacted] | Management Issues | Radiological Protection, Exposure, Contamination |
| 08-04-09 | ACTIVATION OF THE EMERGENCY PROTECTION SYSTEM-III DUE TO INAPPROPRIATE ACTIONS | M | 23-06-11 | 8171 | [redacted] | [redacted] | | |
| 30-10-09 | LEAKING OF PRIMARY COOLANT THROUGH NOZZLES OF UPPER UNIT OF THE ENERGY MEASUREMENT CHANNELS (EMC) | MG | 23-06-11 | 8172 | [redacted] | [redacted] | Equipment Issues | Leakage |
| 02-04-11 | PRIMARY COOLANT LEAK FROM COOLANT PURIFICATION SYSTEM VALVES | M | 23-06-11 | 8173 | [redacted] | [redacted] | Equipment Issues | Leakage |
| 21-02-11 | NON-FULFILLMENT OF TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT DUE TO SET POINT VERIFICATION TESTS IN SAFETY VALVES CARRIED OUT WITHOUT COMPLYING WITH ASME CODE | M | 12-08-11 | 8174 | [redacted] | [redacted] | | |

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| 11-03-11 | TOHOKU-TAIHEIYOU-OKI EARTHQUAKE EFFECTS ON JAPANESE NUCLEAR POWER PLANTS: NRC IN 2011-05 | MG | 01-09-11 | 8175 | USA | | | |
| 28-08-10 | THERMAL ISSUES IDENTIFIED DURING LOADING OF SPENT FUEL STORAGE CASKS: NRC IN 2011-10 | MG | 01-09-11 | 8176 | USA | | | |
| 13-03-10 | CONTROL ROD SWELLING | M | 05-09-11 | 8177 | [redacted] | [redacted] | Equipment Issues | Control Rods & CRDM |
| 01-02-11 | CALIBRATION OF HIGH HEAD SAFETY INJECTION TO RCS COLD LEGS | MG | 06-09-11 | 8178 | [redacted] | | | |
| 06-02-11 | 500 KV OUTGOING LINE TRIPPED ON ELECTRICAL PROTECTION CAUSED BY HILL FIRE | M | 10-08-11 | 8179 | [redacted] | [redacted] | Equipment Issues | External Hazards |
| 04-02-11 | UNEXPECTED INSERTION OF A CONTROL ROD DURING OPERATION | M | 11-08-11 | 8180 | [redacted] | [redacted] | | |
| 18-02-10 | REACTOR TRIPS RESULTING FROM WATER INTRUSION INTO ELECTRICAL EQUIPMENT: NRC IN 2011-12 | MG | 07-09-11 | 8181 | USA | | Equipment issues | Flooding |
| 18-10-11 | REPORT ON FEEDER PIPE CRACK OF CHANNEL S-11 | P | 21-10-11 | 8182 | [redacted] | [redacted] | | |
| 30-05-11 | OVEREXPOSURE OF WORKERS BEYOND ANNUAL REGULATORY LIMIT | M | 13-09-11 | 8183 | [redacted] | [redacted] | Management Issues | Radiological Protection, Exposure, Contamination |
| 01-06-10 | INDICATIONS IN THE SAFE END OF THE NOZZLE OF THE MAIN COOLANT LINE CONNECTING TO THE SURGE LINE | M | 13-10-11 | 8184 | [redacted] | [redacted] | | |
| 14-06-11 | IMPACT OF ELECTRIC BREAKDOWN UPON ELECTRICIAN OF ELECTRIC DEPARTMENT FOLLOWING SHORT CIRCUIT IN 6 KV SWITCH-YARD DUE TO OCCUPATIONAL SAFETY VIOLATIONS DURING SWITCHOVERS | M | 18-08-11 | 8185 | [redacted] | [redacted] | Equipment Issues | Electrical Equipment Events |
| 01-10-10 | POWER REDUCTION DUE TO THE TRIP OF TURBINE-DRIVEN FEED WATER PUMP AS A RESULT OF ALGAE AND SILT INGRESS INTO THE CIRCULATION WATER SUPPLY | M | 07-09-11 | 8186 | [redacted] | [redacted] | Equipment Issues | External Hazards |
| 16-05-11 | UNIT SCRAM DUE TO LOSS OF CIRCULATION IN RPS CHANNELS COOLING CIRCUIT | M | 09-09-11 | 8187 | [redacted] | [redacted] | | |

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|---|---|---|---|---|---|---|---|---|
| 09-06-10 | VIBRATION PHENOMENA OBSERVED IN EFWS PUMPS | M | 20-09-11 | 8188 | [redacted] | [redacted] | | |
| 11-04-10 | REACTOR TRIP ON STEAM GENERATOR-A LEVEL HI-HI | M | 20-10-11 | 8189 | [redacted] | [redacted] | | |
| 27-04-11 | LOSS OF PRIMARY COOLANT INVENTORY DURING REFUELING OUTAGE | M | 24-10-11 | 8190 | [redacted] | [redacted] | Equipment issues | Flooding |
| 11-02-10 | POWER DECREASE DUE TO A BLOCKING OF THE SEA WATER PUMP INLET WITH SEAWEED | M | 11-08-11 | 8191 | [redacted] | [redacted] | | |
| 23-04-10 | MISSING STATUS INDICATION OF SEVERAL QUICK-OPERATED VALVES ON THE FIRST CATEGORY SUBSYSTEM OF THE POST- ACCIDENT MONITORING SYSTEM (PAMS) | M | 19-10-11 | 8192 | [redacted] | [redacted] | | |
| 01-09-09 | CRACK INDICATIONS AT THE REACTOR WATER CLEAN-UP PUMPS | M | 14-11-11 | 8193 | [redacted] | [redacted] | Equipment Issues | Cracks & Corrosion |
| 31-10-09 | LOSS OF OFF-SITE AC POWER SUPPLY CAUSED BY FAILURE OF MAIN TRANSFORMER | M | 11-08-11 | 8194 | [redacted] | [redacted] | | |
| 07-07-09 | DISCONTINUITY FLAWS WELDS OF "COLD"/"HOT" PRIMARY COLLECTOR TO THE STEAM GENERATOR | MG | 17-11-11 | 8195 | [redacted] | [redacted] | | |
| 29-04-11 | AUTOMATIC REACTOR SCRAM DURING UNIT RESTART AFTER UPGRADING DUE TO FAILURE TO DEACTIVATE THE TURBINE HI-HI VIBRATIONS PROTECTION | M | 03-11-11 | 8196 | [redacted] | [redacted] | | |
| 20-06-11 | INTAKE DEBRIS RESULTED IN UNIT POWER REDUCTIONS | M | 10-10-11 | 8197 | [redacted] | [redacted] | | |
| 04-12-09 | NON-AVAILABILITY OF THE MINIFLOW VALVE ON A MEDIUM HEAD SAFETY INJECTION PUMP | M | 30-11-11 | 8198 | [redacted] | [redacted] | | |
| 07-08-10 | UNIT SHUT DOWN DUE TO THE TRIPPING OF THE ONLY OPERATING TG-2 BY THE LO-LO VACUUM PROTECTION CAUSED BY DAMAGE TO THE FLANGE CONNECTIONS OF THE LOW PRESSURE CYLINDER | M | 25-11-11 | 8199 | [redacted] | [redacted] | | |
| 25-01-11 | INOPERABILITY OF BOTH INDEPENDENT CIRCUITS WITHIN THE ESSENTIAL SERVICE WATER SYSTEM DUE TO SAFEGUARD COOLING TOWER VALVES BLOCKED BECAUSE OF COLD WEATHER | M | 29-11-11 | 8200 | [redacted] | [redacted] | Equipment Issues | External Hazards |
| 01-08-10 | CONTROL ROD BLADE CRACKING RESULTING IN REDUCED DESIGN LIFETIME: NRC IN 2011-13 | MG | 12-12-11 | 8201 | USA | | Equipment Issues | Control Rods & CRDM |
| 04-01-10 | FAILURE OF STARTING AIR VALVES OF EMERGENCY DIESEL GENERATORS | M | 02-12-11 | 8202 | [redacted] | [redacted] | | |

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| 08-03-06 | COMPONENT COOLING WATER SYSTEM GAS ACCUMULATION AND OTHER PERFORMANCE ISSUES: NRC IN 2011-14 | MG | 12-12-11 | 8203 | USA | | | |
| 20-08-08 | [DEFICIENCIES IN] CALCULATION METHODOLOGIES FOR OPERABILITY DETERMINATIONS OF GAS VOIDS IN NUCLEAR POWER PLANT PIPING: NRC IN 2011-17 | MG | 13-12-11 | 8204 | USA | | | |
| 04-02-11 | INOPERABILITY OF THE QUICK OPERATIVE ISOLATING VALVES CAUSED BY CABLING DEGRADATION | M | 13-12-11 | 8205 | [redacted] | [redacted] | Management Issues | Control of Modifications |
| 01-01-77 | STEEL CONTAINMENT DEGRADATION AND ASSOCIATED LICENSE RENEWAL AGING MANAGEMENT ISSUES: NRC IN 2011-15 | MG | 13-12-11 | 8206 | USA | | Equipment issues | Containment |
| 01-08-10 | CONCRETE DEGRADATION BY ALKALI-SILICA REACTION: NRC IN 2011-20 | MG | 14-12-11 | 8207 | USA | SEABROOK-1 | Equipment issues | Containment |
| 26-11-99 | RECURRING FAILURES OF PROTECTIVE RUPTURE DISCS ON PRESSURIZER RELIEF TANKS | MG | 06-12-11 | 8208 | [redacted] | | | |
| 13-12-11 | REALISTIC EMERGENCY CORE COOLING SYSTEM EVALUATION MODEL EFFECTS RESULTING FROM NUCLEAR FUEL THERMAL CONDUCTIVITY DEGRADATION: NRC IN 2011-21: | MG | 15-12-11 | 8209 | USA | | | |
| 12-02-09 | EARTHQUAKE INDUCED SUBSIDENCE BETWEEN BUILDINGS | M | 19-12-11 | 8210 | [redacted] | [redacted] | Management Issues | Design & Installation |
| 24-10-08 | BROKEN CONTROL ROD SHAFT EXTENDERS | MG | 21-12-11 | 8211 | [redacted] | | | |
| 21-01-11 | MANUAL REACTOR SHUTDOWN DUE TO UNEXPECTED READINGS FROM THE CONDENSED WATER FLOW METER OF DRYWELL COOLING SYSTEM AND THE DRYWELL HIGH CONDUCTIVITY LIQUID WASTE SUMP LEVEL METER | M | 27-12-11 | 8212 | [redacted] | [redacted] | | |
| 04-12-10 | ERRONEOUS INSERTION OF A CONTROL ROD DURING PERIODIC INSPECTION OUTAGE | M | 27-12-11 | 8213 | [redacted] | [redacted] | Equipment Issues | Control Rods & CRDM |
| 02-11-10 | REACTOR AUTOMATIC SHUTDOWN DUE TO AN ACTIVATION OF A REACTOR WATER LEVEL-HIGH ALARM | M | 27-12-11 | 8214 | [redacted] | [redacted] | | |
| 30-03-11 | UNEXPECTED DECREASE OF THE MODERATOR GADOLINIUM CONCENTRATION DURING MODERATOR REFILL | M | 29-12-11 | 8215 | [redacted] | [redacted] | Management Issues | Criticality Reactivity |

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| 09-06-11 | SIMULTANEOUS RUPTURE OF OVER PRESSURE RELIEF DEVICES OF THE CALANDRIA DUE TO AN EXPLOSION WITHIN THE MODERATOR COVERGAS | M | 23-12-11 | 8216 | [redacted] | [redacted] | Management Issues | Procedures |
| 28-05-09 | INSTRUMENTATION AND CONTROL MODULE HARDWARE, CONFIGURATION, AND PROCEDURE ISSUES: NRC IN 2011-22 | MG | 30-12-11 | 8217 | USA | | | |
| 29-09-10 | REACTOR SCRAM AND MAIN STEAMLINE ISOLATION FROM HIGH RADIATION SIGNAL | M | 30-12-11 | 8218 | [redacted] | [redacted] | | |
| 16-03-10 | PRIMARY COOLANT LEAK CAUSED BY SWELLING AND MECHANICAL FAILURE OF PRESSURISER HEATERS | M | 19-09-11 | 8219 | [redacted] | [redacted] | | |
| 28-03-79 | ACCIDENT AT THREE MILE ISLAND UNIT 2 (MARCH 28, 1979): NRC BACKGROUNDER | MG | 30-12-11 | 8220 | USA | THREE MILE ISLAND-2 | | |
| 30-05-11 | PARTIAL INJECTION OF LIQUID POISON FROM TWO OF THE SIX TANKS OF THE SECONDARY SHUTDOWN SYSTEM DUE TO THE INSTALLATION OF NON-CONFORMING COMPONENTS | M | 31-12-11 | 8221 | [redacted] | [redacted] | | |
| 15-06-11 | AUTOMATIC REACTOR TRIP AFTER SAFETY INJECTION ACTUATION DURING PERFORMANCE OF A SURVEILLANCE | M | 29-09-11 | 8222 | [redacted] | [redacted] | | |
| 31-05-11 | FIRE IN DIESEL GENERATOR EXHAUST SYSTEM LEADING TO DIESEL GENERATOR INOPERABILITY | M | 28-11-11 | 8223 | [redacted] | [redacted] | | |
| 09-04-11 | UNAVAILABILITY OF EMERGENCY FEEDWATER PUMP (EFP) DUE TO TRIP FROM OVERCURRENT OF ITS ELECTRICAL CIRCUIT BREAKER | M | 12-12-11 | 8224 | [redacted] | [redacted] | Equipment Issues | Electrical Equipment Events |
| 16-09-10 | UNAVAILABILITY OF STANDBY DIESEL GENERATOR DUE TO FAILURE OF AN ENGINE BEARING | M | 31-12-11 | 8225 | [redacted] | [redacted] | | |

# ANNEX II. CONTRIBUTORS TO DRAFTING

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| Canada | [redacted] | [redacted] | [redacted] |
| Germany | [redacted] | [redacted] | [redacted] |
| India | [redacted] | [redacted] | [redacted] |
| Pakistan | [redacted] | [redacted] | [redacted] |
| Spain | [redacted] | [redacted] | [redacted] |
| United Kingdom | [redacted] | [redacted] | [redacted] |
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|---|---|---|
| NEA | [redacted] | [redacted] |
| EC | [redacted] | [redacted] |
| WANO | [redacted] | [redacted] |
| IAEA | [redacted] | [redacted] |