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International Atomic Energy Agency

IRS Guidelines

2022 Edition

Joint IAEA and OECD/NEA International Reporting System for Operating Experience (IRS)

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NEA
NUCLEAR ENERGY AGENCY

Vienna, April 2022

IAEA Services Series 19 (Rev. 1)

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

Information on the IAEA's safety standards programme is available at the IAEA Internet site

www.iaea.org/resources/safety-standards

The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at: Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

RELATED PUBLICATIONS

The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

Reports on safety in nuclear activities are issued as **Safety Reports**, which provide practical examples and detailed methods that can be used in support of the safety standards.

Other safety related IAEA publications are issued as **Emergency Preparedness and Response** publications, **Radiological Assessment Reports**, the International Nuclear Safety Group's **INSAG Reports**, **Technical Reports** and **TECDOCs**. The IAEA also issues reports on radiological accidents, training manuals and practical manuals, and other special safety related publications.

Security related publications are issued in the **IAEA Nuclear Security Series**.

The **IAEA Nuclear Energy Series** comprises informational publications to encourage and assist research on, and the development and practical application of, nuclear energy for peaceful purposes. It includes reports and guides on the status of and advances in technology, and on experience, good practices and practical examples in the areas of nuclear power, the nuclear fuel cycle, radioactive waste management and decommissioning.

IRS GUIDELINES

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

IAEA SERVICES SERIES No. 19 (Rev. 1)

IRS GUIDELINES

2022 EDITION

JOINT IAEA AND OECD/NEA INTERNATIONAL REPORTING SYSTEM
FOR OPERATING EXPERIENCE (IRS)

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2022

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FOREWORD

The International Reporting System for Operating Experience (IRS) is jointly operated by the IAEA and the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEA). The fundamental objective of the IRS is to contribute to improving the safety of commercial nuclear power plants worldwide. This objective can be achieved by providing timely and detailed information on lessons learned from designing, constructing, commissioning, operating and decommissioning experiences at the international level. This information may be related to issues and events that are relevant to nuclear safety.

The purpose of these guidelines is to describe the IRS and to give users the necessary background information and guidance to enable them to produce high quality IRS reports while retaining the effectiveness of the system expected by all Member States operating nuclear power plants. These guidelines supersede the original IRS Guidelines publication issued as IAEA Services Series No. 19 in 2010. As the IRS is owned by the Member States of the IAEA and the OECD/NEA, the IRS Guidelines have been developed and approved by the IRS National Coordinators with the assistance of the Secretariats of the IAEA and the OECD/NEA.

The IAEA officers responsible for this publication were D. Zahradka and H. Morgan of the Division of Nuclear Installation Safety.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The fundamental objective of the International Reporting System for Operating Experience (IRS), jointly operated by the International Atomic Energy Agency (IAEA) and the Organization for Economic Co-operation and Development Nuclear Energy Agency (OECD/NEA), is to contribute to improving the safety of nuclear power plants (NPPs) worldwide by providing information on lessons learned from incidents occurring during construction, commissioning, operation and decommissioning of NPPs.

In 2019, the scope of the IRS was expanded to include events reported by OECD/NEA Member States through the construction experience (ConEx) database operated by the OECD/NEA to improve international sharing of lessons learned from such events. Event reports recorded in the ConEx database were migrated into the IRS and future events from construction and commissioning of new NPPs will be shared by IAEA Member States through the IRS.

To support Member States in reviewing the applicability of lessons learned from events reported through the IRS, a new system for assessing event significance was developed and integrated into the IRS web-based system in 2020. Additionally, the publication has been updated to refer to the recommendations provided in IAEA Safety Standard Series No. SSG-50, Operating Experience Feedback for Nuclear Installations [1].

This publication is a revision of Services Series No. 19[1], which it supersedes. This publication and its companion publication, Services Series No. 20 (Rev. 1), Manual for IRS Coding [2], provide new practical guidance on reporting events from the construction and commissioning stages and using the new system for assessing event significance.

1.2. OBJECTIVE

The purpose of Services Series Nos 19 (Rev. 1) and 20 (Rev. 1) is to describe the IRS system and to give users the necessary background and practical guidance to produce IRS reports that meet expectations, and to use reports submitted by other Member States to share lessons learned that could help prevent the occurrence of similar events at their NPPs.

The primary audience of these publications are the IRS National Coordinators, regulatory bodies, operating organizations, technical support organizations, representatives from the IAEA and OECD/NEA, and specialists of the nuclear community who use IRS as a source of detailed information on lessons learned from operating experience.

1.3. SCOPE

The scope of this publication is the information relevant to the use of the IRS by Member States with IRS membership. This publication does not provide guidance for the use of related reporting systems IRSRR and FINAS, which are specific to event reporting for research reactors and fuel cycle facilities, respectively.

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[1] INTERNATIONAL ATOMIC ENERGY AGENCY, IRS Guidelines, Services Series No. 19, IAEA, Vienna (2010).

1.4. STRUCTURE

Section 2 of this publication describes the background of the IRS system and Section 3 describes the scope and objectives of international reporting system. Requirements for membership and access to the system and its user roles are addressed in Sections 4 and 5. Instructions for event reporting and categorization are addressed in Sections 6–8. Additional administrative details are addressed in Section 9.

Five appendices are provided that provide more specific information on event reporting and IRS governance.

2. BACKGROUND OF THE IRS

At the end of 1978, the OECD/NEA established an international system for exchanging information on safety-related events occurring in operating NPPs. In March 1979, the accident at Three Mile Island (TMI) provided further impetus to the development of an effective international operational experience feedback process.

The TMI accident accelerated the process of establishing an international incident reporting system. In January 1980, the Incident Reporting System (IRS) was launched for a two-year trial period. By the end of 1981, OECD/NEA countries formally approved the operation of the system. In April 1983, the IAEA extended the IRS to all its Member States with nuclear power programmes. The accident at Chornobyl in April 1986 resulted in further recognition by regulatory bodies and agencies in various Member States of the importance of an effective event reporting and operating experience exchange system.

With the creation of the first comprehensive database on the IRS, the Advanced Incident Reporting System (AIRS), in 1995, the responsibility of processing and reviewing reports (including quality checking) was transferred to the IAEA. Many topical and other studies have been produced by the IAEA over the years. Topical studies constitute an important component of IRS related activities. Such studies are intended to provide the basis for generating relevant data and in-depth evaluations and to identify topical or generic issues for wider interest and consideration for the users. The identification of such issues may begin with a national assessment by a country that accepts a lead role and then the issues may be studied in depth by experts at the international level, as warranted.

In 1994, the Convention on Nuclear Safety (CNS) [3] came into force. Article 19 of this Convention states:

> "Each Contracting Party shall take the appropriate steps to ensure that:
>
> (vi) incidents significant to safety are reported in a timely manner by the holder of the relevant licence to the regulatory body;
>
> (vii) programmes to collect and analyse operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies".

Requirement 15 of IAEA Safety Standards Series No. GSR Part 1 (Rev. 1) Governmental, Legal and Regulatory Framework for Safety [4] states:

> "**The regulatory body shall make arrangements for analysis to be carried out to identify lessons to be learned from operating experience and regulatory experience, including experience in other States, and for the dissemination of the**

> **lessons learned and for their use by authorized parties, the regulatory body and other relevant authorities."**

Paragraph 3.5 of GSR Part 1 (Rev. 1) [4] states:

> "To enhance the safety of facilities and activities globally, feedback shall be provided on measures that have been taken in response to information received via national and international knowledge and reporting networks."

Paragraph 3.5A of GSR Part 1 (Rev. 1) [4] states:

> "Relevant information and lessons learned from operating experience and regulatory experience shall be reported in a timely manner to international knowledge and reporting networks."

In 2006, the Web Based IRS (WBIRS) was created to facilitate efficient data input and report availability. Passwords are provided to users who are officially registered, and appropriate levels of access are assigned to individuals, thus ensuring a high level of security. Once a new report is posted on the WBIRS, the users are automatically informed by email. With the creation of the WBIRS, easy access to the information was expanded to utilities and plant staff, and the need for CD distribution and hard copies was eliminated. To reflect the evolution of the Incident Reporting System to one which includes an expanded view/use of operating experience feedback, the name of the system was revised to "International Reporting System for Operating Experience" in 2010. The system retained the abbreviation 'IRS'.

In 2019 the scope of the WBIRS was expanded to include events reported through the Construction Experience (ConEx) programme of OECD/NEA. The ConEx programme was initiated in 2008 with the objective of exchanging construction experience within OECD/NEA Member States. The main activity of this programme was the development of a ConEx database of reported construction events. The inclusion of the ConEx database in the WBIRS enables IAEA Member States to benefit from the lessons learned from construction events. New construction events are now reported through the WBIRS.

3. OBJECTIVES AND SCOPE OF THE IRS

The main objective of the IRS is to ensure the feedback from operating experience gained during the design, construction, commissioning, operating and decommissioning phases of nuclear power plants worldwide is broadly shared amongst the international nuclear community to enhance safety. For ease of understanding in the use of these revised Guidelines, the terms 'events', or 'events/information' are intended to mean any events, issues and relevant operating experience, such as good practices, lessons learned, or other findings. The IRS provides a platform for the effective collection, processing, dissemination, and retrieval of operating experience information for Member States. The IRS also ensures the information is useful to the users by providing them with a clear understanding of both the lessons learned and the corrective or regulatory actions taken.

Events/information from operating experience reported to the IRS are expected be of safety significance or have lessons learned that need to be shared with the international community to improve nuclear safety. These events and lessons learned can be useful in preventing the occurrence or reoccurrence of similar nuclear power plant events through the implementation of corrective actions and preventive measures.

Events/information are expected to be reported to the IRS in a timely manner. Early information on significant events in one country may help in avoiding a similar problem in other countries;

this is an important feature of the system. The IRS is a system based on the commitment of the participating countries which benefit from the exchange of information. The IRS relies heavily upon Member States being responsible for selecting the events/information being reported to the system.

As there are differences in design, construction, and operation of nuclear power plants, the IRS has to provide sufficient detail to highlight the wider relevance of the operating experience to the users. Therefore, IRS reports need to provide relevant detailed technical, organizational and human factors information, information on root causes, safety significance, lessons learned, and corrective actions.

The IRS is designed for professionals and specialists of the nuclear community as a source of detailed information on events and lessons learned from operating experience. As such, the information in the IRS is restricted to the nuclear community, in contrast to the information provided under International Nuclear and Radiological Event Scale (INES) service which is open to the public and is intended for communicating to a broader range of entities and stakeholders.

4. IRS MEMBERSHIP

It is assumed that a Member State which intends to participate in the IRS:

- Is planning to embark or has already effectively embarked on a nuclear power programme;
- Has established a regulatory body with the appropriate functions, responsibilities and authority for regulating the safety of nuclear power plants;
- Is a contracting party to the Convention on Nuclear Safety;
- Has established a national operating experience feedback system, in accordance with the recommendations provided in SSG-50 [1];
- Has assigned to an appropriate organization, responsibility for exchange of information on operating experience under the IRS.

A Member State that is planning to embark or has effectively embarked on a nuclear power programme and has committed itself formally to comply with the IRS guidelines, is encouraged to contact the IAEA. The OECD/NEA database for reporting construction experience from new nuclear power plants (CONEX) has been merged with the IRS system. Thus, the IAEA encourages Member States to apply for the IRS membership before starting the construction phase of a new nuclear power plant.

A Member State that wishes to participate in the IRS is expected to make a formal request to the IAEA after satisfactory arrangements have been made to meet the criteria set out above. A Member State becomes a member in the IRS when the membership request is approved by the IAEA. The Member State then nominates an IRS National Coordinator and the IAEA updates the list of IRS National Coordinators. All IRS Member States are expected to designate a person in the organization (usually the regulatory body) responsible for the exchange of information on operating experience under the IRS. This person is hereinafter called an 'IRS National Coordinator'. Member States may designate more than one IRS National Coordinator.

If any of the above assumptions are not valid for a particular Member State intending to participate in the IRS, then the IAEA will review the membership request, taking into account the principles of sharing and use of operating experience through the IRS contained in these guidelines. The final decision on the request is made by the IAEA in consultation with

OECD/NEA and the Advisory Committee of the International Reporting System (See Section 9.5). The decision and its basis are then communicated with the particular Member State. The IAEA informs all Member States participating in the IRS about the result during the Annual Technical Meeting of the International Reporting System for Operating Experience National Coordinators.

The IRS is based on the principle that each IRS Member State provides timely information on operating experience at its nuclear power plants so that it becomes available to all other participants in accordance with the arrangements set out in these guidelines.

5. ACCESS TO AND USE OF THE IRS

The IRS is for use within each Member State by organizations professionally involved in the nuclear power industry, such as:

- Regulatory bodies;
- Utilities with planned or ongoing nuclear power programmes;
- Technical support organizations in the nuclear power field;
- Vendors, designers, manufacturers, constructors and suppliers of nuclear power plants;
- Research establishments and technical universities working in the nuclear power field.

Organizations wishing to nominate their staff to become generic users of the WBIRS need to contact their respective IRS National Coordinator with the request. The organization can then be recognized by the IRS National Coordinator as an organization professionally involved in the nuclear power industry within their country, and generic users are individually approved by the IRS National Coordinator. The IRS National Coordinator then contacts the IAEA with the request via email or an official letter. In the case of organizations that employ workers with foreign citizenships, the same process is to be observed, but the employee's Nucleus account must be associated with their employer’s email address. To establish WBIRS access, users need to register themselves in the IAEA NUCLEUS information resource portal with an official email address provided by their organization. The IAEA can provide the necessary support to help generic users gain access to the WBIRS.

The IRS mainly focuses on events/information that are safety-related and/or have potential for lessons to be learned internationally, including precursors of serious events, the results of the analysis carried out, and the lessons to be learned. The IRS provides an organized set of data, easily transferable to situations in other countries, allowing an efficient feedback process. The IRS also enables the members to share information on the actions they have taken in response to the events/information reported by IRS members from other countries. This efficient exchange of information among National Coordinators can be done through the ‘Countries Actions’ field available for each IRS event report, which is another important source of information that may help prevent the occurrence of similar events in other countries.

The IRS relies upon the national operating experience systems of Member States and complements them by providing an international perspective. IRS Member States are expected to review the operating experience information available within their own countries for potential international applicability and report any relevant information to the IRS.

The IRS is an important source of information for regulatory bodies and their technical support organizations, providing them with insights on important international operating experience to support their regulatory activities. In addition to the IRS and national operating experience feedback systems, all stakeholders are encouraged to establish processes for receiving operating experience information from all available sources to provide insights from a diverse range of perspectives (e.g. the World Association of Nuclear Operators (WANO)).

Information on events, anomalies, situations and conditions usually originates from an operating organization and is then shared with the regulatory body, other operating and research organizations, designers, contractors or other relevant parties, in accordance with national reporting requirements and other established networks. These diverse sources of information provide a range of perspectives which may enhance the identification of potential safety improvements.

Through active participation and dissemination of operating experience reports, all Member States potentially benefit from learning opportunities identified in these reports to improve operation and safety. Feedback can be collected through methods such as the "Countries Action" field in WBIRS and through the technical discussions held during the Technical Meetings of the IRS National Coordinators.

6. COLLECTION AND DISTRIBUTION OF IRS INFORMATION

The WBIRS is a restricted information platform. It allows for the submission of reports and access by registered users. An IRS generic user can view and search for reports. In addition to viewing and searching, an IRS National Coordinator can submit reports. The IAEA reviews, verifies, and approves new reports. Once a new report has been approved, IRS users are notified and the report is made available to them. The WBIRS platform also provides search and visualization capabilities to help identify important events and trends.

As a result of the detailed technical character of the information provided by participating countries, IRS reports are classified as 'Restricted' to encourage open and timely exchange of information among participants. This condition was accepted when the system was established and remains in force. Once an IRS report is transmitted to each IRS National Coordinator, it is their responsibility to decide on its further distribution for official use within their own country.

Information available in the IRS is restricted to registered users. Any IRS user wishing to share IRS information with a third-party person or organization needs to obtain the prior consent of the respective IRS National Coordinator (for national IRS reports) and the IAEA (for all other IRS reports).

7. REPORTING

7.1. SELECTION OF EVENTS AND INFORMATION FOR REPORTING

Reported events/information are expected to be safety significant or reveal lessons learned that could help the international nuclear community improve the safety of nuclear power plants. The information may cover safety related issues, lessons learned, and good practices. These events/information may be reported to the IRS during the lifetime of a nuclear power plant, including design, manufacturing, construction, installation, commissioning, operation, and decommissioning.

In general, events/information are expected to be reported to the IRS when any of the following criteria are met:

(a) Safety significant events/information are considered reportable if there is an actual or potential significant reduction in the plant's defence in depth, for example:

i. A release of radioactive material or exposure to radiation for site personnel or members of the public above the prescribed limits;
ii. Failures, degradation or design weakness of structures, systems, or components, or human errors, that might challenge the operability of safety functions;
iii. Conditions causing vulnerability to internal and external hazards (earthquake, fire, flooding, severe weather, high winds, etc.) that might challenge the operability of the safety functions;
iv. Operating events, typically plant transients, accompanied by additional equipment failures, human errors or other anomalous indications. Operating events with anticipated transients and expected operation of post-trip systems without additional equipment failures, human errors or other anomalous indications need only be reported if they meet any other criteria listed in this section.

(b) Events/information can be considered reportable if they reveal important lessons which may help prevent occurrence of a safety significant event or to strengthen defence in depth, for example:
i. Unanticipated release of radioactive material or exposure to radiation of site personnel or members of the public;
ii. Common cause failure: failures of two or more structures, systems, or components due to a single specific event or cause. For example, the single specific event or cause of failures (which may be failures of different types) could be a design deficiency, a manufacturing deficiency, operation or maintenance errors, a natural phenomenon, human induced event saturation of signals, or an unintended cascading effect from any other operation or failure within the plant or from a change in ambient conditions;
iii. New degradation mechanism, safety analyses or research results, showing a previously unknown weakness in a safety system, or issues with fuel integrity, reactor coolant system pressure boundary integrity, or containment integrity or other findings;
iv. Organizational or human factor issues such as high human error rates, weaknesses in the safety management system, inadequate procedures, inadequate training or inadequate control of contractors;
v. Major changes to surveillance/maintenance programmes or regulatory requirements based on events and information;
vi. Latent weakness or conditions resulting in non-compliance to regulatory requirements.

(c) Operating experience often includes precursors or contributors to more significant events. Consequently, the reporting need not be limited only to the criteria listed in A and B above, but also to lower-level events, good practices, and corrective actions that contain lessons that may be useful to others. Events that are the repetition of similar events previously reported to the IRS may still convey new lessons to the international community.

7.2. REPORTING CATEGORIES

The events/information need to be reported under the following categories:

- Unanticipated releases of radioactive material or exposure to radiation;
- Degradation of barriers and safety system and items important to safety;
- Deficiencies in design, construction (including manufacturing), installation and commissioning, operation (including maintenance and surveillance), safety management/quality assurance system, safety evaluation, and decommissioning;
- Generic problems of safety interest;
- Consequential actions taken by the regulatory body;
- Events (including precursors, emerging trends or patterns) of potential safety significance;

- Effects of unusual events of either human or natural origin;
- Other findings and operating experience information.

These categories are expanded upon in more detail in Appendix I.

7.3. CONTENT, REPORTING TIME, AND FORMAT OF THE REPORTS

The IRS is intended to promote easy and prompt reporting and encourage open contacts among people responsible for operating experience feedback in participating countries.

The reporting is expected to provide the international community with:

- Nuclear power plant information and relevant data;
- A narrative description of the event/information (including the plant specific technical data necessary to understand all the causes and consequences);
- A safety assessment;
- A cause analysis (explaining the direct and root causes and causal factors);
- The lessons learned and corrective actions taken, including any actions by the regulatory body.

Detailed guidance on the preparation of IRS reports is provided in Appendix II.

The lessons learned and actions taken are expected to be clear and understandable to the international community to facilitate the assessment of the applicability to the situation in other countries.

The report also includes a cover sheet with standardized information and an abstract giving the essential characteristics of the event/information, as well as codes facilitating information retrieval (refer to Ref. [2]).

The report is expected to be provided in a timely manner, i.e. as soon as all the necessary information is available. For events with particularly important lessons to be learned and/or the need for information transfer to other countries, a preliminary report needs to be submitted, consisting of a brief description of the event and all relevant preliminary findings. The preliminary report is expected to be submitted as soon as practicable, preferably within one month after the event. This preliminary report is to be followed by a main report.

For safety significant events/information, it is important to understand what Member States have done to address the identified lessons learned. After receiving a report identified as ‘Priority Level 1’ by the Event Review Group of the IAEA (see Section 8), the IRS National Coordinators are encouraged to review and process the report in accordance with their respective national practices and then provide a response to the IAEA using the ‘Countries Actions’ field in the WBIRS. This response is expected to include information on any actions taken by the Member State to address the lessons learned documented in the ‘Priority Level 1’ report.

A ‘Priority Level 1’ designation is intended to indicate the generic applicability of a report determined by the Event Review Group of the IAEA. A Member State may come to a different conclusion after reviewing the report within the respective national context.

The IRS National Coordinators may also provide a response statement for any other IRS report when deemed relevant.

The format for the preparation of IRS reports is to be used to the extent practicable; however, flexibility may be applied for practical reasons, such as specific types of reports or different national requirements. The standard format and content of IRS reports may be considered for adoption into national systems for operational experience feedback to link national and international systems more efficiently.

The IRS is operated in English. While one of the other official languages of the Agencies may be used, Member States are encouraged to submit IRS reports in English to avoid undue delays. IRS reports provided in a language other than English may be included in the WBIRS as an attachment.

8. PRIORITY LEVEL

Each IRS report is assigned a Priority Level, by the Event Review Group (ERG) of the IAEA, based on the safety significance and/or importance of the lessons learned. The main purpose of this Priority Level is to facilitate the application of a graded approach proportionate to the actual or potential consequences of the events/information and the significance of lessons learned. This graded approach can be used to support the development of various IAEA operating experience related publications including the Nuclear Power Plant Operating Experience published jointly with the NEA, and often referred to as the 'Blue Book' [5]. Examples of reports with different Priority Levels are provided in Appendix III.

A Priority Level is not assigned to good practices reported via the IRS.

IRS reports are categorized into three different Priority Levels, as described below.

Priority Level 1 is assigned to the following types of events:

- Events that caused or had the potential to cause a major reduction in the defence in depth or safety functions of a plant;
- Events that caused or had the potential to cause excessive radiation exposure or serious harm to individuals;
- Safety significant events as determined by other inputs including Probabilistic Safety Assessment /Probabilistic Risk Assessment (PSA/PRA) evaluations or INES ratings that may have been provided in the IRS report. A breakdown of multiple barriers with major impact on plant safety is a typical event for this category. These events have significant generic applicability, or provide significant lessons learned.

Priority Level 1 events are used to provide specific suggestions to Member States through Agency publications developed for the industry (such as the Blue Book [5]).

Priority Level 2 is assigned to the following types of events:

- Events that reveal important lessons, which caused or had a potential to cause a limited impact on safety functions. Typically, these events cause an unexpected change in plant conditions, equipment status, or had an adverse effect on plant safety. Lessons learned from these events may help the international nuclear community prevent a recurrence of the event or the occurrence of more serious events in other MS (Level 1 events);
- Reoccurring events that would have been prevented by the implementation of corrective actions and preventive measures based on the lessons learned from previous similar events;
- Events involving organizational or human factor issues such as those caused by a degraded safety culture at a plant.

Individually, Priority Level 2 events might not have generic applicability but may provide inputs on important adverse trends to agency publications including the Blue Book.

Priority Level 3 is assigned to events that did not result in notable consequences but had the potential to cause events that are consequential under slightly different circumstances, and if shared, may help the international nuclear community identify potential precursors and prevent a recurrence of the event or the occurrence of more serious events (Priority Level 1 or 2 events). These events, by themselves, usually have little generic importance but may be useful in providing inputs on adverse trends to agency publications including the Blue Book [5].

Member States that submit a report may make an informed suggestion on the Priority Level; however, the IAEA ERG makes the final determination of Priority Level to ensure consistency in the grading system. It is expected there will be cases where the ERG determines an event has a different Priority Level than what is suggested by the reporting Member State based on consequences, perceived generic applicability, safety significance determination (e.g. PSA/PRA or INES rating), or the importance of the lessons learned. In these cases, the ERG will provide a relevant explanation to the reporting Member State.

9. IRS OPERATION AND MANAGEMENT

This section describes the roles and relationships of the respective parties involved in the operation and management of the IRS system. These are the participating countries, the IAEA and OECD/NEA Secretariats, the Technical Committee of IRS National Coordinators, the meeting of IRS National Coordinators to exchange experience on recent events in Nuclear Power Plants, the OECD/NEA Working Group on Operating Experience, the IRS Advisory Committee and the ERG.

9.1. ROLE OF PARTICIPATING COUNTRIES

Since the introduction of the Convention on Nuclear Safety [1], Contracting Parties have been required to share their national operating experience information. The effectiveness of this sharing depends on the quality of both the selection and presentation of the information being exchanged among the participating countries. Contracting parties to the Convention also need to ensure that all operating experience information with important lessons learned is reported to the international community in a timely manner.

The effectiveness of the IRS also largely depends on its regular use. Therefore, the IRS National Coordinators are expected to promote the use of the IRS in their respective countries. IRS National Coordinators are also expected to monitor IRS usage within their country to help improve and update the system. As users of the IRS, Member States agree on its objectives and decide on the improvements and modifications in reporting, the management of the IRS database, and the related activities to be performed.

Member States need to designate one or more IRS National Coordinator(s) to be responsible for the receipt and distribution of information obtained from the IRS and for the transmission of information to the IRS. The role of the IRS National Coordinator(s) is tied to the role of the Member State: e.g. promoting exchange, conducting training, and providing feedback on the use of IRS information within the Member State. The network of IRS National Coordinators can, via direct contacts, supplement the exchange of information. Member States are expected to allocate sufficient resources to make these exchanges effective.

While the overall responsibility for the use of operating experience lies with the operating organizations, the responsibility and accountability for the effective promotion of the system and its benefits, as well as the training to effectively use the system within the country is the responsibility of the IRS National Coordinator(s).

IRS National Coordinators are expected to:

- Demonstrate ownership of the system, by promoting the use of the system and showing leadership at the national and international level;
- Ensure report quality such that information is sufficiently comprehensive and commensurate with the timeliness of reporting (preliminary or final);
- Be given the necessary authority and tools to openly communicate to the IRS system any information of potential benefit to the international community;
- Participate in the annual technical meeting and present the national perspective by sharing events, lessons learned and corrective actions taken;
- Provide feedback to the Technical Committee of IRS National Coordinators to improve sharing and use of operating experience through the IRS.

Information provided to the IRS needs to be accurate, complete, understandable, user friendly, and easily retrievable.

Special efforts are expected to be made by IRS National Coordinators to ensure information provided is understandable to all IRS users. This includes the avoidance of abbreviations and jargon and the use of broadly accepted terms.

Dissemination of information is more effective if:

- All Member States are committed to not only using the system but also to reporting operating experience from their country to the system;
- System users provide appropriate resources (responsibility of the country and the different operating organizations, vendors, etc.) to ensure proper use of the disseminated information;
- Events and other operating experience information are reported proactively and in a timely manner;
- The information shared is easily understandable.

9.2. ROLE OF THE AGENCIES' SECRETARIATS

The IAEA and OECD/NEA Secretariats provide the legal framework, the infrastructure and technical support to operate the IRS. Both Secretariats coordinate their efforts to ensure activities sponsored by both Agencies are not duplicated and meet the expectations of participating countries.

The primary role of the IAEA is to operate the IRS and provide support to participating countries for the efficient operation and management of the system. In particular, the IAEA serves as a clearing house to:

- Compile, collate, and disseminate all information related to events/information reported to the system by participating countries;
- Translate IRS reports from the IAEA official languages to English, if necessary;
- Review the submitted reports, check for consistency and give feedback to the national coordinators, as needed;
- Request follow-up information, as needed;
- Propose reports to be highlighted and compile periodic reports;
- Compile nuclear power plant operating experience periodic reports from the IRS;
- Establish, operate, maintain, and update the WBIRS;
- Request Member States to provide reports on significant events shared by or available from other sources (e. g. the reports of IAEA peer review missions which are publicly

available; the IAEA Unified System for Information Exchange in Incidents and Emergencies (USIE), press information, etc.);
- Organize the development of topical studies;
- Perform other secretarial services regarding the IRS.

The Secretariats periodically produce a report on nuclear power plant operating experience (the Blue Book [5]). This report is generated from the IRS event reviews. The Secretariats also organize meetings of IRS National Coordinators on an annual basis. The locations of these meetings alternate between the IAEA and NEA/OECD Headquarters.

9.3. ROLE OF IRS MEETINGS

9.3.1 Annual Technical Meeting of the International Reporting System for Operating Experience - Recent Events in Nuclear Power Plants

This IRS meeting is organized each year to review the information received and to exchange information on recent events that have occurred in participating countries. This meeting is for IRS National Coordinators, representatives of the organizations responsible for nuclear power plants under construction, commissioning, operation, or decommissioning such as regulatory bodies, licensees, and technical support organizations registered in the IRS. The Secretariats may select events for in-depth presentation to be discussed during the IRS National Coordinators and OECD/NEA Working Group on Operating Experience meetings which are usually held during the following days.

9.3.2. Technical Committee Meeting of IRS National Coordinators

The Technical Committee of IRS National Coordinators meets annually to review the status of IRS operation, to discuss important operating experience/information presented during the above-mentioned Technical Meeting, and to present the current IAEA and OECD/NEA activities in this area. This meeting is restricted to IRS National Coordinators and their alternates. The participants review and analyse the activities performed within the framework of the IRS.

9.4. ROLE OF THE OECD/NEA WORKING GROUP ON OPERATING EXPERIENCE

The OECD/NEA Working Group on Operating Experience (WGOE) is an international forum for the exchange and analysis of operating experience for the determination of safety issues from a regulatory viewpoint. It also identifies safety issues that derive from operating experience for the Committee on the Safety of Nuclear Installations (CSNI) to consider additional in-depth analysis and research on identified safety issues. The WGOE, with the agreement of the Committee on Nuclear Regulatory Activities (CNRA), works to promote improvements in nuclear safety at nuclear installations, with a primary focus on NPPs.

The main activities undertaken by the working group include:

- Exchanging, analysing and providing expert insight regarding recent operating experience and reaching timely conclusions on lessons learned, trends, and associated responses;
- Producing reports, disseminating conclusions, organizing and sponsoring international meetings and workshops on the use of operating experience to improve nuclear safety;
- If necessary, providing feedback to the IRS Advisory Committee on areas for IRS improvement in the context of WGOE objectives;

- Comparing and, where possible, benchmarking international practices and methodologies applied by Member Countries in the assessment and use of operating experience;
- Maximizing the benefits of cooperation with other existing OECD/NEA working groups and other international organizations (e.g., IAEA, EC, WANO, etc.).

The WGOE membership is managed by the OECD/NEA. The IAEA participates in the WGOE as a member.

The WGOE meeting follows the annual meeting of the Technical Committee of IRS National Coordinators and is organized by OECD/NEA. It provides feedback, if necessary, to the IRS Advisory Committee on areas for IRS improvement in the context of the WGOE objectives. During the meeting, the group discusses the status of various reports related to operating experience. Some of these summary reports, when available, are included in the IRS database. This activity is restricted to WGOE members.

9.5. ROLE OF THE IRS ADVISORY COMMITTEE

The mandate of the IRS Advisory Committee (IRSAC) is to advise participating countries and the two Agencies in making the best use of the IRS and to ensure its effectiveness and performance. It supports Member States in keeping effective control over the system and advises the Secretariats in providing effective technical support.

The IRSAC plays an active role in providing guidance and advice on the IRS guidelines, operation and management of the system, and in giving its views on specific IRS matters as requested by the Agencies, participating countries, and the Technical Committee of IRS National Coordinators. The Advisory Committee provides suggestions for improvement regarding the operation of the IRS but does not carry out technical analysis.

The IRSAC meets a minimum of once a year to:

- Review the IRS objectives and guidelines and submit any proposals for significant change that affects the operation of the IRS to both Agencies for approval. This would normally be done through the Technical Committee Meeting of IRS National Coordinators;
- Provide advice and recommendations on the operation, maintenance, and improvements to the IRS to the Technical Committee of the IRS National Coordinators;
- Examine the quality and content of the incoming IRS reports with the purpose of improving their use for evaluation and analysis. Technical analysis is not within the scope of the IRSAC;
- Identify additional activities that can be performed based on IRS information, and to identify common supporting activities that would need to be performed to enhance the effectiveness of the IRS;
- Examine reporting practices to help ensure that important issues are reported to the IRS;
- Approve IRS reports to be highlighted in the IAEA periodic reports. The committee may use the Priority Levels described in Section 8 to identify appropriate reports.

The IRSAC submits its advice and recommendations regarding IRS operation and management to the Technical Committee of IRS National Coordinators for their review and approval. In case of a specific request, the IRSAC may also give its views and advice to an Agency or to one of the participating countries to whom the information is relevant.

The IRSAC is constituted of members elected by the National Coordinators. The process for nominating, electing, and appointing members is described in Appendix IV. The term of the elected members is four years.

9.6. ROLE OF THE IAEA EVENT REVIEW GROUP

All reports submitted to the WBIRS are reviewed by the IAEA Event Review Group (ERG) for their quality within 30 days of the receipt of the report. The safety significance, root cause analysis, corrective actions implemented and overall compliance with the IRS guidelines are discussed.

The ERG assigns a Priority Level to the reported events/information and discusses the responses provided by the participating countries in the ‘Countries Actions’ field of the IRS database, with a particular focus on the reports that were assigned Priority Level 1.

Members of the IAEA ERG are the IAEA IRS Coordinator and participants from various sections of the IAEA Department of Nuclear Safety and Security. The meeting is official when at least 4 ERG members are present. After the review meeting, any suggestions or comments intended to clarify the content of a report are sent to the respective IRS National Coordinator for review and action as appropriate. If the IAEA does not receive a response to the suggestions or comments within 1 month, the original version of the report that was sent to IAEA is published.

After the end of each calendar year, the ERG will review the events/information reported during the year, focusing on Priority Level 1 and Priority Level 2 reports, to identify the most significant issues in terms of safety and generic lessons learned. Additionally, all reported events/information will be reviewed to identify adverse trends or commonalities. The results of the review will be summarized in an annual IRS summary report. The report will be shared with the IRS National Coordinators.

APPENDIX I - IRS REPORTING CATEGORIES

This appendix discusses the categories of operating experience (events and information) to be reported. It provides background information on the reasons for their selection as well as a general description and examples of relevant events. The categories are intended to characterize the events and other information to be reported to the IRS. Events may fall into more than one category and all applicable categories are to be selected. The summary examples given here are intended to illustrate typical events and other information to be reported under each category to the IRS. The examples are not exhaustive, i.e., many other events or situations which are relevant for reporting to the IRS might not be covered by these categories. It is important to note that a report needs to be prepared not only because an event has occurred, but also because lessons learned have been identified.

The categories are:

- Category 1: Unanticipated releases of radioactive material or exposure to radiation;
- Category 2: Degradation of barriers and safety related systems;
- Category 3: Deficiencies in design, construction (including manufacturing), installation and commissioning, operation (including maintenance and surveillance), safety management/quality assurance system, safety evaluation, and decommissioning;
- Category 4: Generic problems of safety interest;
- Category 5: Enforcement and Consequential actions taken by the regulatory body;
- Category 6: Events of potential safety significance;
- Category 7: Effects of unusual events of either human or natural origin;
- Category 8: Other findings and operating experience information.

CATEGORY 1: UNANTICIPATED RELEASES OF RADIOACTIVE MATERIAL OR EXPOSURE TO RADIATION

Releases of radioactive material from the site directly impact the environment and might result in public exposure. Occupational exposure directly affects the plant personnel. The design and operation of a nuclear power plant incorporates features which prevent undue releases and exposures. Due to weaknesses in operational controls, design, etc., unanticipated releases or exposures may occur. This category is intended to report events/issues addressing actual or potential serious weaknesses in the provisions implemented, even if the prescribed limits have not been exceeded. Unanticipated exposures to workers and the public that exceed defined limits are also addressed in this category.

Category 1.1: Unanticipated releases of radioactive material

This category comprises unanticipated gaseous or liquid releases to the environment or within the site. Unanticipated spills or contamination events on site may pose problems for the safety of personnel or render access to on-site areas difficult, which may also result in challenges to the control of items important to safety.

Examples:

(a) Four electric fan motors from the confinement vapour recovery system and vault vapour recovery system at an NPP were shipped to an off-site vendor facility for disassembly and repair. The motors had been surveyed at the plant and cleared for unconditional transfer. Upon opening the casing of the second motor, approximately 500 ml of water spilled out of the casing. This resulted in a tritiated water release into the public domain with potential for exposure to members of the public. A licensee engineer who was present to witness the motor disassembly stopped the work, cordoned off the area and contacted the licensee’s Radiation Protection staff who cleaned up the water and checked areas adjacent to the spill location for contamination; none was found. The third and fourth motors were not opened and all motors were returned to the plant. Previous practice had been to disassemble and check motors on site, at the dryer from which they were removed. Surveys for the unconditional transfer permit included indirect, direct and gamma surveys of the exterior of the motors, however both the surveyor and verifier who approved the shipment believed that the motor had previously been fully disassembled and decontaminated as per past practice. This event is an example of a small radioactive release;

(b) Routine water samples collected from a storm drain outside the dyke area of downgraded heavy water storage tanks showed tritium activity. Since no liquid radioactive waste is discharged through the storm drain, this was unusual and called for an investigation. The investigation revealed that prior to the event, a design provision used to transfer the liquid radioactive waste from mobile tanker to the downgraded heavy water storage tanks did not function due to problem with the transfer pump. Consequently, the operator independently took a decision to resort to an alternate method. He drained the liquid radioactive waste from the mobile tanker to a sump in the dyke area and then transferred it to the downgraded heavy water storage tanks using sump dewatering pump. The same transfer operation was repeated on two subsequent occasions. The alternate method used by the operator was not appropriate since the inventory of about 100 litres of liquid radioactive waste (equivalent to the sump low level) remained in the sump after the transfer operations. Following the operations, the sump got flooded with rainwater and overflowed leading to spread of the tritiated water on the dyke floor. An isolation valve in the dyke floor drain line was not fully closed and its downstream blind flange was passing due to deteriorated gasket condition. As a result, the tritiated water leaked out from the dyke floor area and found way to a nearby storm drain. The event resulted in release of 370 GBq (3.7E+11Bq) of tritium to the environment through an unauthorized route;

(c) During cooldown and depressurization of the primary heat transfer system for an overhaul at the NPP, the pressurizer drain valve was manually opened by a local operator, resulting in actuating the leak detection alarm in the containment. The results of the investigation verified that the leakage occurred because a wrong valve was opened since the local operator had mistaken one valve for another. It was determined that the operator's error was caused by insufficient application of the human error prevention tools, inadequate management of the valve tags and inadequate management of the locking valves. It was estimated that a total of 4,000 kg of the heavy water leaked into the containment building, and the amount of vapour leakage to the environment was about 260 kg. The amount of tritium released to the environment was about 0.055 % of the annual release limit. There was no adverse effect on the plant safety functions due to this incident. Also, there was neither undue radiation exposure to the workers nor a release of radioactive materials to the environment exceeding regulatory limits. This is an example of a large release.

Category 1.2: Exposure to radiation that exceeds prescribed dose limits for members of the public

An event resulting in exposure to radiation that exceeds prescribed dose limits for members of the public is the consequence of a serious breakdown of the barriers protecting the public. Therefore, all such events are expected to be reported to the IRS.[2]

Category 1.3: Unanticipated exposure to radiation for site personnel

The protection of plant personnel is an important objective for safe plant operation. Events dealing with unanticipated exposure of plant personnel are usually due to degradation of protection equipment and/or deficiencies in operational controls. This category comprises events addressing serious weaknesses (actual or potential) in plant operational controls and safety barriers.

Examples:

(a) In the refuelling shutdown of an NPP, during rod cluster control assembly (RCCA) manoeuvrability testing, the operator observed that the RCCA in the centre of the core had jammed. After the removal of the vessel head, the operator found and extracted a metal ring that was interfering with the movement of the RCCA. On the event date, when dealing with this deviation, the operator removed the central control rod drive mechanism, then cut out and extracted the associated thermal sleeve and resurfaced the adaptor. This was the first time that the activity to remove and replace a thermal sleeve had been carried out at the NPP. The operator therefore had no operating experience feedback to refer to in relation to the intervention. He wrongly assumed that drafted risk analysis of the activity to replace a control rod drive mechanism provided all relevant parameters for the planned intervention. Contamination was detected on five workers during radiation monitoring. The level of internal exposure was lower than 0.1 mSv, below the dosimetry assessment threshold and below the reporting threshold;

(b) During a shutdown, moderator was being drained to the moderator drain tank with the drain tank outlet valve (OV) closed as per the procedure. After two hours, dripping leakage started from the valve bonnet of the OV. The bonnet gasket was re-tightened but the leak did not stop but rather increased. Actions were taken to transfer the drain tank content back into the

[2] Unanticipated exposure to the public which does not exceed the limits is in category 1.1.

calandria. The OV was manually opened and in that moment a moderator leak started in the form of a stream from valve bonnet gasket of the OV. The closing of the valve needed some time to ensure proper seating of the valve. The boiler room purging was terminated, and the spilled moderator water collected. It is estimated that about 1600 kg of moderator water was collected from the reactor building floor. The neoprene bonnet gasket of the OV was replaced as per the preventive maintenance programme and the root cause analysis revealed the lack of procedure adherence in the valve maintenance. The collective exposure during the valve isolation and moderator water recovery was 280 person-Sv internal dose. The four affected individuals exceeded the annual regulatory dose limit (20 mSv);

(c) During a refuelling outage at an NPP, the licensee scheduled the replacement of two reels in the hoisting structure of the refuelling machine. Upon completion of the activity, significant contamination was detected around the head of one of the workers. He was immediately decontaminated. The equivalent dose to the skin was conservatively estimated to be between 200 and 500 mSv. Since the work area was cramped, the worker probably became contaminated during the activity by brushing against the rubber hose that is part of the system for one of the reels. This hose is normally in contact with primary circuit water. At the start of the job, the workers performed dose rate measurements and had available a map of the ambient radiation levels. No contamination greater than 40 Bq/cm^2 had been recorded.

CATEGORY 2: DEGRADATION OF BARRIERS AND SAFETY RELATED SYSTEMS

Safe operation of nuclear reactors is assured by maintaining three fundamental safety functions. They are:

- Control of reactivity;
- Cooling of radioactive material;
- Confinement of radioactive material.

These safety functions are ensured by safety systems which are usually provided with redundancy. The availability of the safety systems is ensured by extensive surveillance programmes as specified in the technical specifications of any given reactor. This category is intended to include events and issues where actual or potential serious degradation has occurred in the systems which are designed to maintain the availability of any of the above safety functions.

Category 2.1: Fuel cladding failure

The fuel cladding is the first barrier that prevents the escape of fission products to the environment. During the normal operation of a reactor, the potential exists for leaks to develop in the cladding of a few fuel elements, despite the care with which the fuel and cladding are fabricated and operated. Limited anticipated leaks which do not prevent continued operation of the plant are in themselves not reportable. However, fuel cladding leaks caused by unexpected factors and other unexpected fuel failure mechanisms need to be reported, especially when generic implications ensue. Reportable fuel failures are not limited to reactor power operation. Events that occur during fuel handling operations (i.e. refuelling, including handling operations in the reactor or in a storage pool), that result in an actual or potential loss of fuel cladding integrity and give rise to important lessons learned are also to be included in this category.

Examples:

(a) Gradual increases of the activity level in the reactor coolant water and in the off-gas system of a boiling water reactor were noticed, which remained below specified limits. Visual inspection during the following refuelling outage of four suspect fuel assemblies, which had been symmetrically positioned in the core, revealed severe cladding damage in the corner fuel rods. The observed damaged was caused by local fuel overpower, resulting in 'dryout' during full power steady state operation. This local cladding overheating was attributed to excessive fuel channel bow of diagonally located neighbouring fuel assemblies of high burnup with reused channels, creating very wide water gaps between adjacent assemblies. Lessons learned from this event are that greater caution needs to be exercised for the reuse of fuel channels and that the treatment of channel bow and gap width variations in core calculations needed improvement. As an immediate consequence of this event, higher values for dryout margins have been defined by the operating organization and the regulatory body has ordered all operating organizations of boiling water reactors to apply, as a function of burnup, additional penalties in calculation of critical power ratios;

(b) Problems during an infrequent maintenance activity on the pressure and calandria tubes, necessitated a non-standard defueling of a fuel channel of a pressurized heavy water reactor in the shutdown state. When transferring fuel from the fuel handling machine to the spent fuel bay, a fuel bundle broke up and a fuel rod fractured into two pieces. Although no radiation overexposures resulted from the event and its clean-up, the potential for more severe fuel damage was high. The event was attributed to a poorly planned, organized, and rehearsed procedure for a fuel transfer operation. The event highlights the need to carefully rehearse infrequent manoeuvres involving used reactor fuel;

(c) During the transfer of fuel assemblies from the reactor to the spent fuel pool, a 50 cm long unknown object was found at the bottom of fuel transfer channel. Later it was determined that this was a segment of a fuel rod. Visual inspection of all fuel assemblies unloaded from reactor core showed open defects in the cladding of 8 fuel rods in 3 fuel assemblies. These 8 fuel rods were all broken with some parts of cladding missing. The primary cause of fuel rods damage was determined to be baffle jetting which can occur at these fuel assembly locations at the core baffle plate. Open fuel cladding defects were predicted based on coolant activity data, two months after the previous refuelling outage. The release of fission products into primary coolant from the damaged fuel occurred during the NPP operation. In the reactor core there were three other leaking fuel assemblies with cladding defects at one fuel rod per fuel assembly. One of these rods could be broken and the other two have only tight fuel cladding defects. The attributed cause for these fuel assemblies is grid-to-rod fretting. Debris fretting is another possible cause;

(d) Routine visual inspection of previously selected fuel assemblies revealed anomalies in the upper part of fuel rods. Already during the unloading of the reactor core, an anomalously large number of particles had been observed on fuel assemblies and on the upper core baffle plate. The analysis of these particles clearly pointed to peeled off zirconium oxide. Further investigations showed the observed anomalies (oxide spalling, discolouration, and high oxide layer thickness) resulted from corrosion of fuel rod claddings significantly exceeding the extent known from operational experience with this type of material. All findings were located at the upper end of the active zone and/or at the upper fuel rod plenum. Only the fuel assemblies of one delivery are affected. The corrosion findings cannot be explained by the present understanding of thermally driven surface corrosion of zirconium alloys with respect to both their extent and their axial position. The potential safety-related importance lies in the fact that the causes for the excessive corrosion are unknown. An extensive cause analysis has been initiated by the fuel assembly manufacturer, which covered different steps in fuel manufacturing and operation (ingot production, cladding production, filling of the claddings with uranium pellets, fuel element assembling and use of the fuel assemblies in the reactor). In order to identify comparable corrosion phenomena in other PWRs and to assess their potential consequences for the safety case, it was recommended to inspect all fuel assemblies with this type of cladding material scheduled for reuse. The current safety requirements related to use of fuel were challenged concerning the fuel rod behaviour during normal operation and accidents.

Category 2.2: Degradation of primary coolant pressure boundary, main steam or feedwater line, or other high energy systems

The reactor vessel and the reactor coolant system, including all the connected equipment (pumps, valves, steam generators, branch pipes up to isolation valves) that are exposed to reactor pressure, form a second barrier to the escape of fission products. They are required to be designed, manufactured and tested to meet the highest standards of quality and reliability. Reportable degradations include significant welding and material defects, cracks and through-wall failures in vessel, pipes or components and loss of coolant events, including reactor coolant system leakage exceeding technical specification limits. Rapid temperature and pressure transients exceeding authorized limits and jeopardizing the integrity of the reactor coolant pressure boundary need also to be reported if important lessons are to be learned from these events.

Other high energy systems include systems such as steam generator blowdown, prior to the heat exchangers, letdown lines prior to heat exchange and pressure reduction, main turbine electro-hydraulic control systems high pressure fluid, auxiliary steam systems, etc.

Examples:

(a) An in-service inspection of the recirculation lines of a boiling water reactor revealed significant intergranular stress corrosion cracking in the heat-affected zones of an unexpectedly large number of welds, whereas in previous inspections nearly half of the number of welds had been inspected and only a few minor cracks had been detected. The cracks were considered to be a severe degradation of the reactor coolant pressure boundary. The nature of the cracks led to concerns about the applicability of the 'leak before break' principle and the sensitivity of the existing leak detection means. An additional lesson learned from this event was that inappropriate criteria for the selection of welds had originally been applied in the in-service inspection programme;

(b) During a cleaning operation of the secondary side of a steam generator tube sheet of a pressurized water reactor, a downward displacement of the steam generator (SG) tube bundle wrapper was revealed. Subsequent inspection showed the failure of 6 support blocks welded to this wrapper. This equipment failure was unexpected. It was due to differences in thermal expansion of SG internal components during temperature transients. The lessons learned are that pressurized water reactor SGs may be subjected to thermally induced mechanical loads that were not taken into account in the original design. These loads could result in a severe damage to SG internals, jeopardizing the integrity of SG tubes, which are part of the reactor coolant pressure boundary. In addition to repair work to the affected SG, a special monitoring programme has been introduced on a large number of plants with SGs of the same or similar design. This event and other initiated a global review of the design of SG internal components;

(c) The detection of a deviation in the chemical composition of the central area of the head and bottom head vessel of a new reactor led the regulatory body to ask the manufacturer and the licensee to draw the entire feedback of this event. The investigations carried out concluded that some channel heads of steam generators could also be affected by a high carbon concentration area. Such carbon content may lead to local degradation of certain mechanical properties. The statement concerns channel heads manufactured by several companies. The assessment of the impact of this phenomenon on the safety demonstration involves a thorough analysis by the operating organization to verify there are no unacceptable consequences for reactors. This demonstration was carried out reactor-by-reactor with regard to fast fracture analysis. In addition, the operating organization has undertaken a substantial programme for characterization of sacrificial parts to reinforce the conclusions of its analysis. The main lessons learned are that heterogeneities induced by forging might cause a generic concern and affect the mechanical properties of components under the break preclusion hypothesis. Operating organizations need to monitor the manufacture of components that are designed so that carbon segregations are kept within specified limits, thereby ensuring a limited impact on mechanical properties;

(d) A Pressurized Heavy Water Reactor (PHWR) experienced a leak from a pressure tube. The magnitude of the leak was such that the reactor underwent automatic shutdown. There was no fuel failure due to the event. The Annulus Gas Monitoring System (AGMS) is provided for detection of coolant leak from pressure tubes. Investigations carried out after the event revealed that the carbon dioxide gas used in AGMS had a small amount of unlisted impurity of hydrocarbons, mainly ethylene. This resulted in shallow localised corrosion on the outer surface of all the pressure tubes in the reactor. The AGMS gas mixture in combination with ethylene and radiation also led to generation of hydrogen which accumulated in the pressure tubes. High hydrogen content in an affected pressure tube eventually led to crack initiation and its propagation through delayed hydride cracking to a critical size crack. The event highlights that the specifications for gases and chemicals to be used in NPPs (particularly gases and chemicals in contact with core components) need to be finalized with due care, taking into account their sources, possible impurities and operating environment conditions. Also, a comprehensive safety evaluation is necessary whenever there is any change in the source of gases or chemicals as this may introduce unlisted impurities that might affect plant systems.

Category 2.3: Degradation of containment function or integrity

Most reactors are enclosed by structures to confine radioactivity if there is a release from the reactor primary coolant system. This confinement function, which may include primary and secondary containment, is the ultimate barrier to prevent the release of radioactive material to the environment. Containment structures are required to withstand pressure and temperatures resulting from design basis accidents without exceeding the design leakage rate. They include passive structures and components (e.g. a steel pressure vessel, leak tight containment penetrations, pressure suppression systems) as well as active components (e.g. containment isolation valves, containment spray and cooling systems, ventilation systems). The integrity of the containment may also be required under shutdown conditions when performing fuel handling operations in the reactor building or when cooling of the fuel could be threatened. Degradation of related systems, structures, and components resulting in actual or potential loss of this safety function are covered by this reporting category.

Examples:

(a) As a result of an investigation of alarms due to small differences between redundant containment pressure measurements, a differential pressure transmission pipe, penetrating the primary containment of a boiling water reactor was found to be ruptured. The rupture, which was located between the containment penetration and the first isolation valve outside the containment, degraded the confinement function. In addition, the defect affected the operability of the containment pressure measurement of the reactor protection system. The event highlights the specific difficulties in detecting leakage of pipes penetrating the containment vessel, which are not pressurized during normal operation but are leak tight during accidents;

(b) During inspection of a steel ice condenser containment vessel of a pressurized water reactor before an integrated leak rate test, significant coating damage and base metal corrosion on the outer surface of the steel shell was discovered. Subsequent investigations also revealed corrosion in areas below the level of the annulus floor. The degradation of the steel shell was caused by boric acid coolant which had leaked from instrument line compression fittings. Existing drains were widely separated and the floor was not sufficiently graded to prevent condensate pooling. Corrosion occurred at locations considered not susceptible to corrosion and inaccessible for inspection and maintenance. Other units were affected by the same problem and were alerted of this potentially generic problem by the regulatory body;

(c) When performing a complete test of the containment spray system of a pressurized water reactor with actual delivery of water to the spray nozzles, it was found that the capacity of the system including spray efficiency was much lower than the design value used in the safety analysis report. The cause was an excessive pressure loss on all heat exchangers of the spray system. The potential consequence of this deficiency, which had existed for a long time, was that relief valves in the room ceiling could open in the event of a design basis loss of coolant accident and that part of the accident steam–air mixture could be directly released into the environment. The findings emphasized the need to carry out complete tests of safety systems in full accident configuration to verify design characteristics. The partial tests which had been performed during plant commissioning did not allow for the discovery of the observed design deficiencies;

(d) The reactor has a containment with a gastight liner designed to withstand high pressure in the case of a steam release in the containment. In connection with restarting the unit after an outage, a containment leak test was carried out with approved results. Upon inspection of the containment, a leakage was found in the upper part of the liner. Three small pitting holes were visible. The investigation showed that water had collected between the liner and the concrete wall of the containment. To create a space between the liner and the wall, insulation was attached to the liner when concrete casting the containment. Leaking pools above the containment were what caused the water to collect. The insulation, combined with intermittent leakage, provided the preconditions for corrosion of the liner. The incident demonstrates the need for well performed preventive maintenance. The incident could have been avoided if maintenance of building structures had been performed in accordance with the developed programme. Preventive maintenance of building structures, including testing procedures, has not been adequately performed for decades. Problems linked to liners are not unknown. Other operating experiences with liners already existed and actions were taken with respect to liner testing at other NPPs.;

(e) During an inspection inside the primary containment a member of staff of the NPP discovered two fire extinguishers that were directly mounted on the inner containment wall with screws. It was observed that 6 holes had been drilled through the primary containment wall from the inside, thereby connecting the inner containment with the annulus. Each hole had a diameter of approximately 5.5 mm. In four holes, there were self-tapping screws to fix the mounting, in one hole, a broken screw was present and one hole was open. All six holes remained undiscovered for almost 6 years, because they were hidden behind the mountings of the 2 fire extinguishers. Various safety analyses were performed to evaluate the impact of the event on nuclear safety. Analyses performed showed that the structural integrity of the primary containment in case of a LOCA was not affected by the holes. While the technical impact of the holes on safety was low, the event shows serious organizational deficiencies in the maintenance process, surveillance of external contractors and quality assurance which could have led to more serious consequences.

Category 2.4: Degradation of systems required to control reactivity

Various systems are provided for reactivity control to bring a reactor from cold shutdown to full power conditions and vice-versa, to compensate for fuel burnup, and to shut a reactor down if the limiting conditions of operation are exceeded. They include movable control rods, neutron poison injection systems, burnable poisons, the moderator dump system of a pressurized heavy water reactor, the recirculation system of a boiling water reactor, etc. Reactivity control may further be affected by failing administrative and operational controls (e.g. undetected errors in core loading or fuel manufacturing, uncontrolled boron dilution of reactor coolant or boron injection systems). Degradation of such systems and controls may lead to reactivity excursions (e.g. rod ejection, boron dilution), local power distribution anomalies and failures to shut down the reactor as designed. Observed degradation or failures of such systems and controls may have generic implications and are expected to be reported if important lessons are to be learned.

Examples:

(a) During initiation of a manual scram, as part of a normal shutdown routine of a boiling water reactor, 75 of the 185 control rods failed to fully insert in the reactor, caused by water accumulation in the scram discharge volume. The event highlighted significant design weaknesses of the scram system, which were not recognized during the initial design evaluation and surveillance test. Corrective actions included design modifications to increase the reliability of the reactor scram system. These modifications involved improvement of the related reactor protection functions, reactor protection instrumentation, and related venting and drain piping arrangements. In addition, emergency operating procedures and operator training were provided for complete and partial scram failure conditions;

(b) The insertion into a pressurized heavy water reactor core of an adjuster rod with a higher reactivity value after a design modification, triggered unexpected neutron flux oscillations. Both the reactor control systems and the operator actions were unsuccessful in controlling oscillations, which continued for 50 hours until the reactor was shut down. Investigation of the safety implications revealed that the large flux tilts experienced had the potential for degrading the reactor neutron overpower protection and thus increasing the risk of fuel and pressure tube damage in the event of inability to control bulk reactor power. This event revealed limitations of the reactor control systems as well as weaknesses in the plant procedures and operator training on recognition and handling of severe flux tilts. It demonstrated the importance of adequate design review of modifications affecting reactivity control and the need to emphasize to the station staff the significance of conservative response to unusual reactor transients;

(c) During the commissioning test of a pressurized heavy water reactor, neutron flux measurements showed the existence of a significant bottom to top flux tilt. The investigation revealed that depleted fuel with two different depletion values had been inadvertently loaded into the reactor core. Plant staff were unaware of the fact that fuel depletion value had been changed between successive fuel batches. Reactor safety analysis and reactor operation computer codes utilize in-core fuel isotopic values to calculate setpoints and detector calibration factors for neutron overpower protection. Further investigation revealed that, for more than three years, depleted fuel with isotopic values different from those postulated had been loaded into reactor cores of other NPPs of the same utility. The loading errors had resulted in the reactor protective system being impaired in a few cases;

(d) The PHWR reactor has two independent shutdown systems. Shutdown System 1 (SDS-1) drops shut-off rods into the reactor core under gravity while Shutdown System 2 (SDS-2) injects liquid poison (gadolinium nitrate solution) in the moderator. The bottom of each poison tank has an injection tube which penetrates the calandria. On actuation of SDS-2, the quick opening valves open and high-pressure helium pushes the gadolinium nitrate solution into moderator. Each poison tank has a plastic ball. The ball is lighter than the liquid poison and floats near the top of the tank when it is completely filled. During poison injection, the ball moves down and seals the bottom seat of the tank thereby preventing helium ingress into the calandria. During manual actuation of SDS-2 (as a prerequisite for containment related tests) and injection of poison through the shutdown tanks, the balls in two tanks (out of 6) prematurely sealed the tanks and allowed only partial injection of poison. It was found that the plastic balls in the tank had an uneven diameter profile that had caused them to get stuck in the tanks. The observed dimensional variations in the ball diameter could have affected the functionality of all the other injection tanks and of those at another NPP;

(e) During an outage, operating personnel noticed that a control rod was tilting somewhat. The day after, during the lifting of the control rod, it was observed that the control rod shaft extender was broken. The operating organization examined other control rods. Cracks were identified in several control rod extenders. During the examination of control rod shafts at another unit of similar design, cracks and a broken control rod extender were identified. Subsequent to exhaustive examinations, testing, and analyses it was concluded that thermal fatigue was the main underlying contributing cause of the identified cracks and of the broken control rod extenders. The thermal fatigue occurred as a result of thermal oscillations caused by the mixing of the crud flow with relatively cold water and significantly warmer primary coolant in an area of the control rod extenders exhibiting stress concentrations and/or geometrical variations. These thermal fluctuations have occurred at the mixing zone of the crud flow (60°C) and the primary coolant by-pass flow (280°C);

(f) In accordance with the Limiting Conditions for Operation (LCO), the boron concentration in every flooded part of the reactor coolant system and the refuelling canal needs to be homogeneous and sufficiently high to ensure subcriticality. During refuelling, certain valves are expected to be locked in the closed position to prevent uncontrolled boron dilution (i.e. to keep the flow paths from non-borated water sources closed). A low-level alarm in the Spent Fuel Pool (SFP) appeared in the Main Control Room (MCR). The MCR operators decided to add demineralized water in the refuelling cavity through the coolant make-up water tank. In order to do that, they had to open a valve, which was locked closed, according to the Technical Specification. When the SFP level was restored they returned the valve to the closed the position. During the MCR shift turnover the off-going shift informed the in-coming shift about the low-level alarm and the water refilling but did not mention the decrease in the boron concentration, though it was written in the MCR log. After some communication misunderstandings between the MCR and the Chemistry Department, the MCR staff asked the Chemistry Department to take a new sample of the boron concentration after noticing that the boron concentration was lower than the previous value. The fuel loading continued after restoring the concentration of the boric acid solution.

Category 2.5: Degradation of systems required to ensure primary coolant inventory and core cooling

Systems are provided which ensure in normal and transient NPP operation sufficient means to remove core power and residual heat (e.g. primary coolant pumps, main and auxiliary feedwater systems, residual heat removal systems in shutdown conditions, pressure relief and safety valves). Failure to remove core power or residual heat can result in uncontrolled primary coolant and fuel temperature increases putting fuel integrity at risk. Unwanted primary coolant system pressure increases may also challenge or jeopardize the integrity of pressure barriers.

Despite the precautions taken in design, failures of the primary coolant pressure boundary can occur which warrant the existence of redundant or diverse emergency core cooling systems (such as high pressure and low pressure injection systems and core spray systems), providing core inventory supply and cooling in the event of loss of coolant accidents. Partial or total failure of such systems may result in large fuel cladding failure, fuel meltdown and significant releases of fission products.

Actual failures of these systems or the existence of significant potential latent failures, e.g. due to shortcomings in inspection and testing programmes, may be of concern to the whole nuclear community and are expected to be reported under this reporting category.

Examples:

(a) In order to reduce the oxygen concentration of the auxiliary feedwater (AFW) tank of a pressurized water reactor in a recirculation mode, the AFW gas stripping system was put into service. This caused the AFW temperature to increase. Because of the delayed operator action on the temperature alarms, the AFW temperature exceeded technical specification limits. Excessive heat up of water in the common AFW storage tank implies a common mode failure during prolonged use of the AFW pumps. The risk was increased by improper temperature measurement locations in the storage tank, which did not allow the detection of the formation of hot water layers. This event revealed several design weaknesses of a non-safety-related support system which could result in the inoperability of an safety system;

(b) Several events have addressed the potential for over pressurization of emergency core cooling systems in boiling water reactors because of failure in the open position of a testable air-operated isolation check valve in the injection line due to maintenance errors. The events were considered to be precursors to an intersystem loss of coolant accident (LOCA) between the reactor coolant system and the emergency core cooling system. The events indicate that the likelihood of an interfacing LOCA is significantly higher than previously assessed. The events also highlighted the need to reduce human errors in maintenance and surveillance testing activities by the improvement and/or standardization of maintenance and surveillance procedures and upgrading of the training and qualification of plant personnel;

(c) During a periodic test of the operation of fire detectors in the electrical building, the water was being purged to the NPP sewer system provided for this purpose. Several alarms were activated in the control room, concerning the safety injection system. Operating personnel observed water leakage onto the electrical cell of one of the low-pressure safety injection pumps, which caused the pump to be unavailable for approximately 31 hours. This leakage was coming from the plant sewer system pipe that passes through the room containing the electrical cell of the pump. This pipe was corroded and had been breached. The main cause of this event was insufficient monitoring of pipes in the plant sewer system. This system, which has no safety classification, was not included in any preventive maintenance programme. Visual inspections of these pipes had been carried out as part of a review focusing on the risk of internal flooding in the electrical building. Defects had been observed and an action plan drawn up, but no follow-up arrangements were made for the NPP. Furthermore, the defect that caused the event had not been identified, as it was on the top part of a pipe that was at a height of 2.5 m. In addition, owing to the severe corrosion of many other pipes of the plant sewer system, water might have come into contact with equipment important to safety, rendering it unavailable. Following this event, approximately 115 linear metres of piping were replaced;

(d) A reactor tripped from 100% power because of an electrical fault on the vital bus of one train. The auxiliary feedwater system actuated as expected, and the corresponding emergency diesel generator started but did not load as designed, due to the lockout of the bus. The bus remained de-energized and the reactor was stabilized in hot shutdown. The high head safety injection (HHSI) pump was unable to be powered from the bus. The safety function is achieved by operation of two of the four pumps which are shared by both units. At this time, both HHSI pumps on the twin unit were out of service for maintenance, resulting in a loss of the safety injection safety function for approximately 2.5 hours on both units. The loss of the bus was caused by an electrical fault created by a conductive foreign material that had entered the current-limiting reactor cubicle that bridged an air gap between an uninsulated bus bar and the cubicle wall. The foreign material was a carbon fibre mesh used to reinforce an installation taking place in the switchgear room.

Category 2.6: Degradation of essential support systems

Active safety systems depend on the operability of essential support systems such as AC/DC power (including emergency diesel generator systems and batteries), service water, instrument air and heating, ventilation and air conditioning systems. The dependence on such systems, if inadequately designed or operated, makes active safety systems particularly vulnerable to common mode failures. Weaknesses in design, maintenance and surveillance of such systems may lead to unexpected failures of the safety systems requiring operation of these support systems. Degradation of such support systems may be of generic interest and are expected to be reported if important lessons can be learned. Degradation of fire protection systems, which might lead to spreading of fires affecting separated redundant trains of safety systems, are also to be covered in this category.

Examples:

(a) Several events involved the inoperability of multiple emergency diesel generators (EDG) due to the degradation of the fuel oil delivery system by deterioration of the quality of fuel oil stored at the site. The events highlighted that the surveillance programme implemented at the NPPs was insufficient to timely detect the degrading fuel oil quality and the resulting deficiencies in the fuel oil delivery system. In addition, the industry-accepted standards for fuel oil quality did not adequately cover the question of particulate contamination in stored fuel;

(b) The NPP was in cold shutdown for refuelling, when a loss of power supply occurred to a 125 V DC busbar being supplied only by its rectifier. At that very time, a discharge test of the battery was being performed in which, the battery was disconnected from the busbar. During attempts to recover the situation, a total loss of power for the control system occurred for about 15 minutes, although core cooling remained uninterrupted. The event sequence revealed difficulties with diagnosing the situation and bringing the plant back to normal in the unusual electrical power supply conditions. Lessons learned from this event include the importance of batteries in ensuring a continuous power supply, especially in connection with maintenance and periodic testing practices. Also, a need to improve methods and instructions for correcting a partial loss of electrical sources was identified;

(c) Due to the isolation of a shutdown heat exchanger outlet valve, a total loss of shutdown cooling capability occurred in a pressurized water reactor which was in mid-loop operation. The outlet valve malfunction was caused by water accumulation in its control air system. The primary coolant temperature rose about 40°C reaching a maximum temperature of 90°C over a 90 minute period, while plant staff diagnosed and corrected the problem. It was also recognized when actions were taken to improve the reliability of air operated valves in the shutdown cooling system. The plant event procedures were primarily directed to address the inoperability of pumps and heat exchanger, rather than flow path unavailability. This event and subsequent analysis revealed the susceptibility of the shutdown cooling system of the affected plant to single failures, even when both trains of the residual heat removal system are operable;

(d) While performing a scheduled surveillance test of an emergency diesel generator (EDG), the EDG tripped due to high temperatures in the engine cooling water system. Subsequent investigation revealed water in the instrument air system, which could have resulted in a common mode failure of the redundant EDG, as well as of other safety-related components of the plant. The water intrusion was due to inoperable check valves in an interconnection between the instrument air system and the fire protection system, which had been modified without adequate evaluation of the safety implications. The tests carried out after implementation of the modification were inappropriate to address the problem. A walkdown of the entire plant instrument air system was made to ensure that all interconnections to the fire protection system were either isolated or removed;

(e) During maintenance and testing of relays on the excitation system of the main generator, the 400 kV unit breaker received an unintentional signal to open. However, one phase of the 400 kV breaker did not open. Due to the main and auxiliary transformers' configurations (Y/D and D/Y, respectively), a substantial line-voltage (>65%) was still available for two out of the three line-voltages. The relay protection, engineered to sense an average undervoltage below 65 % during more than 1 second on the safety-grade auxiliary power bus bars, was not designed to trip in this situation. Consequently, none of the safety-grade auxiliary power bus bars were disconnected, nor did the EDGs start. The phase failure fed the unit and the components phase unbalance relay protection disconnected 146 components (safety and non-safety). Also, some non-safety motors were overheated and failed. Components vital for heat removal from the spent fuel pool and for cooling the EDGs tripped. After approximately 16 min., one EDG train was manually started and the breaker from the 10 kV ordinary grid and 10 kV EDG backup grid was manually opened. Cooling systems for EDG and residual heat removal systems, were available after approximately 35 min. The faulty unit breaker in the 400 kV switchyard was reconnected after 43 min. Local resetting of other objects was finished after approx. 90 min.

CATEGORY 3: DEFICIENCIES IN DESIGN, CONSTRUCTION (INCLUDING MANUFACTURING) INSTALLATION AND COMMISSIONING, OPERATION (INCLUDING MAINTENANCE AND SURVEILLANCE) SAFETY MANAGEMENT/QUALITY ASSURANCE SYSTEM, SAFETY EVALUATION AND DECOMMISSIONING

High standards in design, construction and operation, complemented by in-depth safety evaluation, ensure the overall safety of NPPs. The most important means to maintain safety during the lifetime of a plant are good operational practices to prevent failures, safety management/quality assurance to verify the achievement of the design and operational intent as well as a comprehensive surveillance programme to detect and correct degradations or failures in time. Deficiencies related to these key elements of plant safety highlighting important lessons are expected to be reported in this category.

Category 3.1: Deficiencies in design

The main objective of plant and equipment design is to ensure overall plant safety with sufficient margins. Deficiencies in the design could result in loss of a safety function, loss of safety system or unexpected event sequences. Further, design deficiencies may cause common mode failures that affect the safety of the NPP. All such cases including material compatibility, degradation due to environmental or operating condition, computational errors, etc. are expected to be reported under this category.

Examples:

(a) A test on an emergency 24 V DC battery detected that in certain conditions postulated in the design basis of the plant, the reactor protection cabinets received less voltage than the minimum voltage required by design. In those conditions, it was not guaranteed that, in the event of loss of off-site power, the emergency batteries would be able to start up the emergency diesel generators. During the design of the plant, the voltage drop due to elements such as cables, fuses, breakers, etc. had not been considered. In addition, the batteries of the system were undersized;

(b) Reactivity calculations for the spent fuel pool were performed using different computer codes that showed relatively high deviations in results. The causes of the potential errors in these calculations were approximations used in the calculations that were not appropriate in the presence of a highly absorbing material. The utilities evaluated spent fuel storage rack design changes, additional criticality analyses, and changes to the plant technical specifications to allow use of their fuel storage racks. The safety consequence of this event is a potential uncontrolled criticality event in the spent fuel pools;

(c) In an NPP, there was a brief loss of communication of signals in the control processing unit (CPU) of the main turbine digital electrohydraulic control (DEHC) system. This communication process is based in field nodes that supply input and output data to the DEHC system. The communications node, in standby of CPU “A”, tried to take control but generated erroneous signals in the process, therefore, was not capable of performing this action. CPU “B” did not identify the erroneous signals in its communications nodes; therefore, it lost the field signals without transferring control to CPU “A”. Because of this the CPU “B” identified as failed the vacuum sensors of Condenser “A” and generated the main turbine trip, almost simultaneously with the occurrence of the first failure. This resulted in a reactor scram. The event identified vulnerabilities in the redundancy design of DEHC system as well as need to update current knowledge in the performance of digital equipment;

(d) The NPP was on low power mode, with power at 0.95%. All control rods were in manual mode, and pressurizer pressure and water level were in automatic control mode. The reactor operator verified relevant functions and identified that the main control room operator station was not available. The operator implemented the procedure for handling operator workstation unavailability, while notifying instrumentation and control personnel for inspection and recovery. The operator selected the parameter to check the set point but configured the main control room operator station to the wrong parameter so the system diagnosed the wrong command, which might have caused incorrect equipment operation. Consequently, the main control room operator station determined the wrong data configuration and automatically stopped the control function server. It was recognized that stopping the control function server and the subsequent incorrect equipment operation just by diagnosing a wrong parameter configuration is not an acceptable design solution. The system function that stopped the server because of the wrong data configuration was clarified with the manufacturer. During operation or maintenance, a number of barriers need to be in place to ensure proper control and accuracy of data modification;

(e) The NPP was in the shutdown mode. During the storage of historical data, instrumentation staff identified the failure of the historical data processor. After the failure was fixed and the historical data processor was put into operation, the plant computer information and control system platform halted suddenly, and the operator station in the main control room showed just a blank screen. The operating personnel switched the control mode from the plant computer information and control system to the backup panel in accordance with the procedure. The instrumentation staff restarted the plant computer information and control system platform and operating personnel executed the procedure to transfer the control mode to the backup panel of the plant computer information and control system. Switching off the plant computer information and control system platform during this mode of operation is required by the plant Technical Specification. The inoperability of the plant computer information and control system was due to the restart of the historical data processor, which is responsible for historical data processing and storage. During data retrieval the CPU load of the historical data processor may be up to 100% and subsequently it can cause shutdown of the plant computer information and control system due to an incorrect prioritization of function implementation. The storage of large amounts of data in the process may result in the CPU overloading and shutdown of the plant computer information and control system platform. In accordance with the design of the plant computer information and control system, the average load rate of the CPU of the platform is not to exceed 40% under normal conditions and 50% under busy conditions. The manufacturer performed an insufficient estimation of the actual data load in different plant modes of operation and failed to reserve enough data for calculating the load allowance, which resulted in the shutdown of the plant computer information and control system platform. The manufacturer updated software, optimized CPU utilization during historical data synchronization and reduced data processing.

Category 3.2: Deficiencies in construction (including manufacturing), installation, and commissioning

Deficiencies in construction and installation may cause significant deviations from the desired plant status. These deficiencies can occur during initial construction of the plant and during back fitting of equipment. If construction deficiencies cannot be detected by testing or maintenance, they may cause latent failures that degrade plant safety.

The results of commissioning activities demonstrate that the design requirements and intent, as stated in the safety analysis report, have been met. The results also define the initial characteristics of systems. Safe commissioning of equipment, systems and the plant is essential to the subsequent safe operation. The commissioning process covers all the activities to be performed on structures, systems and components to bring them to an operating mode. Deficiencies detected and corrected during commissioning and latent deficiencies that have led to events during operation may be reported under this category. This category may also include events that occur in the conduct of 'ITAAC' (Installation, Testing, Analyses, and Acceptance Criteria) activities for reactor construction.

Examples:

(a) The discovery of through-wall cracks in underground pipes of an essential service water system, caused by misalignment of civil structures, led the operating organization to survey the underground piping at all its NPPs. This survey revealed that all NPPs in the utility were affected to varying degrees. The degradation observed was due to poor construction quality of civil structures associated with the underground piping and inadequate supervision of civil structures and associated pipes during operation. Though indications of misalignment had been noticed at the end of construction of some plants, it had not been taken into account in underground pipe monitoring programmes. The potential risks were total loss of essential service water systems. These faults revealed the importance of monitoring the movement of civil structures in addition to regular inspections of underground pipes and structures;

(b) During a post repair liquid penetrant test of a weld, situated in a main steam line between the containment penetration and a main steam isolation valve, surface cracks were discovered. Previous pre-service and in-service inspections, using ultrasonic and gamma radiographic tests, had not detected these weld faults. Additional inspections in several plants were performed, showing similar weld defects in some units. The event highlights the difficulty to detect surface faults in certain pipe geometries by some commonly used in-service inspection techniques. In addition, quality assurance deficiencies in the manufacturing of the pipe work were highlighted. Manufacturing and inspection procedures have been modified as a result of this event;

(c) The reactor was operating at full rated power. The main control room personnel observed an alarm present, related to main turbine average speed signals (speed deviation greater than 0.02% in any speed channel/sensor). The personnel inspected the channel/sensors and while they were troubleshooting, there was an automatic scram caused by a main turbine trip. The cause of the trip was spurious oscillating signals related to the over speed protection of the main turbine, with no real changes in the speed of the turbine shaft. In order to find for causes and restore the operability of the system, measurements and testing for each sensor of the over speed protection system were made. Following these activities, the plant was restarted for operation. The same problem occurred again after two weeks. In the process of investigation, plant personnel again proceeded with testing of the system and components. The turbine case was opened to see the physical conditions of the components and the causes for the oscillations in the values detected by the over speed protection system. It was found that the cables of the over speed protection system were installed inadequately, and this was the cause of the spurious signals. The cause of the event was related to inadequate procedures and incomplete technical guidance for the maintenance and calibration of the speed sensors. A contributing factor was the inadequate supervision during routing of the cables of the speed sensor channels;

(d) During preparations for commissioning tests it was discovered that valve internals had been mixed up during piping installation. Further investigations by the licensee and vendor identified that hundreds of valves had internals that did not correspond to design specifications for the installation location in question. All valves were inspected. Valve assembly, valve type and actuator type were checked and catalogued. This catalogue was compared to design specifications for the valve location. From the comparison, four distinct cases were identified: a) valve welding head material thickness did not correspond to design specification (467 valves), b) valve safety classification or seismic classification did not correspond to design specification (86 valves), c) valve actuator did not correspond to design specification (508 valves), and d) valves that had had their valve body and valve internals delivered as separate shipment to the site (198 valves).

Category 3.3: Deficiencies in operation (including maintenance and surveillance

Safe operation and effective maintenance (including inspection and surveillance activities) are the result of qualified and well-trained NPP staff, adequate procedures and tools, and good management. Deficiencies in human performance (including licensed operating personnel, other plant personnel, and contractor personnel) may degrade the defence in depth. They may result in the degradation or loss of safety related systems or challenges the operability of safety systems, as a result of transients due to component failure or loss of operational control.

Examples:

(a) When draining steam generator tubes in plant shutdown condition, the residual heat removal (RHR) function was completely lost for 25 minutes. The event was caused by too large a decrease in the water level in the reactor coolant system, resulting in air being sucked into the RHR system and loss of both RHR pumps in operation. Subsequently, the operating personnel started a third pump without adequate diagnosis of the situation. This pump failed due to the same operating conditions. Procedures in support of steam generator tube draining operations have been improved and personnel have been better informed of the risks related to this operation. The precision of existing level measurement equipment has been improved and additional level measurement equipment based on different physical principles has been installed;

(b) During a functional test in cold shutdown on the isolation valve of a safety injection accumulator tank, 10 m^3 of demineralized water flooded into the primary circuit. Before the event, a hydrostatic test of the accumulator tank had been carried out using demineralized water. Due to incomplete draining after the test, the non-borated water remained undetected in the tank and entered the primary circuit when testing the isolation valve. The consequences of the event were limited: an overflow at the vessel head seal level occurred and a limited dilution of the reactor coolant took place. However, studies carried out using worst case scenarios showed that the risk of a non-borated water plug reaching the core could not be excluded. The hydraulic tests are now performed with borated water;

(c) A scheduled maintenance was under way at an NPP unit. The reactor was in refuelling mode and all nuclear fuel had been unloaded from the reactor vessel. While contractor staff were carrying out work without permission to install a new cable line (making holes in a cable penetration), without having the appropriate work authorization and without coordinating with plant staff, a control cable was damaged (cores were shorted) in the protection system for a reactor coolant pump and, as a result, a spurious command was generated to disconnect buses BB and BW of the normal and emergency 6-kV power supply, respectively. When the emergency 6-kV power supply bus BW was disconnected, a diesel generator automatically started up to supply power to the bus. The damage to the cable inside the cable penetration occurred as a consequence of improper laying of the cable in the penetration, in violation of construction standards and rules, at the time the unit was installed and as a consequence of unauthorized work performed by the contractor when installing a new cable line. The event is example of deficiencies in organizing maintenance;

(d) An NPP unit was operating at full power when a human error at the 380-kV switchyard resulted in the inadvertent trip of a 380 kV circuit breaker, the loss of grid connection, and automatic transition to island mode (house load operation). As expected during this transient, the steam dump valves of the turbine bypass system opened fully to divert steam directly to the condenser. However, they soon became inhibited due to insufficient condensation of steam by de-superheating water provided by the condensate pumps and subsequently only worked intermittently. Inhibition of the turbine bypass system resulted in a heat-up transient, with opening of the pressure operated relief valves on the pressurizer as well as opening of main steam safety valves and relief valves. The simultaneous opening of main steam safety valves, relief valves and steam dump valves to the condenser resulted in the actuation of the reactor protection system on low steam line pressure (reactor trip and safety injection signal initiating a start of the emergency diesel generators). The event also involved an electrical transient on the vital and non-vital busses as well as a trip of the reactor coolant pumps. This transient resulted in a significant release of steam to the atmosphere through the main steam safety valves and main steam relief valves. One of the workers was severely burned by wet steam entering the building. This event involves deficiencies in operation and highlights the importance of correctly labelling equipment, adequate configuration management and operational decision making, adequate risk management, and adequate communication.

Category 3.4: Deficiencies in safety management or quality assurance system

The safety management/quality assurance system achieves and enhances safety by bringing together requirements for managing the organization, including planned and systematic actions providing confidence that the requirements are satisfied. The management system ensures that health, environmental, security, quality and economic requirements are integrated with safety requirements. The quality assurance (QA) programme ensures that the plant is constructed and operated within the licensed conditions. Deficiencies in quality assurance may occur in the QA programme as well as in specific QA measures.

Event reports in this category are expected to highlight deficiencies that have an impact on plant safety.

Examples:

(a) A vendor delivered an argon–hydrogen mixture instead of pure hydrogen to a pressurized water reactor plant. The delivery of the wrong gas mixture was not detected on receipt and the wrong gas mixture was fed into the hydrogen supply system. The error was detected by monitoring devices in the off-gas system and the turbine-generator cooling system. The argon-41 activity in the primary coolant system increased markedly. The event was caused by poor labelling of the gas cylinder bundle. The labelling was in conformity with the related technical standards, but the label was not clearly visible. Additionally, deficiencies in the Quality Assurance (QA) measures of the vendor and in the reception inspections of the utility were revealed. The vendor and utility agreed on improvements in the labelling of the cylinder bundles. The event shows that QA deficiencies in non-safety-related systems may have effects on safety systems;

(b) During cooldown for refuelling, the primary circuit boron concentration was supposed to be maintained at 2200 ppm and was measured continuously by automatic boron titration, and intermittently by manual titration. When filling the reactor cavity with borated water from the storage tanks there was an apparent increase in measured boron concentration. The investigation revealed that the previously adjusted boron concentration of the primary circuit was only 2040 ppm. The reason was an error in the correction factor for the lower concentration of the sodium hydroxide solution used in the course of manual analysis. Since the manual analysis had also been the basis for the calibration of the automatic titrator, the latter also showed a higher value. The generic lessons learned showed that diverse measurements might not be independent if they are based on the same measurement standard. Subsequently, the regulatory body required additional precautionary measures and use of additional checks for the calculation of the correction factor;

(c) In normal operation, a leak from the vent hole of the heated junction thermocouple (HJTC) of the reactor vessel was confirmed. As access to the area of the leak for maintenance was not possible, the operating personnel reduced power and manually tripped the reactor. An investigation confirmed that the leakage was caused by a corroded vent ball which seals the HJTC vent hole. The leakage amount was low and did not necessitate entry into LCO including a reactor trip. It was estimated that 888.8 litres leaked inside the containment building. Material analysis showed that the intact HJTC ‘A’ vent ball was composed of stainless steel, but the damaged HJTC ‘B’ vent ball was made of carbon steel. It was confirmed that the HJTC vent balls were replaced during the last overhaul and the HJTC disassembling results proved that a carbon steel vent ball was incorrectly installed. This event highlighted inadequate control of purchasing management, because the plant did not request the material test report when placing the order. Furthermore, there was inadequate management of the HJTC vent ball quality class. If the vent ball had been classified in a higher quality class, the quality assurance department as another barrier would have been able to identify the inadequate material in advance;

(d) The NPP unit was in hot standby conditions, when a leak was spotted on a load shut-off valve in the chemical and volume control system (CVCS) charging line. To repair the valve, a decision was made to interrupt the plant start-up and bring it to hot shutdown conditions. Due to fact that the condensers were not yet available, the main steam relief valves had to be used to remove the heat from the primary circuit. The turbine operator decided to start the cooldown operation, using just one of the three steam generators. This resulted in a pressure difference between the steam generators, which activated a safety injection signal. The event was attributed to a lack of plant knowledge of the turbine operator, who had a wrong mental picture of the configuration of the main steam system. The safety management of the operating organization was not sufficiently robust with respect to plant cooldown operations in a specific context and to a change of planning (from plant start-up to shutdown): it relied completely on operating personnel skills, which were variable and not sufficiently supported by training, procedural guidance or adequate supervisory methods. Furthermore, operating personnel fundamentals (i.e. essential knowledge, skills, behaviours and practices that operating crews need to apply to operate the plant effectively and safely), were not respected during the event. There was no adequate pre-job briefing of the operating personnel team, procedural guidance was inadequate, monitoring of plant parameters was inadequate and supervision of control room operations was inadequate. The event illustrates the importance of adequate guidance in (operating) procedures with a level of detail, which is adapted to operating personnel team experience and performance. Furthermore, the necessity of adequate supervision and coordination of control room operator actions by the shift supervisor, as well as anticipating possible operational difficulties when a change of planning occurs, supported by appropriate risk analysis and sufficient knowledge of the management team with respect to the operating practices of different crews is highlighted.

Category 3.5: Deficiencies in the safety evaluation

The safety evaluation (also commonly referred to as safety assessment) covers the analysis of postulated operating conditions, all design basis events and the related safety measures. This category addresses deficiencies in the safety evaluation of systems, event sequences and operating conditions considered in the design analysis, as well as deficiencies in the original scope of the safety evaluation (i.e., unanalysed event sequences or conditions). Due to these deficiencies, unexpected situations may occur which significantly compromise plant safety. Related event reports are expected to provide information about the deficiencies identified and the responses of operating personnel to control the event, and lessons learned to prevent recurrence. Typical examples in this category are environmental conditions being inadequately taken into account, unforeseen system interactions, non-conservative calculations, and deficiencies in the safety evaluation of maintenance procedures.

Examples:

(a) Severe grid disturbances as a result of a blizzard led to the reactor shutdown. The weather conditions continued to deteriorate and a combination of a storm-force winds and a high tide carried waves over the sea wall, flooding the cooling water (CW) pump building. All sea water cooling was lost and the six CW pump motors were damaged. As the flooding of the CW pump building occurred after the shutdown, plant safety was not affected. The original design basis for the sea wall around the site was only intended to contain the general rise of sea tide. The assessment of this event showed that the conditions during the event were extreme but not unpredictable. To reduce the possibility of future flooding, the sea wall was increased in height and further water barriers were placed in the vicinity of the (CW) pump building;

(b) During surveillance on the reactor core isolation cooling initiation logic for a boiling water reactor, a wrong valve was opened which caused the recirculation pumps to trip. During the resulting transient, cooler feedwater was supplied to the reactor. This caused excessive neutron flux oscillations resulting in power surges. The reactor eventually scrammed on high neutron flux. The investigation revealed that stability analysis methods previously used had not been conservative. The transient reactor behaviour that was observed during this event was not predicted. In addition, operating personnel were not trained for this type of event. Further investigations led to a more in-depth knowledge of the power oscillation phenomena in boiling water reactors;

(c) An internal fault on the 4-kilovolt (kV) bus caused a high-energy arc flash and lockout of the bus. One of the workers involved in ongoing Thermo-Lag installation was injured by the blast. It also damaged parts of the bus and associated bus bars. The plant tripped on the resulting undervoltage condition on a vital bus. The force of the pressure wave destroyed the fire door between the two 4-kV switchgear rooms, resulting in smoke migration into the adjacent switchgear room. The most likely cause of the arc flash was conductive material that entered the reactor coil cabinet (or current-limiting reactor. An inductor-type protection device that serves to limit any fault current seen by the low side of the bus (which is constructed with smaller bus bars and breakers) to a value the low side is capable of withstanding). The conductive material also entered a cubicle in the switchgear between the high side of the switchgear, which provides power to the non-safety-related reactor coolant pump and main feedwater pump, and the low side, which provides power to safety-related loads including the high-head safety injection pump. Testing by the operating organization after the event found that the Thermo-Lag 75 used in the installation of Thermo-Lag 770-1 is composed entirely of an electrically conductive carbon fibre. Some pieces and several fibres of the mesh material were found inside the reactor coil cabinet after the event. The foreign material entered either through the unscreened ventilation louvers or through gaps where the bus bars entered the cabinet and created an electrical bridge from the bus bars to the wall of the cubicle. The licensee's installation requirements for Thermo-Lag did not provide adequate guidance to address foreign material. The importance of foreign material exclusion controls has long been recognized, but such controls have traditionally focused on the vulnerability of open equipment, particularly during maintenance periods. This event shows the importance of correctly evaluating that electrical equipment that is not designed to be airtight can be vulnerable to airborne dust and debris in its normal operating configuration;

(d) As a result of operating experience at another plant from an engineering review of mission time requirements for Technical Specification related equipment, a deficiency was discovered regarding the emergency operating procedure (EOP) for natural circulation cooldown with a stagnant loop. This condition could be the result of a postulated main steam line break with loss of off-site power. During a natural circulation cooldown with a faulted steam generator, flow in the stagnant reactor coolant system (RCS) loop associated with the isolated faulted steam generator could stagnate and result in elevated temperatures in that loop. This could become an issue when RCS depressurization to residual heat removal system (RHR) entry conditions is attempted. The liquid in the stagnant loop would flash to steam and prevent RCS depressurization. In this condition, the time needed to complete the cooldown would be sufficiently long such that the nitrogen accumulators associated with the atmospheric steam dumps and turbine driven auxiliary feedwater pump flow control valves would be exhausted. As a result, the atmospheric steam dumps and turbine driven auxiliary feedwater pump would not be capable of performing their specified safety functions of cooling the plant to entry conditions for RHR operation. The EOPs had been revised in 2007 with the intent to allow plants to avoid flow stagnation in the RCS loops by adhering to guidance on cooldown rates versus active loop differential pressure. The changes underwent an EOP review, including simulator validation. Simulator validation did not cover the entire accident from start to finish but terminated after the altered portions of the procedure were successfully diagnosed and entered. Neither the EOP review nor the simulator validation detected the fact that the cooldown curve incorporated in the EOP would involve cooldown rates that were sufficiently slow that cooldown durations would exceed the values used in radiological consequence analyses in the licensing bases. Consequently, the mission time capabilities of the nitrogen accumulators and the turbine driven auxiliary feedwater pump control valves would be insufficient. This event highlights the importance of providing an enhanced level of review to increase the likelihood to identifying deficiencies in generic (nuclear) industry guidance prior to incorporation of the generic guidance into plant procedures.

Category 3.6: Deficiencies in decommissioning

Deficiencies in decommissioning could result in generation of large quantities of radioactive wastes, nonconforming radioactive waste (not meeting the acceptance criteria for disposal or storage), unacceptable quantities of non-radioactive pollutants and/or hazardous wastes, breach of safety barriers (spread of contamination) and unacceptable radiation exposure to workers, the public and the environment. All such events/issues are expected to be reported under this category.

Examples:

(a) The operating personnel noticed the damage to the threaded joints of main coolant pump B and of valve A in the residual heat removal system caused by decontamination chemicals. All main coolant pumps were put into operation for performance of the full system decontamination. Due to increasing shaft vibrations, main coolant pump A had to be switched off manually after 17 days. After continuation of the full system decontamination, 6 days later pump B also had to be switched off manually after similar vibrations. The full system decontamination was then interrupted after the second of seven scheduled cycles. Further investigations included an inspection of the emergency core cooling/residual heat removal system. One control valve could not be moved over the entire stroke length due to a missing stud bolt, resulting in a loosened valve cone that was no longer completely screwed onto the threaded pin. In response to the damage identified, all control valves of the same type were inspected with the result that the stud bolts of five other valves showed damage resulting from material abrasion to a greater or lesser extent. All stud bolts were made of martensitic stainless steel. Functional impairment had only been observed for one valve. The decontamination was performed by an engineering company using a newly developed advanced system decontamination by oxidizing chemistry for pressurized water reactors. A trial decontamination had been successfully performed the previous year in the chemical and volume control system. Subsequently, qualification of the method for use in the primary circuit and the adjacent auxiliary systems took place. Subsequent investigations revealed that in preceding laboratory experiments material abrasion was observed on steels with a chromium content of less than 13%. According to a second test series under dynamic test conditions, the resistance of steels with a chromium content of less than 13% decreases significantly at higher flow velocities. For material combinations with steels with different chromium content that are in direct contact, minor local corrosion damage has also been detected on steels with higher chromium content (14 to 17%). Besides the damages to primary components, the potential safety significance is due to the fact that in the event of loss of offsite power or failure of the operational fuel pool, cooling decontamination chemicals could be injected into the fully loaded spent fuel pool, thus affecting plants in decommissioning or associated preparatory works;

(b) The site procedures for the composition of the low level waste (LLW) were updated as required by the Low Level Waste Repository (LLWR). As a result, processes for measuring radioactivity associated with LLW were reviewed and changed but the equivalent processes for clearance of material and waste were not reviewed and continued to use previous composition data. A regulatory inspection raised the question of instrument settings. For a period of approximately 9 months, the site had cleared some materials and waste using data that is not the most up to date available; it is thought that nothing had left site that would have caused a non-compliance when the updated clearance level data is used. The potential consequences of the event could have resulted in radioactive waste being incorrectly cleared as out of scope of further regulatory control. There would have been only a small risk to the public as the site continued to operate to existing alarm settings that were limiting. The most significant factor in this case is the potential loss of stakeholder confidence. The associated cost to the business for that loss of confidence is difficult to quantify. The site procedures for radiological clearance refer to routine LLW composition of the waste but do not directly address the validity of data being used. Additionally, there is no procedure that triggers timely response to the addition or change of data. The document review of the notification of the composition review was inadequate and set at the end of the process, instead of a much earlier stage in the process which would have allowed for proper assessment and review of potential implications and necessary further changes.

CATEGORY 4: GENERIC PROBLEMS OF SAFETY INTEREST

Events that reveal deficiencies that affect or might affect several plant systems or components, or might have implications for other plants, may be reported in this category. Events which have been recurrent in nature indicate the existence of generic problems of safety significance and are also expected to be reported under this category. These generic problems might not have been adequately identified or addressed by operating experience feedback, research or regulation. The purpose of reporting such events is to draw attention to such problems and enable initiation of corrective action to prevent events with serious consequences.

Examples:

(a) Recent operating experience relating to operator actions in the control of engineered safety features (ESF) equipment had been evaluated. The result of this evaluation indicated that management had not consistently determined, communicated, and implemented a policy defining when it was — and when it was not — appropriate to bypass, defeat, or turn off a safety system. In some events, procedures and other written guidance did not provide clear, consistent guidance. Poor control practices in the areas of communications, shift turnovers, control board walkdowns, verification of automatic actions, and response to alarms contributed to inappropriate ESF defeats and delayed their recognition and recovery. In the events included in the evaluation, recovery from operator defeats of ESFs occurred prior to any serious safety consequence. Accidents and literature on human error show that operator recovery from an inappropriate ESF defeat is not certain. This experience highlights several lessons for strengthening operator control of ESFs;

(b) Several events revealed control rod insertion problems in pressurized water reactors, when operating personnel noted on the basis of digital rod position indications that one or more control rods were not fully inserted in the reactor core by a few steps after a reactor trip. Although most subsequent testing demonstrated that the control rods reached the dashpot region of the guide tube and that adequate shutdown margins had been maintained, there have been indications of degraded rod drop times and stuck rods well above the dashpot region. Thus, the events raise the concern that they may be precursors of more significant control rod binding problems in which the necessary shutdown margins and drop times may be violated;

(c) In the plants where IEEE Std. 946-2004 was utilized to estimate short circuit current contributions in DC distribution systems, licensees need to consider performing a comprehensive review of the entire DC system protection. The IEEE Std. 946-2004 provides information that when the battery charger is connected in parallel with the battery, the battery capacitance will prevent the battery charger contribution from rising instantaneously. Therefore, the maximum current that a charger will deliver on short circuit will not typically exceed 150% of the charger full load ampere rating. Instantaneous battery charger current rise need only become a concern during periods when the battery is disconnected. The regulatory body collaborated in a battery testing programme with a national laboratory to determine if the battery and battery charger current contributions to the fault on the DC distribution circuit would be different when connected individually or when connected in parallel, which could impact the DC system device coordination. The testing validated that the initial fault current contribution to a downstream fault from a battery charger is much higher — in the range of 7 to 10 times the charger full load ampere rating — during the first 100 ms than what is currently stated as 150% in IEEE Std. 946-2004. The test results indicated that the initial short circuit contribution from the charger is not limited when connected in parallel with the battery. The initial higher short circuit current contribution from the battery charger could impact the coordination of protective device settings on the battery charger and downstream devices. Specifically, licensees are encouraged to review their fault current calculations, make any necessary revision to size, and coordinate the protective device settings based on the new information;

(d) Intermingling of signal cables of the safety and non-safety systems was found under the main control room floor of a unit of a nuclear power plant (NPP). The cables need to have been separated from each other by separation plates or barriers. This event resulted in the regulatory body directing all NPP licensees to investigate cable installation conditions, to verify their quality management systems and to report the investigation results. Accordingly, improper cabling was found in many units. The causes were as follows: 1) The NPP licensees ordered the cabling work without clearly stating the requirements for divisional separation in the procurement specifications; 2) The licensees did not have any verification processes for the cabling work that meets the procurement requirements. The root cause was the licensees' direct order for cabling work to the contractors without involving NPP manufacturers who used to supplement the licensees' procurement requirement specifications. The measures taken to prevent a recurrence include the following: (a) Incorporation of the divisional separation requirements in the cabling work specifications that are being provided to contractors, (b) Development of the impact assessment of cabling work on the safety, (c) Implementation of training and education on divisional separation of safety systems to the licensees' and contractors' workers;

(e) A large corrosion hole was found on heating, ventilation, and air-conditioning (HVAC) ducts for the main control room (MCR) during a regular outage at an NPP situated on the coast. The corrosion hole could impede the expected MCR environment control (MCREC) function in certain accident conditions. The possible causes or mechanism were as follows: 1) The pre-filtration subsystem has not been operated except in stormy weather condition. 2) Moisture naturally adheres inner surface of the duct pieces in which air flow direction changes. In such duct pieces, corrosion often grows and penetrates through the duct wall. Salty moisture leaks out through the penetration and corrodes the duct outer surface. 3) Salty water trapped by the guide vanes has dropped and retained on the lower surface of the duct pieces. 4) Visual inspection on the duct outer surface has been performed without removing the thermal insulation. Duct inner surface visual inspection has been carried out only near the outside air intake. 5) Regular MCREC function and performance tests have been successfully performed owing to the duct thermal insulation, which could protect unfiltered air in-leakage. Lessons learned from the event are: 1) MCREC function and performance tests do not necessarily assure the duct robustness or air tightness. 2) Corrosion on the duct inner surface could penetrate the wall and grow on the outer surface under the thermal insulation.

CATEGORY 5: ENFORCEMENT AND CONSEQUENTIAL ACTIONS TAKEN BY THE REGULATORY BODY

This category is intended to include significant enforcement or consequential actions taken by the regulatory body and resulting from lessons learned from reported events. Consequential actions would be changes to regulatory requirements related to the licensing process, the design or operation of NPPs. They include: important modifications to the design basis; changes to regulatory requirements for design assessment; changes to fault trees considered in probabilistic safety analysis; important changes to the regulatory requirements for construction, commissioning, surveillance and decommissioning of plant systems; changes to regulatory requirements for plant staff; changes to off-site emergency planning, etc.

Examples:

(a) As a follow-up of an event in a boiling water reactor with unexpected clogging of emergency core cooling system (ECCS) suction strainers, the impact on related regulatory requirements in another country was reported. The event revealed a type of common cause failure for the ECCS function which was not adequately addressed by regulation. The immediately performed safety analysis showed significant design differences with regard to the affected plant but concluded that clogging of suction strainers could not be completely excluded. Therefore, as an interim measure, accident management procedures for alternate core cooling and backflushing of strainers have been developed. After further detailed analysis, final measures to increase the suction strainer area were decided. Taking into account that at that time a research programme for quantifying the design impact was still under way, the new strainers were designed with sufficient safety margin so that later redesign could be avoided. The following regulatory assessment confirmed the proposed design change;

(b) As a follow-up to an event in another country with excessive channel box bow that could result in a loss of thermal margin in boiling water reactors, actions taken by a regulatory body and nuclear fuel vendors were reported. Although no known failures of this nature had occurred, the operating organizations of boiling water reactors were requested to determine whether any channel boxes were being re-used after their first bundle lifetime, and, if so, to ensure that the effects of channel box bow on the critical power ratio calculations were properly taken into account. Fuel vendors submitted generic reports to the regulatory body describing their methodology for incorporating the effects of channel box bow in their analysis;

(c) Upon receiving notification of a carbon segregation issue on the large forged components of nuclear pressure equipment from a foreign nuclear safety authority, the regulatory body conducted a hearing from the domestic forging companies who manufacture large forged components of nuclear pressure equipment, to understand their manufacturing and product qualification processes of their forged components. The regulatory body learned that the dominant factor in the carbon segregation issue was "sufficient discard" of high carbon concentration volume from the steel ingot and the half-finished product. The regulatory body requested all the NPP licensees to report on the forging methods applied to the larger forged components of their NPPs. The licensees were also asked to check the manufacturing records and the material inspection certificates at the forging companies. The regulatory body received the reports and concluded that large forged components of nuclear pressure equipment used in the NPPs in service are free of this carbon segregation issue;

(d) The USNRC has conducted a case analysis of events involving high-energy arcing faults (HEAF) since the early 2000s. In addition, the OECD/NEA set up a working group on HEAF in 2009. A domestic NPP has also experienced an HEAF event. HEAF has been drawing attention around the world as there is a need to develop a method for evaluating the impact of HEAF. The regulatory body having engaged in research on HEAF, learned that shortening the duration of continuous arcing by applying high-speed circuit breakers can mitigate the impact of explosions attributed to arcing and can prevent the occurrence of arcing fires. The regulatory body amended the ordinances on technical standards for nuclear installations and relevant standard review plans. After public hearings were carried out, they were promulgated and put into effect;

(e) An NPP shut down after an open phase condition (OPC) event. The shutdown was caused by unbalanced electrical voltage coming into the plant from the regional electric grid. The plant (and also other NPPs) was not designed to automatically detect the offsite power source in OPC because the OPC detection systems were not available at that time. The regulatory body amended the regulatory guide to stipulate that measures should be taken to detect an OPC on the primary side of the transformers connected to the safety bus of nuclear installations;

(f) Some NPP licensees discovered their residual heat removal (RHR) systems were potentially inoperable during shutdown periods because of elevated system temperatures at the RHR pump suctions. The elevated system temperatures resulted from the lack of adequate procedures to ensure RHR system operability during all modes of operation. At affected plants, the fluid in the piping between the reactor coolant system hot leg to the RHR system connection and the RHR minimum-flow line return connection remained stagnant and at elevated temperatures following forced cooling as a result of unrecognized system flow characteristics; namely, forced flow did not occur in that section of pipe. Consequently, each licensee concluded incorrectly that the RHR system was properly cooled, prior to shifting the RHR system to emergency core cooling system injection mode, when the system temperature was such that the affected RHR systems could have incurred steam voiding had they been used for emergency core cooling purposes. The regulatory body issued an information notice to inform licensees of the discovery. Recipients were expected to review the information for applicability to their facilities and consider actions to avoid similar occurrences.

CATEGORY 6: EVENTS OF POTENTIAL SAFETY SIGNIFICANCE

This category is intended to include events which did not have actual safety consequences of any significance, but which nonetheless are of potential safety significance. It covers events where protective systems were actuated to mitigate the consequences of an event or where these systems had been challenged unnecessarily. It especially includes near-miss situations that may also be precursors to more serious events. It may include events that lead to potential loss of a safety function. (Note: Once an abnormal trend has been identified, it may be treated as an event.)

Examples:

(a) Prior to start-up after the refuelling outage, the primary system in a pressurized water reactor was degassed under vacuum. This process was carried out at mid-loop operation. During the degassing process, the shutdown cooling (SC) function was completely lost. Core cooling was restored after venting of the SC pump, increase of the primary coolant level and lowering of the SC flow rate. As the coolant inventory at mid-loop is low, the time for recovery to prevent boiling in the primary circuit is limited. Additional failures might have led to serious consequences. The degassing procedure was modified to limit the maximum SC flow rate, to increase the primary coolant level and to introduce hold points to verify the operability of primary coolant level measurements;

(b) During a refuelling outage in a pressurized water reactor, the pneumatic seal which seals the annulus between the reactor vessel and the bottom of the refuelling cavity failed. Despite attempts to minimize the resulting drainage of the refuelling pool water, approximately 750 cubic meters of borated and mildly contaminated water was lost to the containment floor. The only actual consequences were exposure of equipment and structures in the containment to water. The most serious potential safety consequence was uncovering of the spent fuel elements. The cause of the event was inadequate design of the replacement pneumatic seals. Corrective actions included containment dewatering and decontamination, equipment damage assessment, seal design modification, integrated event safety analysis and procedure review;

(c) Arc flashes in a 4160-volt electrical system resulted in two phases of the system faulting to ground and a phase-to-phase fault. The high differential current actuated the lockout on the plant start-up auxiliary transformer (SAT), which started all emergency diesel generators. The loss of power from the SAT resulted in shutdown of both reactor recirculation system pumps and subsequently a manual scram. The manual scram automatically shut down the main turbine and main generator. With the main generator offline, the power circuit breakers for the unit auxiliary transformer (UAT) opened and de-energized the UAT. Since the SAT was already locked out, the power source for emergency busses could not transfer to the SAT. Therefore, a loss of offsite power condition existed on emergency electrical busses. In the case where the plant emergency diesel generators would not start properly and supply power to the affected emergency busses the reactor core isolation cooling system and the high-pressure coolant injection system would not start to control reactor water level and pressure, respectively;

(d) In the spent nuclear fuel storage facility, it was discovered that power cables in three cable trays had caught fire. The appearance of smoke inside the facility activated the fire alarm system. The fire was extinguished by staff and a rescue unit of the Federal Fire Service. This was a potentially dangerous event at the spent nuclear fuel storage facility. Arrangements were made for the fire to be extinguished and for additional fire suppressing equipment (fire extinguishers) to be brought to the location of the fire. The event had no radiological consequences. The operability of the process equipment in the spent nuclear fuel storage facility workshop's processing and storage sections was not affected;

(e) While implementing a new maintenance programme, it was identified that the steel structures supporting emergency diesel generator (EDG) expansion tanks were not included in any scheduled maintenance programme. Neither were the structure anchors. Further investigations revealed that several beams of the supporting structure were missing on other EDGs. Further investigations revealed that more than 10 support systems components of EDGs were affected with insufficient anchoring depth and the specification drawings for the anchors were missing. In addition to incorrect anchoring, investigations highlighted the fact that several expansion tanks were corroded, due to rainwater trapped between the tanks and the thermal insulation. In the event of a maximum historically probable earthquake, all affected EDGs would have failed, due to common-cause failure. The ultimate electric power supply (combustion turbine and ultimate diesel generator) is not designed to be seismic resistant. In the event of a total loss of external and internal power supply, it would not be able to power the pump ensuring primary circuit integrity by injecting cold water at the primary pump seals. Therefore, primary integrity could be lost and safe shutdown of the reactor could not have been ensured, possibly resulting in a nuclear fuel melt accident.

CATEGORY 7: EFFECTS OF UNUSUAL EVENTS OF EITHER HUMAN OR NATURAL ORIGIN

This category is intended to include those events caused by an external act or condition that might challenge the ability of the plant to continue to operate, or to shut down, or to maintain shutdown conditions in a safe manner. The category includes internal and external hazards such as natural events (e.g. high wind, earthquake, flood, ice formation, pollution of river water or seawater, lightning strikes, heavy rain or snowfall), human induced external events (e.g. explosion, fire, industrial transportation accident affecting the plant, aircraft crash) and internal events (e.g. explosion, fire, flooding, toxic gas release, turbine missiles).

Examples:

(a) A grid disturbance in winter triggered a turbine generator trip, a reactor scram and a switching-over of the electrical supplies to the ‘auxiliary external source’. During the event, several failures of outside equipment were caused by low temperatures. These affected a supply line to the emergency feedwater storage tank, level measurements of the 2000 ppm borated water storage tank, and boiler safety valves. This event confirms that cold weather can initiate common mode failures of equipment of different systems;

(b) In a pressurized water reactor, unjustified engineered safety feature actuation occurred twice, initiated by erroneous tripping of the ‘main steam header’ rupture signal. At the same time, similar but smaller steam line pressure decreases were observed on the other three units on the same site. The investigation revealed that all events were caused by the use of radio transmitters/receivers. The subsequent laboratory tests confirmed that these radiofrequency devices were able to emit high energy signals even when they were switched off. The lessons learned showed that shielding measures need to be considered to protect sensitive instrumentation and control equipment against electromagnetic interference;

(c) Aggravated wind conditions together with an incoming tide led to a significant influx of seaweed to the station cooling water inlet. Due to the impaired cooling system, both reactors at the site were manually shutdown. In one of the units, there was a partial temporary loss of the reactor seawater cooling system. This system was used to cool safety systems and items important to safety including the concrete pressure vessel and main reactor gas coolant circulator lubrication system. The lesson learned revealed the need for comprehensive guidance to operating personnel for the actions to be taken if a large ingress of seaweed occurs. The event confirmed the effects of a change in weather conditions on the plant operation;

(d) A fire broke out in the hill near the NPP. Heavy smoke from the fire led to tripping of the outgoing transmission line and the loss of off-site power: the unit was operated in ‘island’ mode and was subsequently shut down. This hill was covered with vegetation and thus was a major unconsidered fire hazard that had been inadequately assessed and no safety separation area along the outgoing transmission had been established. After the event, the safety impact of the hill vegetation was assessed to identify and implement the relevant safety and technical measures;

(e) Subsequent to an earthquake and tsunami, off-site power to the NPP was lost. Due to loss of normal power, a laboratory sump in a radiation controlled area had a continuous supply of water through the solenoid valve (design to open on loss of power) for the seal water of the sump pump. The water level high signal of the sump did not actuate due to loss of power supply. The water filling the laboratory sump flowed back into a funnel in the battery room of a non-radiation-controlled area. Upon investigation, it was found that this battery room was earlier changed from a radiation controlled area to non-radiation-controlled area, but the piping connecting to sump with the funnel was left unchanged. The event resulted in the release of radioactivity outside the radiation controlled area. This event was initiated by the external hazard and it progressed due to a design deficiency.

CATEGORY 8: OTHER FINDINGS AND OPERATING EXPERIENCE INFORMATION

This category is intended to include: new perspectives; industry initiatives; information gained from the results of analysis, research, or benchmarking/review of events; and issues in other industries. Reporting in this category may include the effects of changes in safety standards or new regulatory requirements that may impact NPP systems. Research results in the area of digital instrumentation and controls, the human-machine interface, etc., which may be useful in the review of NPP design changes/modifications may also be reported under this category.

Examples:

(a) The failures of certain materials in other industries that may have applicability with respect to materials being used or considered by the nuclear industry;

(b) The implications of ultra-low sulphur diesel fuel or biodiesel on the performance of emergency diesel generators;

(c) New safety requirements such as for severe accident management guidelines may have implications for NPP emergency planning and operational procedures;

(d) Results of the use of operating experience information from other sources, including good practices that have been identified as improvement opportunities that could be useful in the prevention of future events or the lessening of the severity of events;

(e) Experience with risk informed optimization of technical specifications;

(f) Results of the tests that demonstrated that the hazards from a high energy arcing fault may be substantially greater for electrical equipment that contains aluminium components than for those with only copper components;

(g) Operating experience that challenge existing regulatory guidance regarding the effects of in-reactor service on fuel assembly component response to externally applied forces.

APPENDIX II - PROCEDURE FOR THE SELECTION OF EVENTS AND INFORMATION AND FOR THE PREPARATION OF IRS REPORTS

II.1. INTRODUCTION

The objective of this procedure is to help the user to prepare an IRS report on an event/ information so that important lessons learned are effectively transferred to the international nuclear community. This procedure focuses on the content of the information to be provided in the report rather than on its format.

In practice, it is recognized that the compilation of the information for an IRS event report is normally performed in a logical order different from the standard IRS report format. This is reflected in the procedure.

For events where the lessons learned are primarily derived from human performance, more detailed guidance on the specific information to be supplied is provided in the procedure. This guidance differs from that for the provision of technical information and takes into account that the engineering world is usually less familiar with human performance analysis than with technical analysis.

The WBIRS is designed to assist IRS National Coordinators in sending IRS reports and it provides the structure for the preparation of a report. Information on how to use the WBIRS is described in the IRS coding manual [2] and is not part of these guidelines.

II.2. SELECTION OF EVENTS/INFORMATION FOR REPORTING

Events/information to be reported to the IRS are expected to be selected according to the general criteria provided in Section 7.1.

II.3. TYPES OF REPORTS

The types of reports are as follows:

- **Preliminary report** – A preliminary report is expected to be prepared for an event with actual and significant safety consequences, or for an event with an actual or potentially significant reduction in defence in depth. The purpose of preliminary reporting is to make the nuclear community aware of the risks related to the event and to allow the other Member States to promptly respond to the situation, as appropriate;
- **Main report** – The main report may be an initial report if a preliminary report was not needed or may supplement a preliminary report. There are two types of main report:
    - A specific main report associated with a single event/information;
    - A generic main report, associated with a set of events/information related to each other, and produced to focus on common lessons learned from the events/information.
- **Follow-up report** – A follow-up report is expected to be prepared when new or different information is identified that may improve understanding of the event/information and the effectiveness of its associated corrective and preventive actions. The responsibility for identifying the need for a follow-up report lies with the IRS National Coordinator. Depending on the significance of the event/information, a preliminary report may be useful (see II.4), otherwise a main report is to be provided (see II.5). As soon as the necessary information is available, a main report is expected to be prepared to supplement the preliminary report.

As a general guidance for submitting reports, IRS National Coordinators are expected to consider the following:

- Do not wait until exhaustive information is available in order to prepare and send a full report. Send the report whenever sufficient material is available;
- Consider that, if new elements become available later, you still have the opportunity to send a follow-up report.

II.4. PRELIMINARY REPORT

The preliminary report is expected to summarize the information available at the time the report is prepared. It includes a short description of the event/information, the preliminary safety evaluation and the short-term actions taken and lessons learned, if available.

Member States are expected to submit a preliminary report to the IRS preferably within thirty days of the date of the event. This especially applies to events that have significant generic applicability, or provide significant lessons learned.

II.5. MAIN REPORT

The following provides guidance for each section on the IRS reporting template.

II.5.1. Identification of the necessary information

From the available information, extract and sort the following items (if available):

i. General data, such as plant name/unit and date/time of the event;
ii. Plant conditions before the event and methods of event discovery (in case of a deficiency);
iii. The factual event sequence as observed, including any observed degradations or malfunctions of systems and the reasoning or reactions of people at the time, and the impact on the event sequence. Clearly identify the observed cause–consequence relationships;
iv. A consequence analysis to determine whether or not some aspects of the event are indicators of indirect problems or weaknesses which, under other circumstances, could also lead to a safety significant event or a serious accident;
v. An analysis of the event, identifying the root causes and causal factors, the impact(s) on safety, and the investigative and corrective actions taken. The causes and corrective actions are expected to address technical as well as human and organizational factors/aspects/deficiencies. If possible, provide an indication of how each given deficiency has been corrected;
vi. Assessment by the regulatory body, to the extent possible;
vii. If the event description and/or analysis need additional plant-related information to be made available to readers to facilitate understanding, then provide the necessary information on plant features.

II.5.2. Formalization of the collected information into the event report

To properly apply the general format for IRS reports, the following process is to be applied.

II.5.2.1. Prepare the narrative description

Plant Features:

- Provide the technical, human and organizational data (factors/aspects/deficiencies) necessary to understand the event;

- Reactor systems and terminology are not universal. It may be helpful for other IRS users to include a brief description of the systems and diagrams and/or flowsheets, practices, procedures and/or organizational characteristics that influenced the event, if feasible. For a better understanding, descriptive names for equipment need to be used rather than internal identification codes.

Event sequence and personnel reactions:

Provide all relevant information on what happened during the event and on the general context of the event. The following information is expected to be provided:

(a) Situational aspects:

i. Plant conditions prior to the event;
ii. Operating modes or testing conditions;
iii. Equipment status.

For events where human performance plays a significant role, the following information is expected to be provided where available (see Ref. [2] for more details):

i. Plant staff involvement;
ii. Type of activity at the time of the event;
iii. Characterization of the personnel and individual task related work practices;
iv. Characterization of the working conditions;
v. Any other relevant organizational aspects.

(b) Chronological information:

i. Chronological information indicating relevant timescales;
ii. Identification of failures and successes in responding to the event, including any that occurs during the recovery actions phase.

For events where human performance plays a significant role, the following information is expected to be provided where available:

i. Information on the nature and timing of recovery actions. Such information may provide additional insight into the complexity of the situation and the difficulties for the operating personnel to detect and diagnose the problem at hand. Lessons may also be learned from the positive role of plant personnel involved in the event. If relevant, include a discussion of the recovery actions, providing information on how and when the recovery was achieved. Identify the types of plant staff involved in the recovery actions;
ii. Detection and diagnosis activities, including delays encountered;
iii. More specific information on any time delays encountered in detection and diagnosis activities is useful for the evaluation of human errors, system failures, and the safety problem presented by the event. Indicate, if applicable, any factor leading to a lengthy delay before a problem was detected or diagnosed;
iv. Any human and/or organizational errors involved. This includes errors of commission as well as omission, and what would have been the correct action(s), if known;
v. Intra- and extra-team communication aspects.

This description need not focus too much on causes, in order not to duplicate the cause analysis.

Previously related events or precursors are expected to be indicated.

If available, add figures, including layouts, photographs and/or drawings, to facilitate a better understanding of the environment in which the event occurred.

II.5.2.2. Prepare the safety assessment:

- Address the actual and potential consequences of the observed problem(s). In particular, a discussion of the barriers which were broken by the observed deficiencies and the effective barrier that terminated the event is expected to be included;
- When relevant, safety aspects related to human performance are expected to be included;
- If the assessments by the licensee and by the regulatory body are different, this needs to be indicated;
- If an event/information is recurrent, the analysis is expected to include an explanation why the already implemented corrective action(s) has not been effective.

II.5.2.3. Prepare the cause analysis:

Indicate clearly here, when relevant, the direct causes as well as the root causes:

- The presentation and discussion of the direct causes (i.e. the failures, actions, omissions or conditions that immediately produced the event) is expected to identify the technical, human and organizational deficiencies and answer the question; "How did it happen"? The presentation and discussion are expected to also provide, to the extent possible, the results of the analysis identifying the nature of failures or errors;
- Human factors investigations have shown that human performance is strongly dependent on the context. Interaction between operating personnel and plant systems creates a dynamic context where operating personnel receive plant information. Using human error prevention tools, training and procedures, most of the operating conditions are easily managed. Overconfidence in one's knowledge of systems (machine) can affect operator actions which, consequently, could affect the plant response to transients and the overall event sequence. This is but one of the factors that can lead to unsafe conditions;
- Other key elements which can affect the evolution of a normal operating condition to a potential event situation involve the work environment, communication, the effectiveness of shift turnover, etc.. The event description is expected to address such latent or contributing failures as well as errors causing the initiating event or degrading the plant condition;
- All possible situations where human factors affect event development are expected to be considered during event analysis;
- For events where human performance played a significant role, the types of observed human errors that contributed to the initiation of the event or directly affected the response of operating personnel or the system to the event are expected to be provided, where available. The human performance related causal factors and root causes are addressed in Ref. [2]. In addition, plant staff involved in the event have also to be identified. Causal factors and root causes are expected to be carefully evaluated to distinguish individual human performance and organization related factors;
- The report is also expected to provide the identified causal factors relevant to the message to be conveyed. These causal factors are causes that, if corrected, would not by themselves have prevented the event, but are important enough to implement worthwhile corrective actions to improve the quality of the process or product;

- A presentation and discussion of root causes is expected to follow. These are fundamental causes that, if corrected, will prevent recurrence of the event or of its adverse environment. Both causal factors and root causes provide the answers to the question; “Why did it happen”? Organizations are encouraged not to stop a cause analysis at a human factor, but to continue the analysis to identify an underlying root cause;
- If possible, include an event and causal factor chart to illustrate the analysis results.

II.5.2.4. Prepare the lessons learned and corrective actions

Describe the set of corrective actions taken by the operating organization to address the observed technical, human and organizational deficiencies. The priority of the various corrective measures is also expected to be provided if it improves understanding of the significance of the various causes. Corrective actions may cover administrative measures as well as hardware modifications taken to lower the likelihood of technical, human and organizational deficiencies.

Corrective actions can be divided into three types:

- Immediate corrective actions taken promptly to restore normal conditions or eliminate problems, for example, equipment is repaired, procedures are updated;
- Interim corrective actions that are short term actions to reduce risk of recurrence while awaiting long term corrective actions. They can be accompanied by compensatory corrective actions;
- Corrective actions to prevent reoccurrence. These actions are most important to prevent events from happening again.

For events where human performance plays a significant role, include when available:

- Changes in behaviours, attitudes or habits of persons or groups;
- Changes to the initial and continuing training programmes. Indicate what was lacking in terms of knowledge and know-how;
- Changes to procedures, or new procedures;
- Organizational changes;
- Improvement in ergonomics;
- Hardware modifications which affect the human-machine interface;
- Conduct of re-training to correct specific knowledge deficiencies.

Describe also any specific or enforcement actions taken by the regulatory body in response to the event. An indication of the generic character of the regulatory actions taken or of difficulties in designing or implementing the corrective actions may be useful.

The content and formulation of the lessons learned need to be practical and applicable to other nuclear power plants and be consistent with the basic safety message to be conveyed.

II.5.2.5. Prepare the abstract

The objective of the abstract is to convey the main messages contained in the report, essential for the understanding of the relevance of the event or conditions. A good abstract provides, in a concise form, a brief description of the event, its safety relevance, its causes, the lessons learned and the corrective actions taken.

II.5.2.6. Choose a title

The title is to be a short characterization of the event, emphasizing its most significant features. The title, in any case, need not be too generic.

II.5.2.7. Prepare the cover page, codes and Priority Level

A report can be directly submitted into the WBIRS using the various chapters. Alternatively, text can be uploaded from any word file into the various chapters.

Cover page

Fill in the cover page information to identify the event:

- Indicate the report type as described in Section 3 of this Appendix;
- Fill in the title as determined in Section 5.2.6 of this Appendix:

  Fill in the plant name and code. For specific reports, only a National Coordinator can input information for/from her/his own country and can select the plants displayed for this county from a drop-down menu. For generic reports, the WBIRS allows for inputting of one or more plant names for a given country or no plant name at all. For follow-up reports, the WBIRS allows already existing information to be updated.
- Fill in the date of event. In the case of specific reports, the user has to choose the incident date. In the case of generic reports, the WBIRS allow for the user to provide a specific date, a date range, a specific month, a month range, a specific year or a year range.

Codes:

- As the codes are provided for retrieval purposes, they need to reflect the event conditions, the observed phenomena and the problems encountered;
- More than one code can be selected under each category. The more detailed the coding, the better;
- Refer to IAEA Service Series 20 [2] for the definitions and usage of the available codes.

Priority Level:

- Each IRS report is assigned a Priority Level, by the ERG, based on the safety significance and/or importance of the lessons learned. Member States that submit a report may make an informed suggestion on the Priority Level; however, the IAEA ERG makes the final determination of Priority Level to ensure consistency in the grading system.

5.3. Formalization of the collected information

The WBIRS allows for documents and figures to be uploaded in the following formats: .tif, .jpg, .jpeg, .pdf (the Microsoft Word file format is not recommended).

Reporting of other operating experience information can be done using the same structure as mentioned above or by using the full report format.

II.6. FOLLOW-UP REPORT

When corrective actions prove to be insufficient or further information is revealed, a follow-up report is expected to be issued to explain the failure to solve the problem. For example, if longer term corrective actions related with human factors have been identified in a main report, a follow-up report discussing those corrective actions may be made.

APPENDIX III - PRIORITY LEVEL

For the convenience of the reader, the criteria from Section 8 is repeated below)

IRS reports are categorized into three Priority Levels:

Level 1:

- Events that caused or had the potential to cause a major reduction in the defence in depth or safety functions of a plant;
- Events that caused or had the potential to cause excessive radiation exposure or serious harm to individuals;
- Safety significant events as determined by other inputs including Probabilistic Safety Assessment /Probabilistic Risk Assessment (PSA/PRA) evaluations or INES ratings that may have been provided in the IRS report. A breakdown of multiple barriers with major impact on plant safety is a typical event for this category. These events have significant generic applicability, or provide significant lessons learned.

Level 1 events are used to provide specific suggestions to Member States through agency publications developed for the industry such as the Blue Book [5].

Examples:

(a) Enhanced inspection and repair of systems susceptible to the effects of environmental corrosion was undertaken at an NPP. Seismic restraints and pipework associated with additional feed, backup feed, pressure sustaining unit, auxiliary cooling water, backup instrument air, essential cooling water recirculation, water spray fire, reserve feedwater and backup cooling water systems, as well as the backup diesel generators, were found to be corroded to an unacceptable condition. This was due to an inadequate level of technical rigour and oversight applied to the quality of walkdowns, safety justifications and prioritization of work. The event is assigned Priority Level 1 because of concerns related to the defence in-depth and potential consequences associated with common-mode failure due to the degradation of multiple systems;

(b) The essential service water (ESW) is the ultimate heat sink for most safety systems and consists of two trains, each of 100% capability. Before the event, at 100% power, train A of ESW water was operating. When the standby pump of train B was started up for a post-maintenance check, a manhole pipe downstream of the pump broke circumferentially and the water flowed across the complete section. The licensee shut down the plant to a safe condition. The cause of the break was corrosion at an inspection manhole neck in the pipes of the ESW. Pervasive corrosion was found at most inspection manhole necks in both trains. The event is assigned as Priority Level 1 because of concerns related to potential consequences on the ultimate heat sink for most safety loads, including components cooling water, motors of safety injection pumps, emergency diesel generators, and safety ventilation systems.

Level 2:

- Events that reveal important lessons and which caused or had a potential to cause a limited impact on safety functions. Typically, these events cause an unexpected change in plant conditions, equipment status, or had an adverse effect on plant safety. Lessons learned from these events may help the international nuclear community prevent a recurrence of the event or the occurrence of more serious events (Level 1 events);

- Reoccurring events that would have been prevented by the implementation of corrective actions and preventive measures based on the lessons learned from previous similar events;
- Events involving organizational or human factor issues such as those caused by a degraded safety culture at a plant.

Individually, Priority Level 2 events might not have generic applicability but may provide inputs on important adverse trends to Agency publications including the Blue Book [5].

Examples:

(a) In various events, steam leaks were observed in the feedwater circuit. The direct cause of the steam leaks was flow-accelerated corrosion (FAC). The root cause was attributed to the deficiencies in FAC programmes. In one of the events, the data input to the FAC software model was incorrect which caused the software to underestimate the predicted wear rate. In another event, a replacement history to identify susceptibility to FAC was not used. While the events did not involve safety system and items important to safety, they did result in reactor trips that were complicated by the loss of the normal heat sink due to the necessity of isolating the high-energy leak in the feedwater system. This event is assigned Level 2 as it revealed important lessons to apply appropriate engineering judgment and not to place overreliance on the FAC programme software;

(b) The NPP unit was in operation at full power. During regular comprehensive testing of safety system train No. 2 according to the automatic load sequencing procedure, diesel generator No. 2 successfully started, but it was soon tripped by the overspeed protection system, due to a failure of the speed controller. Operating personnel focused their efforts on finding the malfunction and did not perform comprehensive testing of safety system trains Nos 1 and 3 to confirm their operability. In doing so, the personnel violated the safe operation conditions established in the plant technical specifications. Tests of safety system trains Nos 1 and 3 were later performed in compliance with the planned schedule and no deficiencies were found. The diesel generator No.2 speed controller failed because of drawbacks in planning and performance of maintenance. This event is assigned Level 2 as it involved organizational and human factor issues degrading the safety culture at the plant.

Level 3 is assigned to events that did not result in notable consequences but had the potential to cause events that are consequential under slightly different circumstances, and if shared, may help the international nuclear community identify potential precursors and prevent a recurrence of the event or the occurrence of more serious events (Priority Level 1 or 2 events). These events, by themselves, usually have little generic importance but may be useful in providing inputs on adverse trends to agency publications including the Blue Book [5].

Example:

(a) An emergency diesel generator was declared unavailable as per its technical operating specifications owing to an error committed during preventive maintenance on the motor of a fan used to cool the generator set. The error consisted of the fan's direction of rotation having been reversed and not having been identified during the following post-maintenance test. It was detected during a periodic test on the diesel generator during which all four fans were in operation. The staff member in charge of the test noticed that one of the four fans was rotating in the opposite direction to the others. If there had been a loss of offsite power in hot weather during the period from the post-maintenance test to the periodic test, the emergency diesel generator would not have been able to perform its functions and provide emergency power to the dedicated train equipment that is essential to maintaining the reactor in a safe state.

All events/information (Levels 1, 2 and 3) will be reviewed by the Event Review Group (ERG) for trending purposes once a year at minimum and if any adverse trends or commonalities are identified they will be evaluated and communicated as appropriate to Member States. An example is the spurious actuation of the plant protection system (PPS) due to the random failure of the PPS relays. These types of events can be reviewed by the Member States at their discretion and will be trended by the ERG.

APPENDIX IV - IRS ADVISORY COMMITTEE (IRSAC): CONSTITUTION AND ELECTION PROCESS

IV.1. IRSAC MEMBERSHIP

The IRSAC is constituted of a total of eight members including six elected IRS National Coordinators, one IAEA representative, and one OECD/NEA representative.

The IRSAC includes a minimum of two representatives from an OECD/NEA country and two representatives from a non-NEA country.

IRSAC membership is granted only to participating countries or respective organizations (IAEA and OECD/NEA).

An IRSAC member country, the IAEA, and the OECD/NEA may change the designated representative without affecting the status of its IRSAC membership for the remainder of the elected term.

IV.2 CONDUCT OF IRSAC ACTIVITIES

The IRSAC members including the IAEA and OECD/NEA representatives designate one elected IRSAC member to act as Chair of the IRSAC. The Chair of the IRSAC presides over the IRSAC meetings for the remainder of the elected term. If the IRSAC members decide not to designate a Chair, by default the meeting will be chaired by the IAEA representative (IRS Coordinator).

The IAEA representative and the OECD/NEA representative are responsible for coordinating the IRSAC activities associated with the meetings, including organization of the IRSAC meetings and the recording and distribution of the meeting proceedings.

The IRSAC quorum consists of a minimum of five members as follows:

- An IAEA representative;
- An OECD/NEA representative;
- Three elected National Coordinators.

IRSAC decisions are made by majority vote during meetings in which the quorum is established (IRSAC meetings which do not meet the quorum may be held for information purposes only). In the event of a tie, the decision will be made in accordance with the vote of the IRSAC Chair.

The IRSAC agenda always includes the following items:

- Roll call (confirmation of quorum);
- Approval of the agenda;
- Adoption of the minutes of the previous IRSAC meeting;
- Report from the Chair;
- Report from the IAEA representative;
- Report from the OECD/NEA representative;
- Actions arising from the minutes of the previous IRSAC meeting;
- Preparation of the next Technical Meeting of the IRS National Coordinators;
- Any other business.

Representatives from any Member State wishing to attend an IRSAC meeting as observers may do so at their discretion; however, they are expected to notify the IAEA representative, the

OECD/NEA representative, and the Chair of the IRSAC of their intent to attend no less than one week prior to the date of the IRSAC Meeting.

IV.3. ELECTION PROCESS

The term of elected IRSAC members is four years.

A minimum of three IRSAC National Coordinator seats are subjected to an election every two years.

The IAEA and OECD/NEA representatives are responsible for the conduct of the elections and for validating the results.

Each participating country is entitled to vote for as many candidates as the number of IRSAC seats that are up for election. The National IRS Coordinator for a participating country is responsible for the votes cast on behalf of the respective country.

Newly elected IRSAC Members take office immediately after the end of the term of the previous elected IRSAC members.

Elections are conducted in 4 steps, as follows:

1. Candidacy;
2. Voting;
3. Counting;
4. Announcement of the results.

Elections may be conducted using electronic means of communications or by a vote held during a Technical Meeting of the IRS National Coordinators using confidential ballots.

IV.3.1. Candidacy

The IAEA representative issues a request for candidates to all IRS National Coordinators and the OECD/NEA representative approximately 90 days prior to the Technical Meeting of the IRS National Coordinators or whenever the Chair of the IRSAC determines that an election is necessary.

The member country of the IRS National Coordinator wishing to submit a candidacy may include a brief introductory statement relating to professional background and/or work experience.

The IAEA representative confirms that enough participating countries have submitted their candidacy to fill all IRSAC positions up for election and issue a notice to formally close the candidacy period.

The IAEA representative is also responsible for notifying the National Coordinators of the election candidates and their credentials approximately 30 days prior to the Technical Meeting of the IRS National Coordinators.

IV.3.2. Voting

The IAEA and OECD/NEA representatives organize and select the date for the election and vote.

Voting may be conducted electronically or during the Technical Meeting of the IRS National Coordinators. The conduct of an electronic vote lasts approximately 25 days. Votes have to be received by the IAEA and NEA representatives by the close of business on the closing date of the voting.

The IAEA and the OECD/NEA representatives duly collect the votes cast by the IRS National Coordinators in a confidential manner.

IV.3.3. Counting

The IAEA and OECD/NEA representatives tabulate and count the number of votes received by each of the candidates.

The IAEA and OECD/NEA representatives determine which candidates received the largest number of votes. The country candidates receiving the highest number of votes are declared elected provided the IRSAC includes a minimum of two representatives from OECD/NEA countries and two representatives from non-OECD/NEA countries. A candidate with insufficient votes may, nonetheless, be declared elected if this is necessary to ensure a minimum of two representatives from OECD/NEA countries and non-OECD/NEA countries.

If two or more candidates receive an equal number of votes, the IRS National Coordinators will be asked to break the tie through a second confidential vote.

The IAEA and OECD/NEA representatives review and validate the election results.

IV.3.4. Announcement of the results

The IAEA and OECD/NEA representatives issue a combined notice to formally announce the IRSAC election results within 5 days of the close of the vote counting.

APPENDIX V - DEFINITIONS

The following definitions are specific to this publication and are not provided in the IAEA Safety Glossary [6].

causal factor. Causes that, if corrected, would not of themselves have prevented the event, but are important enough to be recognized as needing corrective action to improve the quality of the process or product. Also, a factor that influences the outcome of a situation. The reasons for an action that was taken or an event that occurred in the sequence of events that led to the grounds for an investigation.

dependent failure. A failure which occurs due to interactions or failures within a system or due to interactions with or failures of other systems or equipment, or due to human error. Some examples of dependent failures are:

- Shared equipment dependencies;
- Functional dependencies;
- Common cause initiators;
- Physical interaction failures;
- Human interaction;
- Common cause failures.

human errors. Groups and/or families of attributes to characterize wrong human behaviour (understanding, intention, and action). Examples of such groups are:

- Violation (the person has a good understanding, but develops an intention not in compliance with that understanding);
- Mistake (the intention of the person is wrong because his/her understanding is not in compliance with the prescribed task);
- Slip (the intention was good, but the action is wrong).

human factors. A general term summarizing the various aspects of human behaviours in working conditions, including the behaviour itself and the factors important to understand the behaviour. This includes cognitive, ergonomic, technical and organizational factors.

human performance. The capabilities and characteristic behaviours of human beings in complex or stressful task environments such as nuclear power plant engineering, operation, and maintenance. Deficiencies in human performance (including licensed operators, other plant personnel, and contractor personnel) may degrade the defence in depth.

mistake. A mistake is an intended action resulting in an undesired outcome in a problem-solving activity: a person made a wrong action because he did not understand the system, the procedure, the specific context, the prescribed task, etc.

operating experience. A valuable source of information for learning about and improving the safety and reliability of nuclear installations. It is essential to collect such information in a systematic way that conforms with agreed to reporting thresholds for events occurring during the design, construction, commissioning, operating, and decommissioning phases of nuclear power plants to enhance safety. This includes information on deviations from normal performance by systems and by personnel, which could be precursors of events.

recovery actions. Activities to terminate the event and to bring the plant to a safe state.

slip. A slip is an unconscious unintended action or failure to act, resulting from an attention failure or a memory failure in a routine activity.

NOTE: Definitions in the previous version of the guidelines that are provided in the IAEA Safety Glossary [6] are listed below:

- *Accident precursor;*
- *Anticipated operational occurrence;*
- *Authorized limit (prescribed limit);*
- *Barrier;*
- *Common cause failure;*
- *Defence in depth;*
- *Direct cause;*
- *Diversity;*
- *Event;*
- *Failure;*
- *Management system;*
- *Normal operation;*
- *Observed cause;*
- *Operation;*
- *Operational limits and conditions;*
- *Protection system;*
- *Quality assurance;*
- *Redundancy;*
- *Regulatory body;*
- *Residual heat;*
- *Root cause;*
- *Safety (nuclear safety);*
- *Safety culture;*
- *Safety function;*
- *Safety system;*
- *Screening;*
- *Structures, systems and components (SSCs).*

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