IAEA
International Atomic Energy Agency

# IRS Guidelines

## Joint IAEA/NEA International Reporting System for Operating Experience

IAEA

AEN NEA

Vienna, March 2010

Services Series 19

# IAEA SAFETY 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, and also general safety (i.e. all these areas of safety). The publication categories in the series are **Safety Fundamentals, Safety Requirements** and **Safety Guides**.

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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 P.O. Box 100, A-1400 Vienna, Austria.

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# IRS GUIDELINES

# JOINT IAEA/NEA
# INTERNATIONAL REPORTING SYSTEM
# FOR OPERATING EXPERIENCE

**Services Series No. 19**

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

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# FOREWORD

The International Reporting System for Operating Experience (IRS) is an international system jointly operated by the International Atomic Energy Agency (IAEA) and the OECD Nuclear Energy Agency (OECD/NEA).

The fundamental objective of the IRS is to contribute to improving the safety of commercial nuclear power plants which are operated worldwide. This objective can be achieved by providing timely and detailed information on lessons learned from operating and construction experience at the international level. This information could be related to issues and events that are related to safety.

The purpose of these guidelines is to describe the system and to give users the necessary background and guidance to enable them to produce IRS reports meeting a high standard of quality while retaining the effectiveness of the system expected by all Member States operating nuclear power plants. As this system is owned by the Member States, the IRS Guidelines have been developed and approved by the IRS National Co-ordinators with the assistance of both Secretariats (IAEA/NEA). The IAEA officer responsible for this publication was X. Bernard-Bruls.

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## *EDITORIAL NOTE*

# I. BACKGROUND OF THE IRS

I.1. At the end of 1978, the OECD Nuclear Energy Agency (NEA) took the initiative to establish an international system for exchanging information on safety related events occurring in operating nuclear power plants. In March 1979, the accident at Three Mile Island (TMI) provided further impetus to the development of an effective international operational experience feedback process.

I.2. 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, 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 Chernobyl in April 1986 resulted in further recognition by regulatory bodies and agencies of various nations around the world of the importance of an effective event reporting and operating experience exchange system.

I.3. With the creation of the first comprehensive database on the IRS, Advanced Incident Reporting System (AIRS), in 1995, the responsibility of processing and reviewing reports (including quality checking) was transferred to the IAEA. A large number of topical and other studies have been produced by the IAEA over the years. Topical studies constitute a major component of IRS related activities. Such studies are intended to provide the basis for generating in-depth evaluations and to identify topical or generic issues for wider consideration. These issues begin with a national assessment by a country that accepts a lead role, and is then studied in depth by experts at the international level, when warranted.

I.4. In 1996, the Convention on Nuclear Safety 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."

There are also clear guidelines regarding national reports under the Convention on Nuclear Safety.

Moreover, international operating experience feedback can only be valuable if at the national level the appropriate arrangements have been made.

I.5. In 2006 the web based IRS was created to facilitate efficient data input and report availability. Each IRS report becomes part of this web based system. Passwords are provided to users who are officially registered and appropriate levels of access are allocated to individuals, thus ensuring a high level of security. Once a new report is posted on the web based IRS, the users are automatically informed by email. With the creation of the web based system, easy access to the information was expanded to utilities and plant staff, and the need for CD distribution and hard copies was eliminated.

I.6. 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'. The system will retain the term 'IRS'.

## II. OBJECTIVES AND SCOPE OF THE IRS

II.1. The main objective of the IRS is to ensure that feedback of operating experience gained from nuclear power plants worldwide on safety related events is widely shared amongst the international nuclear community to help prevent occurrence or recurrence of serious events. IRS provides a platform for the collection, processing and effective dissemination of construction, operating and decommissioning experience information among Member States. IRS not only ensures the distribution of the information but also implements measures to make the information useful, understandable and easily retrievable, in order to make users aware of lessons learned and any corrective actions identified.

II.2. Events and operating and construction experience information reported to the IRS should be of safety significance for the international community in terms of causes and lessons learned. .

For ease of understanding in the use of these guidelines, the terms 'events', or 'events/issues' are intended to mean any events, issues, and operating experience information, such as good practices, lessons learned or other findings. Additionally whenever the term operating experience is used in these guidelines, it is intended to include the construction, operation and decommissioning phases of a nuclear power plant.

II.3. 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 events and operating experience information to be reported to it.

II.4. The information should be provided to the IRS in a timely manner; indeed, early information on significant events in a country can assist in avoiding the same problem in other countries; this is an important feature of the system.

II.5. As there are differences in design, construction, and operation of nuclear power plants, IRS has to provide sufficient detail to highlight the wider relevance of the operating experience to the recipient. Therefore, an IRS report should provide detailed technical, organizational and human factors information on root causes, safety significance, lessons learned, and corrective actions.

II.6. Since the IRS focuses on significant events important for the international community, it should be viewed neither as a source for statistical studies nor for component reliability studies. In addition, the system is designed for specialists of the nuclear community as a source of detailed information on analysis and lessons learned from operating experience, as opposed to a simple description aimed at the public (note: the International Nuclear and Radiological Event Scale (INES) information service is designed for the media and public information).

## III. IRS MEMBERSHIP

III.1. It is assumed that a Member State which intends to participate in the IRS:

- has effectively embarked on a nuclear power programme;
- has established a regulatory body with the appropriate authority for regulating the safety of nuclear power plants;
- has established a national operating experience feedback system (in accordance with the IAEA Safety Guide NS-G-2.11: A System for the Feedback of Experience from Events in Nuclear Installation);
- has given an appropriate organization the responsibility for exchange of information on operating experience with IRS;
- is a contracting party to the Convention on Nuclear Safety.

III.2. If any of the above assumptions are not valid for a particular member state, then the IAEA would consider proposals from the Member State to clarify the interpretation of the IRS Guidelines without compromising the principles contained in the document. After reaching an agreed interpretation, the IAEA would inform all participating countries of the outcome of this agreement.

III.3. A Member State that wishes to participate in the IRS shall 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 participant in the IRS when it is placed on the list of participants maintained by the IAEA. All participants shall designate a person (by position) in the organization (usually the regulatory body) to be responsible for the exchange of information on operating experience under the IRS. This person is hereinafter called an 'IRS National Co-ordinator'.

III.4 The IRS is based on the principle that each participant will provide timely information on its nuclear power plants operating experience so that it is available to all other participants. A Member State intending to participate in the IRS shall therefore commit itself to send information to the IRS in accordance with the arrangements set out in the these Guidelines.

III.5. A Member State that has effectively embarked on a nuclear power programme, and has committed itself formally to comply with the IRS requirements, should contact the IAEA. The purpose being to make arrangements to participate in the IRS and thus profit from the lessons already learned from the worldwide exchange of operating experience from nuclear power plants.

## IV. USE OF THE IRS

IV.1. The IRS is for use, within each Member State, by organizations professionally involved in the nuclear industry, such as:

- Safety regulators
- Utilities with planned or ongoing nuclear programmes
- Nuclear power plant staff
- Technical support organizations in the nuclear field

- Vendor companies in the nuclear field (design firms, engineering contractors, manufacturers, etc.)
- Research establishments and technical universities working in the nuclear field.

IV.2. The IRS focuses on safety related events/issues with potential for lessons to be learned internationally, in particular, precursors of serious events. The reports communicate to experts of other countries the results of the analysis carried out and the lessons to be learned. Thus, IRS provides an organized set of data, easily transferable to situations in other countries, allowing an efficient feedback process. Events or issues should be reported if they affect plant safety and/or show significant lessons to be learned.

IV.3. The IRS relies upon national operating experience systems and complements them by providing an international perspective. National regulatory bodies or technical support organizations should review operating information available in their own countries for potential use internationally, in the form of IRS reports.

IV.4. The IRS is an important source of information for regulators and their technical support organizations, providing them with insights on important international operating experience for use in their regulatory activities.

IV.5. Operating organizations and others mentioned above receive additional information with different perspectives (e.g. from World Association of Nuclear Operators), this information is considered by their own operating experience feedback systems. All of these sources of information on operating experience are important in helping to identify improvements to safety. In addition, vendor companies may be able to improve their design and manufacture of structures, systems and components (SSCs) important to safety by incorporating lessons learned.

Information on events, anomalies, situations and conditions usually starts at the plant level. It is generally communicated within the operating organization and then, in accordance with the relevant requirements, to the regulatory body, to other operating organizations and to research organizations, designers, contractors and other relevant parties.

IV.6. Through active participation and dissemination of operating experience reports, all Member States potentially benefit from the learning opportunities identified in these reports, to improve safety. Feedback takes many forms, and is commonly conducted through the IRS National Co-ordinators' meetings.

## V. COLLECTION AND DISTRIBUTION OF IRS INFORMATION

V.1. The web based IRS is a secure online system; it allows for report publication and access by registered users all year round, The generic user can view and search for reports. In addition to viewing and searching, the IRS National Co-ordinator can create reports. The IAEA verifies and approves the new reports which are then notified and made available to web based IRS users.

V.2. As a result of the detailed technical character of the information provided by participating countries, IRS reports are generally classified as 'Restricted' in order to encourage open and timely exchange of information among participants. This condition was accepted when the system was established and remains valid. Once an IRS report is transmitted to IRS National Co-ordinators, it is their responsibility to decide on its further distribution for official use within their country.

# VI. REPORTING

## VI.1. OPERATING EXPERIENCE INFORMATION TO BE REPORTED[1]

Events and other information (good practices, operating experience reports…) may be reported to IRS during any of the life cycle phases of a nuclear power plant, including manufacturing, construction, installation, commissioning, operation and decommissioning.

To meet the objective of the IRS, events/issues to be reported to the IRS should be selected according to the following general principles:

(i) Events/issues: there is an actual or potential significant reduction in the plant's defence in depth, for example:

1) Actual operating events, typically plant transients accompanied by equipment failures, human errors or other anomalous indications;

2) Actual failures of safety related systems, structures or components (sscs), or human errors, that may or may not have caused a plant transient;

3) Adverse safety conditions such as design weaknesses, degraded safety equipment or aging effects that could lead to failures of systems, structures or components;

4) External challenges to safety such as vulnerability to severe weather, flooding, high winds;

5) Organizational or human factor issues such as a degraded safety culture at a plant, high human error rates, weaknesses in the safety management/quality assurance system, inadequate procedures, inadequate training or inadequate control of contractors at a plant site.

(ii) The event/issue reveals important lessons that, if learned, will help the international nuclear community to prevent occurrence or recurrence of a serious event/issue in terms of safety, e.g. new research results or new safety analyses, showing a previously unknown weakness in a safety system, or issues with fuel integrity, reactor coolant system pressure boundary integrity, or containment integrity.

(iii) The event/issue is a repetition of a similar event previously reported to the IRS, but highlights new important lessons to be learned by the international community.

## VI.2. REPORTING CATEGORIES

In order to decide on the events/issues to be reported, the following categories should be considered:

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[1] Operating experience information does not mean only a single occurrence; it may include a series of events, lessons learned, good practices or findings important for the international community.

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

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

## VI.3. CONTENT, REPORTING TIME AND FORMAT OF THE REPORTS

VI.3.1. The IRS is intended to be not too formal in order to promote easy reporting and encourage open contacts among people responsible for operating experience feedback in participating countries.

VI.3.2. The reporting should provide the international community with:

- A narrative description of the event/issue (including the plant specific technical data necessary to understand all its consequences);
- A safety assessment;
- A cause analysis (explaining the direct and root causes and causal factors);
- The lessons learned and corrective actions taken.

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

VI.3.3. The lessons learned and actions taken should be clear and understandable to the international community to facilitate the assessment of the applicability to the situation in other countries.

VI.3.4. The report should also include a cover sheet with standardized information and an abstract giving the essential characteristics of the event/issue, as well as codes facilitating information retrieval (See Appendix C).

VI.3.5. The report should be provided in a timely manner, i.e. as soon as all the necessary information is available.

VI.3.6. For events with particularly important lessons to be learned and/or the need for information transfer to other countries, a preliminary report should be submitted, consisting of a brief description of the event and all relevant preliminary findings. The preliminary report should be submitted as soon as practicable, preferably within one month after the event. This preliminary report is to be followed by a full report.

VI.3.7. For the more safety significant events/issues, it is important to understand what Member States have done to address the identified lessons learned. After receiving a safety significant report identified as ‘response requested’ by the IRS Advisory Committee, the IRS National Co-ordinator should, after appropriate review and processing in accordance with his/her organization’s requirements, provide a response, indicating actions taken in his/her country for this event/issue. Response statements may also be made by IRS National Co-ordinators for any other IRS report.

VI.3.8. The format recommended for the preparation of IRS reports should be used to the extent practicable. However, flexibility should be maintained for practical reasons, such as specific types of reports or different national requirements.

VI.3.9. The standard format and content of IRS reports may be considered for adoption into national systems for operational experience feedback in order to link national and international systems more efficiently.

VI.3.10. The IRS is operated in English. Whilst one of the other official languages of the Agencies may be used, Member States are encouraged to submit IRS reports in English in order to avoid undue delays.

The web based system, however, allows for IRS reports to be sent in their original language and to be included under ‘attachments’.

## VII. IRS OPERATION AND MANAGEMENT

VII.1 This section describes the role and relationship of the respective parties involved in the operation and management of the system. These are the participating countries, the IAEA and NEA Secretariats, the Technical Committee of IRS National Co-ordinators; the meeting of IRS National Co-ordinators to exchange experience on recent events in Nuclear Power Plants, the NEA Working Group on Operating Experience and the IRS Advisory Committee.

VII.2. ROLE OF PARTICIPATING COUNTRIES

VII.2.1. The viability of the IRS was originally based on the voluntary commitment of the participating countries. With the introduction of the Convention on Nuclear Safety, there is now an obligation on the Contracting Parties to share operating experience. The effectiveness of this sharing process depends on the quality of the selection and on the quality of the presentation of the information exchanged among countries. Therefore, the primary role of participating countries is to ensure that all operating experience with important lessons in terms of safety to be learned by the international community is reported in a timely manner.

VII.2.2. The effectiveness of the IRS also largely depends on its regular use. Therefore, the participants should promote, in their respective country, the use of the IRS. National Coordinators should also collect feedback within their country on IRS usage, to help improve and update the system.

VII.2.3. As users of the IRS, participating countries should keep full control over it. In particular they have to agree on its objectives, decide on improvements and modifications in reporting and on management of the database and related activities to be performed by both Secretariats.

VII.2.4. Participating countries shall designate an IRS National Co-ordinator to be responsible for receipt and distribution of information received from the IRS and for the transmission of information to the IRS. The role of IRS National Co-ordinator is tied to the role of the participating country: e.g. promoting exchange, conducting training, and providing feedback on the use of IRS information from his/her own country.

VII.2.5. The network of IRS National Co-ordinators can, via direct contacts, supplement the exchange of information. Participating countries should allocate sufficient resources to make these exchanges effective.

VII.2.6. It is also the responsibility of the Member State to periodically review how the country makes use of the system and to identify any weaknesses to report back to the Technical Committee of IRS National Co-ordinators.

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 Co-ordinator.

The IRS National Co-ordinator should:

- Demonstrate ownership of the system, by promoting the use of the system and showing leadership at the national and international level;
- Ensure control of the quality of the reports so that the information is sufficiently comprehensive in a way commensurate with the timeliness of reporting (preliminary or final);
- Be given the necessary authority and tools to openly communicate to the system any information that could be of benefit to the international community.

VII.2.7. Dissemination of information:

The information should be accurate, complete, understandable, user friendly, and easily retrievable.

Special efforts should be made by the authors of the reports to ensure information is understandable by most users. This also includes the avoidance of acronyms and slang and the use of broadly accepted terms.

Dissemination of information is more effective if:

- All Member States are committed not only to use the system but also to report 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 is reported proactively and in a timely manner.
- Information shared is easily understandable.

VII.3. ROLE OF THE AGENCIES

VII.3.1. The IAEA and NEA Secretariats provide the legal framework, the infrastructure and the technical support to operate the IRS. Both Secretariats co-ordinate their efforts so as to make sure that activities sponsored by both Agencies are not duplicated and meet the expectations of participating countries.

VII.3.2. The primary role of the IAEA is to operate the IRS, to provide technical support to participating countries for 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/issues reported to the system by participating countries;
- Translate IRS reports from official languages to english, if necessary;
- Review the reports and check their consistency and give feedback to the national co-ordinators, if needed;
- Request follow-up information as required;
- Make sure that all information received is periodically distributed in the manner requested by participating countries;
- Compile biannual highlights report;
- Establish, operate, maintain and up-date the web based system;
- Request participating countries to report based on other information (INES, press information);
- Produce topical studies;
- Perform other consultant services regarding the IRS.

VII.3.3. The Secretariats periodically produce a report on nuclear power plant operating experiences, called the 'Blue Book', generated from the IRS event reviews.

VII.3.4. The Secretariats organize on an annual basis meetings of IRS National Co-ordinators, alternating the locations of these meetings.

VII.4. ROLE OF THE MEETINGS OF IRS NATIONAL CO-ORDINATORS

VII.4.1. Role of the Technical Committee of IRS National Co-ordinators

The Technical Committee of IRS National Co-ordinators regularly reviews the status of IRS operation and management and provides recommendations to both Agencies regarding implementation on actions to be taken to correct problems, if necessary, and implement any action agreed by the Technical Committee.

VII.4.2. Role of the Meeting to Exchange Experience on Recent Events in nuclear power plants

National co-ordinators meet each year to review the information received and to exchange information on recent unusual events which have occurred in their respective countries. The Secretariats select events for in-depth presentation to be discussed.

## VII.5. ROLE OF THE NEA WORKING GROUP ON OPERATING EXPERIENCE

The mandate of the NEA Working Group on Operating Experience (WGOE) is the following:

The working group shall constitute an international forum for the exchange and analysis of operating experience for the determination of safety issues from a regulatory viewpoint. It will also identify safety issues that derive from operating experience for consideration by the Committee on the Safety of Nuclear Installations (CSNI) in relation to additional research. WGOE, with the agreement of the Committee on Nuclear Regulatory Activities (CNRA), will plan its work to promote improvements in nuclear safety at nuclear installations, with primary focus on Nuclear Power Plants.

To this end, the working group shall:

1. Exchange, analyse and provide expert insights and knowledge from relevant operating experience to reach timely conclusions on lessons learned, trends, and effectiveness of responses. Prepare reports, disseminate conclusions and sponsor international workshops on use of operating experience for improving nuclear safety.

2. If necessary, provide feedback to IRS Advisory Committee on areas for system improvement in the context of WGOE objectives.

3. Compare, and where possible, benchmark international practices and methodologies applied by Member countries in the assessment and use of operating experience.

4. Maximise the benefits of co-operation with existing NEA working groups and other international organisations (e.g. IAEA, EC, WANO, etc).

## VII.6. ROLE OF THE IRS ADVISORY COMMITTEE

VII.6.1. The mandate of the IRS Advisory Committee 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 the countries in keeping effective control over the system and advises the Secretariats in providing effective technical support.

VII.6.2. The IRS Advisory Committee 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, as well as the Technical Committee of IRS National Co-ordinators. The Advisory Committee is clearly neither intended to make decisions regarding the operation of the IRS, nor to carry out technical analysis.

VII.6.3. More specifically, the IRS Advisory Committee meets as a minimum once a year:

1 - To review IRS objectives and guidelines. Any significant changes affecting the operation of the IRS would require approval of both Agencies through the Technical Committee of IRS National Co-ordinators.

2 - To provide advice and recommendations to, and for review and approval by the Technical Committee of the IRS National Co-ordinators on the operation, maintenance and improvements to the IRS.

3 - To examine the quality and content of the incoming IRS reports with a view of improving their use for evaluation and analysis. Technical analysis would not be within the scope of this Advisory Committee.

4 - To identify additional activities that can be performed based on IRS information, and to identify common supporting activities which would need to be performed to enhance the effectiveness of the IRS.

5 - To examine reporting practices to help ensure that important issues are reported to the IRS.

VII.6.4. Usually the IRS Advisory Committee forwards its advice and recommendations with regard to IRS operation and management to the Technical Committee of IRS National Co-ordinators for their review and approval. In case of a specific request, the IRS Advisory Committee will give its view and advice to an Agency or participating country.

VII.6.5. The IRS Advisory Committee is constituted of elected members, three from OECD countries, two from non-OECD countries, and one representative of each Agency. The term of the elected members is three years.

# APPENDIX A

## Procedure for Preparation of IRS Reports

### Contents

# 1. INTRODUCTION

The objective of this procedure is to help the user to prepare an IRS report on an event so that important lessons learned are most 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.

Practically, it is recognized the compilation of the information for an 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 human performance is dominant to draw lessons, more detailed guidance on the specific information that should be supplied is spelled out in the procedure. This guidance differs somewhat 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 web based system is designed to assist IRS National Co-ordinators in sending IRS reports and it provides the structure for the preparation of a report. Information on how to use the web based IRS is described in the IRS manual and is not part of these guidelines.

In order to facilitate the use of the present procedure, the basic process it describes has been summarized in a flowchart.

# 2. SELECTION OF EVENTS/INFORMATION FOR REPORTING

A. The first step is to determine whether the event/issue has an actual or potential impact on safety.

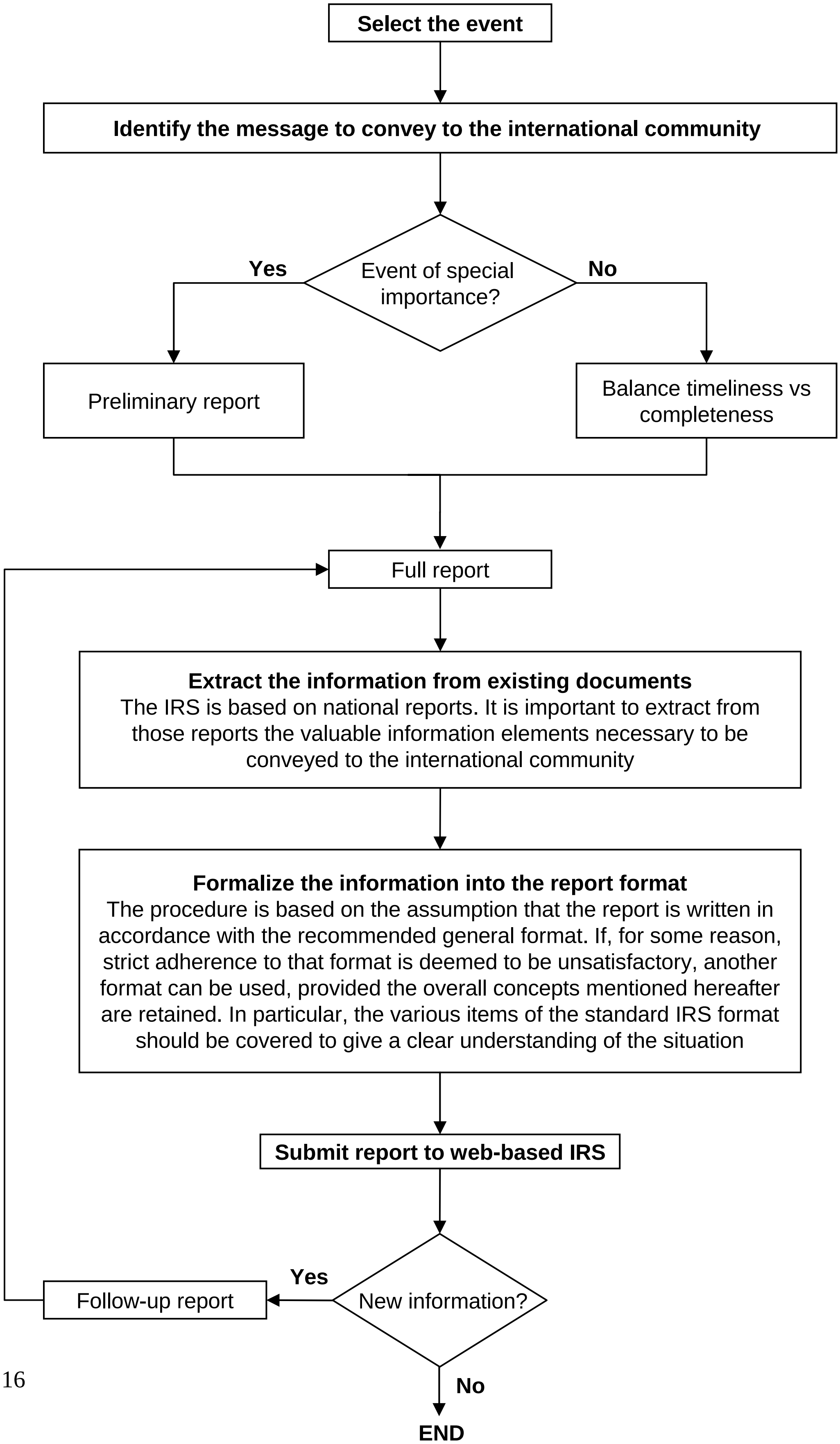
   Reporting categories are described in Appendix B, which sets out general definitions, cases and examples of such events.

B. The second step is to determine whether important lessons can be learned by the international community in order to prevent the occurrence or recurrence of a serious event/issue in terms of safety.

   Events/issues which are the repetition of similar events/issues reported to the IRS may still convey new lessons learned to the international community.

C. If the event/issue satisfies criterion (A) and (B), it should be reported to IRS.

D. Operating experience often includes precursors or contributors to more significant events/issues. Consequently, the reporting should not be limited only to points A–C above but also to lower level events and issues (problems, good practices and corrective actions) that contain lessons which may be useful for others.


Select the event
Identify the message to convey to the international community
Event of special importance?
Yes
No
Preliminary report
Balance timeliness vs completeness
Full report
Extract the information from existing documents
The IRS is based on national reports. It is important to extract from those reports the valuable information elements necessary to be conveyed to the international community
Formalize the information into the report format
The procedure is based on the assumption that the report is written in accordance with the recommended general format. If, for some reason, strict adherence to that format is deemed to be unsatisfactory, another format can be used, provided the overall concepts mentioned hereafter are retained. In particular, the various items of the standard IRS format should be covered to give a clear understanding of the situation
Submit report to web-based IRS
New information?
Yes
Follow-up report
No
END

## 3. MESSAGE TO BE CONVEYED

As soon as the event/information to be reported has been selected, the message to be conveyed to the international nuclear community has to be clearly identified.

## 4. TYPE OF REPORT

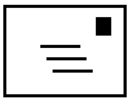
Depending on the significance of the event/information, a preliminary report may be useful (go to Section 5 immediately below), otherwise a main report shall be sent (go to Section 6). As soon as the necessary information is available, a main report shall supplement the preliminary report.

There are two types of main reports:

1. A standard main report, associated with a single event/issue;

2. A generic main report, associated with a set of events/issues related to each other, and produced to focus on common lessons learned from those events/issues.

If, after the publication of a main report, new relevant information becomes available, an update of the previous main report should be prepared (follow-up report, follow-up generic report).



Do not wait until exhaustive information on the event/issue is available in order to prepare and send a full report. Send the report whenever sufficient material is available. Judgement should be exercised to balance completeness of information and timeliness of reporting.

Consider that, if new elements become available later, you still have the opportunity to send a follow-up report.

## 5. PRELIMINARY REPORT

The preliminary report should summarize the information available at the time the report is prepared with emphasis on the lessons for the international community. It should include a short description of the event/issue, the preliminary safety evaluation and the short term actions taken, if available.

If there are no lessons found after the creation of a preliminary report a main report is not needed but such information in the abstract of the report will be indicated.

## 6. MAIN REPORT

### 6.1. IDENTIFICATION OF THE NECESSARY INFORMATION

From the available information on the event/issue, extract and sort the following items (if available):

A. General data, such as plant name/unit and date/time of the event.

B. Plant conditions before the event and methods of event discovery (in case of a deficiency).

C. The factual event sequence as observed, possibly including the observed degradations or malfunctions of systems and the reasoning or reaction of people at the time, with their impact on the event sequence. Identify clearly the observed cause/consequence relationships.

D. A consequence analysis, in order to determine whether or not some aspects of the event are indicators of indirect problems/weaknesses which, under other circumstances, could also lead to a safety significant event or a serious accident.

E. An analysis of the event, identifying the root causes and causal factors, the impact(s) on safety and the investigative and corrective actions taken.

F. The causes and corrective actions should address technical as well as human and organizational factors/aspects/deficiencies. If possible, provide an indication of how each given deficiency has been corrected.

G. Assessment by the regulatory body, to the extent possible.

H. If the event description and/or analysis require additional plant related information be made available to readers in order to provide a better understanding, then provide the necessary information on plant features.

## 6.2. FORMALIZATION OF THE COLLECTED INFORMATION INTO THE EVENT REPORT

In order to follow the recommended general format for IRS reports, the following process should be applied.

### 6.2.1. 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, practices, procedures and/or organizational characteristics that influenced the event, especially if these are known to be unique to the plant or country. For a better understanding, descriptive names for equipment should 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 should be provided:

**A. Situational aspects**

(a) Plant conditions prior to the event.

(b) Operating modes or testing conditions.

(c) Equipment status.

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

(d) Plant staff involvement (Appendix C, Section 5.3).

(e) Type of activity at the time of the event (Appendix C, Section 5.4).

(f) Characterization of the personnel & individual task related work practices (Appendix C, Sections 5.5.1, 5.5.2 and 5.5.10).

(g) Characterization of the working conditions (Appendix C, Sections 5.5.4 and 5.5.5).

(h) Any other relevant organizational aspects (Appendix C, Sections 5.5.3 and 5.5.6 to 5.5.9, 5.5.11 and 5.6).

**B. Chronological information**

(a) Chronological information indicating relevant time-scales.

(b) Identification of failures and successes in responding to the event, including any that occurs during the recovery actions phase (Appendix C, Section 9).

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

(c) 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 operators 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.

(d) Detection and diagnosis activities, including delays encountered.

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.

(e) Any humans errors involved.

Include errors of commission as well as omission, and what would have been the correct action(s), if known.

(f) Intra- and extra-team communication aspects.

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

Previously related events or precursors should be indicated.

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

### 6.2.2. Prepare the safety assessment

Address here the actual and potential consequences of the observed problems. In particular, a discussion of the barriers which were broken by the observed deficiencies and the effective barrier that terminated the event should be included.

When relevant, safety aspects related to human performance should be included.

If the assessments by the licensee and by the regulatory body are different, this should be indicated.

If an event/issue is recurrent, the analysis should include an explanation why the already implemented corrective action(s) has not been effective.

### 6.2.3. 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 which immediately produced the event) should answer the question ‘how did it happen?’, identifying the technical, human and organizational deficiencies (Appendix C, Section 5). It should also give, to the extent possible, the results of the analysis identifying the nature of failures or errors (Appendix C, Section 8).

Human factors investigations have shown that people’s performance is strongly dependent on the context.

Interaction between operators and plant systems creates a dynamic context where operators receive plant information. Using training and procedures, most of the operating conditions are easily managed. Overconfidence in ones knowledge of systems (machine) can affect operator actions which, consequently, could affect the plant’s 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 work environment, communication, the effectiveness of shift turnover, among others. The event description should 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 should be considered during event analysis.

For events where human performance plays a significant role, the following information should be provided where available: include the type of observed human errors (Appendix C, Section 5.1.10) which contributed to the initiation of the event or affected in a direct way the operator or system response to the event. The human performance related causal factors and root causes are addressed in Appendix C, Sections 5.5 and 5.6. In addition, plant staff involved in the event has also to be identified (see 5.3).

The report should also 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 in order to improve the quality of the process or product.

A presentation and discussion of 'root causes' should follow. These causes are fundamental causes that, if corrected, should 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?'.

If possible, include an event and causal factor chart to illustrate the analysis results.

### 6.2.4. Prepare the lessons learned and corrective actions

It is important to address the root causes and causal factors identified by investigation and to relate the causes to the corrective actions proposed.

Describe the set of corrective actions taken by the utility to address the observed technical, human and organizational deficiencies. The priority of the various corrective measures should also 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 written
- Interim corrective actions which 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 recurrence. These actions are most important to prevent events from ever happening again.

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

(a) Changes in attitudes or habits of persons or groups

(b) Changes to the initial/continuing training programmes. Indicate what was lacking in terms of knowledge and know-how.

(c) Changes to procedures or new procedures

(d) Organizational changes

(e) Improvement in ergonomics

(f) Hardware modifications which influence man-machine interaction

(g) Conduct of re-training to correct specific knowledge deficiencies.

Describe also any specific actions taken by the regulatory body in response to the event.

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 should be practical and applicable to other nuclear power plants, and should be consistent with the basic safety message to be conveyed.

### 6.2.5. 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 should give, in a concise form (maximum 25 lines), a brief description of the event, its safety relevance, its causes, the lessons learned and the corrective actions taken.

### 6.2.6. Choose a title

The title should be a very short characterization of the event, emphasizing its most significant features.

### 6.2.7. Prepare the cover page and codes

It should be noted here that a report can be directly submitted into the web based system using the various chapters. Alternatively, text can be uploaded from any word file into the various chapters.

#### 6.2.7.1. Cover page

Fill in the cover page information to identify the event.

**Report type**

Indicate the report type as described in Section 4.

**Title**

Fill in the title as determined in Section 6.2.6.

**Plant Name and code**

*Standard full reports*

In any case the national co-ordinator can only input information for/from her/his own country and can select the plants displayed for this county from a drop down menu.

*Generic reports*

In the case of a generic report the web based system allows for inputting of one or more plant names for a given country or no plant name at all.

*Follow-up reports*

The web based system allows for updating the already existing information.

**Date of event**

*Standard full reports*

The date of event can be filled.

*Generic and follow-up generic reports*

In the case of generic reports the web based system allows for a specific date, a date range, a specific month, a month range, a specific year and a year range.

#### 6.2.7.2. Codes

As the codes are provided for retrieval purposes, they must reflect the event conditions, the observed phenomena and the problems encountered.

More than one code can be selected under each category. Naturally, the more detailed the code, the better. However, if a detailed code is selected, its parent codes should not be selected.

Refer to Appendix C, Sections 1 to 9 for the definitions and usage of the available codes.

### 6.3. FORMALIZATION OF THE COLLECTED INFORMATION

The web based system allows for documents and figures to be uploaded in the following formats: .doc, .rtf, .tif, .jpg, .jpeg, .pdf etc…

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

## 7. FOLLOW-UP REPORT

A follow-up report may be necessary when new or different information is identified that would be helpful in the understanding of the overall event and the effectiveness of its associated corrective and preventive actions.

When corrective actions prove to be insufficient a follow-up report should 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 B

## IRS Reporting Categories

**Contents**

## INTRODUCTION

This appendix discusses the categories of operating experience information (events, issues, etc.) to be reported. It provides background information on the reasons for their selection as well as a general description and examples of relevant events. As stated in the main text, the categories are intended to provide a basis for identifying safety related events and other information to be reported to the IRS. It is important to note that a report may be prepared not only because an event has occurred, but also because lessons learned have been identified. The examples given here are intended to illustrate typical events and other information to be reported to the IRS under each category. It should also be noted that complex events may fall into more than one category. 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.

The categories are:

1. Unanticipated releases of radioactive material or exposure to radiation
   - 1.1. Unanticipated releases of radioactive material
   - 1.2. Exposure to radiation that exceeds prescribed dose limits for members of the public
   - 1.3. Unanticipated exposure to radiation for site personnel
2. Degradation of barriers and safety related systems
   - 2.1. Fuel cladding failure
   - 2.2. Degradation of primary coolant pressure boundary, main steam, feedwater line or other high energy systems
   - 2.3. Degradation of containment function or integrity
   - 2.4. Degradation of systems required to control reactivity
   - 2.5. Degradation of systems required to ensure primary coolant inventory and core cooling
   - 2.6. Degradation of essential support systems.
3. Deficiencies in design, construction (including manufacturing), installation and commissioning, operation (including maintenance and surveillance), safety management/quality assurance system, safety evaluation and decommissioning
   - 3.1. Deficiencies in design
   - 3.2. Deficiencies in construction (including manufacturing), installation and commissioning
   - 3.3. Deficiencies in operation (including maintenance and surveillance)
   - 3.4. Deficiencies in safety management/quality assurance system
   - 3.5. Deficiencies in safety evaluation
   - 3.6. Deficiencies in decommissioning
4. Generic problems of safety interest
5. Consequential actions taken by the regulatory body

6. Events of potential safety significance
7. Effects of unusual events of either man-made or natural origin
8. Other findings and operating experience information.

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

Releases of radioactive material from the site directly impact the environment and may affect the public. Exposure of personnel directly affects the plant staff. 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 in excess of limits to plant staff and the public are also addressed in this 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 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) During attempts to reduce gas circuit pressure in a gas cooled reactor by blowdown, two discharges of reactor cooling gas occurred in the vacuum pump room (VPR) through a sampling valve on the auxiliary gas system which was inadvertently left open. The VPR is situated within the reactor building which contained the releases. The event resulted in limited radiation levels in the VPR and limited personnel doses. The event identified generic weaknesses in the design and administrative control procedures. It led to generic design recommendations for open ended valves like the sampling valve affected. Administrative control procedures and instructions related to the surveillance and locking of these valves were strengthened.

(b) A boiling system water reactor event involved a minor unmonitored release of radioactive material to the environment. The event occurred during reactor restart after an outage. Due to improper valve manipulations, concentrates from a waste concentrator were drawn into the auxiliary boiler system, which were then vented to the atmosphere. Radioactive contamination occurred on nearby building exteriors, as well as adjacent ground areas. The event highlighted previously unidentified routes for release of radioactive materials to the environment which could not be monitored by the plant equipment. In addition, procedural inadequacies in the management of radwaste operation were discovered.

(c) While transferring the charge of deuterated resin from resin tank to ion exchange column, a rubber hose pipe connected at the bottom of the tank cracked, resulting in heavy water spillage. Later inspections showed that the hose pipe was of poor quality. It was installed due to the unavailability of a hose pipe of required specifications. The lesson learned was the need for stringent quality control, even in auxiliary systems handling radioactive material.

## 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 should be reported to the IRS.[2]

## 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) Based on indications, personnel observed increased activity of steam at the fuel channel outlet which indicated leaking fuel rods. During fuel assembly discharge operation, radiation monitoring system alarms actuated periodically in the reactor hall. The investigation showed that fuel fragments were found outside the fuel shipping cask. The checking of personal dosimeters of eight workers who took part in the work revealed that two persons had external exposure doses exceeding the annual limit. The root cause of the event was inadequate dose control.

(b) Preparation for refueling was being performed and the reactor cavity was being filled with water. An examination of the sump area was planned by looking through the access door only. A worker was provided with a key to the sump area and was cautioned not to enter the sump area. The task was delayed until the next shift. The key was passed on but the caution was not. Two workers entered the sump area in spite of the warning on the door. One worker received 13 mSv (whole body) and the other received more than 2 mSv. This event revealed deficiencies in establishing appropriate controls including administrative control of keys, adequate training of related personnel and specific procedures for entry into potentially hazardous radiation fields. The event is one example of a recurring problem.

(c) An apparently empty shipping container was sent to the decontamination area, where decontamination personnel established that the dose rate from the container had been underestimated. Their personal dosimeters alarmed, indicating higher than expected radiation fields. It was discovered that a small piece of metallic wire, probably from a neutron flux detector assembly, having a contact dose rate in excess of 10 Sv/h, was responsible for the unexpectedly high activity from the container. The event draws attention to inadequacies of administrative controls.

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

# 2. CATEGORY 2: DEGRADATION OF BARRIERS AND SAFETY RELATED SYSTEMS

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

(1) Control of reactivity.

(2) Cooling of radioactive material.

(3) Confinement of radioactive material.

Each of these safety functions is in turn ensured by safety systems which are usually provided with redundancy and the availability of which 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.

## 2.1. FUEL CLADDING FAILURE

The fuel cladding is the first barrier which 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 should be reported, especially when generic implications ensue. Reportable fuel failures are not limited to reactor power operation. Events which occur during fuel handling operations (i.e. refueling, including handling operations in the reactor or in a storage pool), and which result in 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 refueling 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 must 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 utility and the safety authority has ordered all boiling water reactors owners to apply, as a function of burnup, additional penalties in calculation of critical power ratios.

(b) Following a routine surveillance of the reactor coolant activity at a pressurized water reactor, the plant was shut down because of fission product activities exceeding plant technical specification limits. Inspections revealed that some rods in about 20 fuel assemblies presented indications of cladding damage. Visual inspection revealed indications of fretting in the spacer grid regions. The observed fuel damage was caused by spacer grid spring tension reduction due to manufacturing deficiencies. As a result of this event the fuel manufacturer has improved the design of the spacer grids.

(c) Problems during an infrequent maintenance activity on the pressure and calandria tubes, required a non-standard defueling of a fuel channel of a pressurized heavy water reactor in 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 cleanup, 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.

## 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 should be reported as well, if important lessons are to be learned from these events.

Other high energy systems may include systems such as S/G 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 pressure transient of the reactor coolant system of a pressurized heavy water reactor, two bleed condenser relief valves lifted. After the initial opening, one of the relief valves began to chatter. This led to damage of the valve and the valve piping, resulting in a major reactor coolant leakage into the reactor building sump. The event occurred after implementation of a design modification which was not adequately reviewed. An improved bleed condenser overpressure relief system has been implemented as a result of this event. All other relief valves in the reactor coolant system and connected auxiliary systems have been reviewed to ensure that similar inadequate configurations are not present. In addition, the plant design modification process has been reviewed in order to prevent a reoccurrence of similar process deficiencies.

(c) 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. Lessons learned are that pressurized water reactor SGs may be subjected to thermally induced mechanical loads which 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.

## 2.3. DEGRADATION OF CONTAINMENT FUNCTION OR INTEGRITY

Most reactors are enclosed by structures to contain radioactivity, should this be released from the primary coolant system. The containment function, which may include primary and secondary containment, is the ultimate barrier to prevent release of radioactive material to the environment. Containment structures must 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, leaktight containment penetrations, pressure suppression systems) as well as active components (e.g. containment isolation valves, containment spray and cooling systems, ventilation systems). In shutdown conditions containment integrity may be required too, 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 ruptured. The rupture, which was located between the containment penetration and the first isolation valve outside the containment, degraded the containment 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 should be leaktight during accidents.

(b) At the start of the annual outage, an inspection of the reactor containment revealed that a door between the dry well and the wet well was not closed. This jeopardized the pressure suppression function in case of a loss of coolant accident. Checking procedures for dry well/wet well leakage, performed before startup, were found to be ineffective to confirm the leaktightness of the door. The event highlights the meticulous attention that must be paid even to conventional components which are part of non-single failure proof passive safety systems. Corrective actions taken by the utility included modification of the door lock mechanism, provision of position marks to the door handle as well as door and handle position indications in the control room.

(c) 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 authority.

(d) 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 case 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 in order to verify design characteristics. The partial tests which had been performed during plant commissioning did not allow to discover the observed design deficiencies.

## 2.4. DEGRADATION OF SYSTEMS REQUIRED TO CONTROL REACTIVITY

Various systems are provided for reactivity control to bring the reactor from cold shutdown to full power conditions and vice-versa, to compensate for fuel burnup and to shut the 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 should 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 in order to increase the reliability of the reactor scram system, involving improvement of 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) A discrepancy found between theoretical critical boron concentration and measured values during core physics testing of a pressurized water reactor during startup revealed a failure of the boron concentration measurement device. The contraction of the standard alkali solution, which is used in the analytical instrument to determine boric acid concentration and which was recently replaced, had not been entered into the instrument memory. This common cause failure resulted in invalid boron concentration measurements in the unit where the problem was discovered as well as in the sister unit which was at full power operation. Four tanks in total had boron concentrations which did not comply with technical specifications. This event highlights that improper use of measuring instruments, even when adequately standardized, may result in measurement errors which are not easily identified and have a large potential for common mode failures.

(c) 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’s actions were unsuccessful in controlling oscillations, which continued for 50 hours until the reactor was shut down. Investigation of the safety implications revealed that large flux tilts as 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 station procedures and operator training in the area of 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.

(d) 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 plants of the same utility. The loading errors had resulted in the reactor protective system being impaired in a few cases.

## 2.5. DEGRADATION OF SYSTEMS REQUIRED TO ASSURE PRIMARY COOLANT INVENTORY AND CORE COOLING

Systems are provided which ensure in normal and transient plant 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 may 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 may occur which warrant the existence of redundant or diverse emergency core cooling systems (such as high and low pressure injection systems and core spray systems), providing core inventory supply and cooling in case of loss of coolant accidents. Partial or total failure of such systems may result in large fuel cladding failure, fuel meltdown and important 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 should be reported under this reporting category.

Examples:

(a) After placing the gas stripping system installed on the auxiliary feedwater (AFW) system of a pressurized water reactor in a recirculation mode in order to reduce the oxygen concentration of the auxiliary feedwater tank, water temperature started to increase. Because of delayed operator action on temperature alarms, 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 engineered safeguard system.

(b) During startup of a pressurized water reactor after refueling outage, the high pressure safety injection (HPSI) isolation valves were found closed on the three redundant trains with power to the valve operators tagged out. This was discovered three days after the primary coolant temperature exceeded 180°C, for which the technical specifications require full availability of the HPSI system. The closure of the valves occurred due to tests that were performed without checking the proper valve position for startup. The event highlights the potential for safety relevant degradations of system availability due to inappropriate test scheduling and test procedure deficiencies.

(c) Several events addressed the potential for overpressurization 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 improvement and/or standardization of maintenance and surveillance procedures and upgrading in training and qualification of plant personnel.

## 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 should be reported, if important lessons can be learned. Degradation of fire protection systems, which may 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 inoperability of multiple emergency diesel generators (EDG) due to degradation of fuel oil delivery system by deterioration of the quality of fuel oil stored at the site. The event highlighted the surveillance programme implemented at the plants to be 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 plant was in cold shutdown for refueling, 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. For that purpose, the battery had been disconnected from the busbar. During attempts to recover the situation, a total loss of power supplies for control system occurred for about 15 minutes, although core cooling remained uninterrupted. The event sequence revealed the difficulty to diagnose the situation and bring the plant back to normal in the unusual electrical supply conditions. Lessons learned from this event include the importance of batteries to ensure continuous power supply, especially in connection with maintenance and periodical testing practices. Also the need was identified to improve methods and instructions for correcting partial loss of electrical sources.

(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. Primary coolant temperature rose about 40K, reaching a maximum temperature of 90°C over a 1.5 hour period, while plant staff diagnosed and corrected the problem. It was also recognized as actions were taken to improve the reliability of air operated valves in the shutdown cooling system. The plant event procedures were primarily directed to inoperability of pumps and heat exchanger, rather than to 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.

## 3. 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 nuclear power plants. The most important means to maintain this safety level 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 should be reported in this 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 plant safety. All such cases including material compatibility, the degradation due to environmental or operating condition, computational errors, etc. should be reported under this category.

Examples:

(a) Cracks of various sizes were found in a large number of bolts which hold the core grid. They occurred due to stress corrosion cracking as a result of original material sensitivity and initial overtightening. This event demonstrates a major material problem that had not been anticipated during the design phase and was subsequently revealed by operating experience. The bolts are being replaced by others of different material.

(b) 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 case of loss of off-site power the emergency batteries would be able to start up the emergency diesel generators. During the design the voltage drop by elements such as cables, fuses, breakers, etc. had not been taken into account. In addition, the batteries of the system had been undersized.

(c) 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 in order 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.

## 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 installation of the plant and during backfitting of equipment. If construction deficiencies cannot be detected by testing or maintenance they may cause latent failures that degrade plant safety.

Commissioning of new equipment, systems, or a plant is essential to the subsequent safe operation of the plant. The results of commissioning activities demonstrate that the requirements and intent of the design as stated in the safety analysis report have been met. The results also define the initial characteristics of systems. 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 which have lead 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 new 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 utility to a survey of underground piping on all its plants. This investigation revealed that all power plants 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 respective 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 geometry 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) While performing the annual tests of the diesel backed electrical systems, it was discovered that neither of the two boron injection pumps would have started automatically after a loss of offsite power. However, it was possible to start the pumps manually. The failure was caused by missing wires in the automatic control of both pumps and by a defective control card in one of them. There had been no such systematic check before, not even in the startup testing phase of the plant. The licensing authority subsequently required the automatic restarting to be checked with respect to all components important to safety. The event highlights difficulties to detect latent failures of stand-by safety systems if initial tests are not properly designed to identify construction deficiencies.

### 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 plant staff, adequate procedures and tools, and good management. Deficiencies in human performance (including licensed operators, 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 to 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 a too large drop of 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, operators 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) During outage, emergency core cooling system (ECCS) operability tests were in progress. During ECCS activation, 6 out of 8 isolation valves in the cooling water supply lines to the ECCS pumps failed to open. The failures were caused by incorrect adjustment of the clutches of the electric motors which activate the valves due to deficiencies in the respective maintenance procedures. The corrective actions include the revisions of the adjustment procedure of the limiting clutches of the valve motors, the amendment of training on the procedure, and modifications in scope and frequency of ECCS testing. The event shows the vulnerability of safety systems to inadequate maintenance procedures.

## 3.4. DEFICIENCIES IN SAFETY MANAGEMENT/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 safety 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 should highlight the deficiencies revealed which have 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 measures of the vendor and the utility’s reception inspections 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 related systems.

(b) During cooldown for refueling, 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 may not be independent if they are based on the same measuring standard. Subsequently, the licensing authority required additional precautionary measures and use of additional checks for the calculation of the correction factor.

(c) During functional testing in the commissioning phase, difficulties in operating the minimum flow isolation valves of the low pressure safety injection pumps were encountered. The problem was insufficient leaktightness obtained under motor control.

Further investigations of the same standardized design at this plant and others revealed the generic nature of this problem. In addition, analysis showed that the means used for testing valves during commissioning and in operation do not always enable detection of valve deterioration before failure to operate occurs.

## 3.5. DEFICIENCIES IN SAFETY EVALUATION

The safety evaluation should cover 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. not analysed event sequences or conditions). Due to these deficiencies, unexpected situations may occur which significantly compromise plant safety. Related event reports should provide information about the deficiencies identified and the responses of plant operators to control the event, and lessons learned to prevent recurrence. Typical examples in this category are environmental conditions not adequately 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, no safety hazard was encountered. 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, operators 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) Prior to the planned replacement of the steam generators of a pressurized water reactor, the current analysis of a main steam line break (MSLB) inside containment was reviewed. The existing analysis assumed that the limiting MSLB inside containment was at hot standby conditions. This assumption did not consider that the limiting condition for peak containment temperature would occur at full power and that the limiting condition for maximum mass discharge to the containment would be at full power if the main feedwater regulating valve is assumed to fail open. The lessons learned are that pressurized water reactors, whose safety analysis assumed that the most severe MSLB in containment was at hot standby, could be operated in an unanalysed condition such that an MSLB could result in the plant's reactor containment building being subjected to pressures and temperatures significantly above their maximum design values. Additionally, the pressures and temperatures could be above ratings for the safety related electrical equipment in the containment required for safe shutdown of the plant.

## 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 occupational workers, the public and the environment. All such events/issues should be reported under this category.

Examples:

(a) During a personnel training session at a decommissioning plant, the 52 tonnes maximum weight cask was hooked to the 55 tonnes maximum capacity crane, at about 15-20 cm from the floor. At that moment, one end of the rope, fastened to the balancing beam, became unthreaded. A broken clip fell, followed by another one. Immediately after, the end of the second rope became unthreaded as well, and other clips fell. The cask fell from about 15-20 cm to the underlying floor and stayed vertically. No injuries occurred to the personnel. No significant damage appeared to the cask. A large plastic deformation was made to the metallic manhole cover, which had to be replaced. The first broken clip showed a surface defect, a crack, in the failed section. Old clips, made of ductile iron, experienced repeated rubbing against the sheaves and the frame, with consequent wear and tear due to a very narrow location of the balancing beam. They were changed to new ones, of a more fragile type, in the general maintenance of the crane in 1984. The common failure of all the other clips could be explained by:

- insufficient torque used for the clips tightening
- insufficient number of clips.

The root cause is to be found in the design and layout of the balancing beam and sheaves. The corrective actions are related to modifications in design and testing. As a consequence of the event the following corrective actions were made:

- replacement of the thimble U bolt clip arrangement with a different type of clamping to the balancing beam, consisting of a socketing integrated to the rope with a plastic matrix and held to the balancing beam by a passing-pin (Fig. 2). The new assembly was tested at 140% of the full capacity of the crane in the hypothesis of one rope unthreaded,
- modification of the balancing, beam, aimed to avoid rubbing of the ropes.

(b) Inspection activity, results and licensee practices/guidance for free release of material in a decommissioning plant:

The inspector reviewed the implementation of the (plant's) material release radiation survey program for the Unit 1 decommissioning. This included review of the regulatory requirements and standards for 'Free Release Surveys,' material release procedures that used for Unit 1 release surveys, free release survey records, survey instruments, and licensee self-assessments of free-release activities.

As a result of the previous inspection in March 2000, the licensee had conducted two evaluations of the material release program. The implementing procedure SO123-VII-20.9.2, Revision 2, 'Material Release Surveys,' was reviewed for its applicability to Unit 1 decommissioning activities by the inspector and the licensee. The principle objectives of the procedure were:

- To describe survey methods and criteria for the removal of materials from radiological controlled areas (RCA) and radioactive material areas (RMAs) to preclude the release of licensed radioactive material to unrestricted areas.

- To describe employees' responsibilities for performing material release surveys. The basis for the licensee's free release survey program for items potentially contaminated with licensed radioactive material was derived from NRC Inspection and Enforcement Circular 81-07, 'Control of Radioactivity Contaminated Material' dated May 14, 1981, as supplemented by NRC Information Notice 85-92, 'Surveys of Waste before Disposal from Nuclear Reactor Facilities' dated December 2, 1985.

In preparation for the demolition of the diesel generator building and free release of the concrete and steel, the licensee conducted a directed self-assessment of this decommissioning activity in March 2000. The inspector reviewed the self-assessment including an attachment to the assessment, 'Questions and Proposed Answers Regarding the Release of Large Amounts of Material Which are Believed to be Free of Licensed Radioactive Material,' dated November 8, 1999. The licensee had determined that the existing site procedure for material release surveys, SO123-VII-20.9.2 which was written for PLANT Units 1, 2, and 3 during operational status was applicable and appropriate for Unit 1 decommissioning.

Guidance used by the licensee for release of material from the Unit 1 included the following:

- Items and material will not be removed from the restricted area until they have been surveyed or evaluated for potential radioactive contamination by qualified persons.

- Care will be taken to ensure that no licensed radioactive material is released offsite by using survey methods for detecting very low levels of radioactivity.

- Final measurements of each package of aggregated waste will be conducted to ensure that an accumulation of licensed material resulting from a build-up of non-detectable radioactivity had not occurred.

- The free release criterion for direct and smear surveys is less than 100 counts per minute above background using a Geiger-Mueller (GM) count rate meter and pancake probe.

- The free release criterion for aggregate and indirect surveys using a micro Roentgen meter or scintillation detector was 'no detectable activity.'

## 4. CATEGORY 4: GENERIC PROBLEMS OF SAFETY INTEREST

Events that reveal deficiencies which 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 should also be reported under this category. These generic problems might not have been adequately identified or addressed by operating experience feedback, research and 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 either. 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 operators 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 required shutdown margins and drop times may be violated.

(c) Many cases of loss of proper function and damage to internal components of safety related power operated gate valves, caused both by thermally induced pressure locking and thermal binding, have been reported. Based on the history of industry problems, it was concluded that these potentially significant common mode valve failure mechanisms can occur to gate valves used in the safety related systems of all nuclear power plants. Despite industry awareness of the problem and guidance having been provided for identifying susceptible valves and for performing appropriate preventive and corrective measures, pressure locking and thermal binding events continue to occur. Valve susceptibility to pressure locking may not be detected or identified during normal valve surveillance testing. A comprehensive evaluation to identify all gate valves that are susceptible to pressure locking or thermal binding is needed, including a detached analysis to ensure that all plant operating and accident modes are considered. Moreover, it appears that a design modification is a simple, effective, and necessary step to ensure proper operation of valves susceptible to pressure locking.

## 5. CATEGORY 5: CONSEQUENTIAL ACTIONS TAKEN BY THE REGULATORY BODY

This category is intended to include significant 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 nuclear power plants. They include important modifications to the design basis, changes to design assessment requirements, changes to fault trees considered in probabilistic safety analysis, important changes to the requirements for construction, commissioning, surveillance and decommissioning of plant systems, changes to requirements on 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 excluded. The following regulatory assessment confirmed the proposed design change.

(b) A regulatory body identified deficiencies in the reactor operator training programme of a licensee and became aware that applications for reactor operator licenses contained apparently false information. Had these problems gone undetected, the probable consequences would have been that the plant would have undertaken initial startup and operation with operators not fully trained to accomplish their jobs. The cause of these occurrences was failure to exercise management control. Low priority was given to the reactor operator training programme and management relied heavily on unmonitored contractors to train and certify completion of training. The licensee was committed to conduct a review of the previous training of all licensed operators, shift technical advisors, and on-shift operations advisors. Certain operators were removed from licensed duties until they could be retrained and retested. An independent recertification examination of each licensed operator was performed by the regulatory body.

(c) 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 operators 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.

## 6. CATEGORY 6: EVENTS OF POTENTIAL SAFETY SIGNIFICANCE

This category is intended to include events which did not have any significant actual safety consequences, but which nonetheless are of potential safety significance. It covers events where protective systems were actuated to mitigate consequences of an event or where these systems had been challenged unnecessarily. It includes especially near-miss situations which 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) During shutdown mode, a turbine driven pump of the auxiliary feedwater system rotated without lubrication due to the start of a parallel motor driven pump. This occurred because the check valve downstream of the pump was in an open position. The resulting reversed flow through the pump caused bearing damage. The check valve deficiency resulted from the installation of new locking plates on the rotation axis of the valve. This fault was found to be generic, because other valves were equipped with the same new locking plates. The main lesson to be learned from this event is the importance of reviewing even minor modifications that might have significant consequences on plant safety. This is particularly the case if such consequences may not be detected during periodic or requalification testing following routine or unscheduled maintenance operations.

(b) Prior to startup after the refueling 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.

(c) During a refueling outage in a pressurized water reactor, the pneumatic seal which seals the annulus between the reactor vessel and the bottom of the refueling cavity failed. In spite of attempts to minimize the resulting drainage of the refueling 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/decontamination, equipment damage assessment, seal design modification, integrated event safety analysis and procedure review.

## 7. CATEGORY 7: EFFECTS OF UNUSUAL EVENTS OF EITHER HUMAN-INDUCED OR NATURAL ORIGIN

This category is intended to include those events caused by an external act or condition which 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 sea water, lightning strikes, heavy rain or snow fall), external man-made events (e.g. explosion, fire, industrial transportation accident affecting the plant, aircraft crash), internal events (e.g. explosion, fire, flooding, toxic gas release, turbine missiles), as well as external and internal acts of sabotage or terrorism.

Examples:

(a) For a few days in the year, the river providing cooling water to the plant is strongly polluted with leaves and wood. Due to personnel error, the cleaning of the intake sieve drums was not carried out, resulting in their clogging. As a result, a bypass flap opened and dirty water entered the plant cooling water circuit. The control room operators reacted immediately after detecting the rise of the intermediate cooling water temperature. Lessons learned showed that the water intake is highly sensitive to operational failures when the river water is full of leaves and wood.

(b) 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.

(c) In a pressurized water reactor, unjustified engineered safety feature (ESF) 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 transmitter/receivers. The subsequent laboratory tests confirmed that these radio frequency devices were able to emit high energy signals even when they were switched off. The lessons learned showed that shielding measures should be considered to protect sensitive I&C equipment against electromagnetic influences.

## 8. CATEGORY 8: OTHER FINDINGS AND OPERATING EXPERIENCE INFORMATION

This category is intended to include new perspectives, industry initiatives, and lessons learned 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 certain standards or new regulatory requirements that may impact nuclear plant systems. Research results in the area of digital instrumentation and controls, man-machine interface, etc. that may be useful in the review of nuclear plant design changes/modifications may also be reported under this category.

Examples:

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

(b) The implications of ultra-low sulphur diesel fuel or 'Bio-diesel' on the performance of nuclear station emergency diesel generators.

(c) New safety requirements such as severe accident management guidelines (SAMG's) may have implications for operating plant 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 useful in the prevention of future events or the lessening of the severity of events. (Opportunities for improvement identified for the Forsmark grid interaction event or the loss of the Northeast grid in the USA are possible examples).

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

# APPENDIX C

## IRS Coding

**Contents**

# DICTIONARY OF CODES

## REACTOR TYPES

| | |
|---|---|
| BWR | Boiling water reactor |
| FBR | Fast breeder reactor |
| GCR | Gas cooled reactor (graphite or heavy water moderated; includes AGR, HTGR and HWGCR) |
| HWLWR | Heavy water moderated, boiling light water cooled reactor |
| LWGR | Light water cooled, graphite moderated reactor (e.g. RBMK) |
| PHWR | Heavy water moderated, pressure tube reactor |
| PWR | Pressurized water reactor (includes WWER) |
| SGHWR | Steam generating heavy water reactor |
| OTHER | Other type of reactor |

## COUNTRY CODES

| | |
|---|---|
| AM | Armenia |
| AR | Argentina |
| BE | Belgium |
| BG | Bulgaria |
| BR | Brazil |
| CA | Canada |
| CH | Switzerland |
| CN | China |
| CZ | Czech Republic |
| DE | Germany |
| ES | Spain |
| FI | Finland |
| FR | France |
| GB | United Kingdom |
| HU | Hungary |
| IN | India |
| IT | Italy |
| JP | Japan |
| KR | Republic of Korea |
| LT | Lithuania |
| MX | Mexico |
| NL | Netherlands |
| PK | Pakistan |
| RO | Romania |
| RU | Russian Federation |
| SE | Sweden |
| SL | Slovenia |
| SK | Slovakia |
| UA | Ukraine |
| US | United States of America |
| ZA | South Africa |

## 1. REPORTING CATEGORIES

1.1 Unanticipated releases of radioactive material or exposure to radiation

1.1.1 Unanticipated releases of radioactive material – without exposures beyond limits

1.1.2 Exposure to radiation that exceeds prescribed dose limits for members of the public

1.1.3 Unanticipated exposure to radiation for site personnel

1.2 Degradation of barriers and safety related systems

1.2.1 Fuel cladding failure

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

1.2.2.1 Degradation of primary coolant pressure boundary

1.2.2.2 Degradation of main steam or feedwater lines

1.2.2.3 Degradation of other high energy systems

1.2.3 Degradation of containment function or integrity

1.2.4 Degradation of systems required to control reactivity

1.2.5 Degradation of systems required to ensure primary coolant inventory and core cooling

1.2.6 Degradation of essential support systems

1.3 Deficiencies in design, construction (including manufacturing), installation and commissioning, operation (including maintenance and surveillance), safety management/quality assurance system, safety evaluation and decommissioning

1.3.1 Deficiencies in design

1.3.2 Deficiencies in construction (including manufacturing), installation and commissioning

1.3.3 Deficiencies in operation (including maintenance and surveillance)

1.3.4 Deficiencies in safety management/quality assurance system

1.3.5 Deficiencies in safety evaluation

1.3.6 Deficiencies in decommissioning

1.4 Generic problems of safety interest

1.5 Consequential actions taken by the regulatory body

1.6 Events of potential safety significance

1.7 Effects of unusual events of either man-made or natural origin

1.8 Other findings and operating experience information

## 2. PLANT STATUS PRIOR TO THE EVENT

2.0 Not applicable

2.1 On power

2.1.1 Full allowable power

2.1.2 Reduced power (including zero power)

2.1.3 Raising power or starting up

2.1.4 Reducing power

2.1.5 Refueling on power

2.2 Hot shutdown conditions

2.2.1 Hot standby (coolant at normal operating temperature)

2.2.2 Hot shutdown (coolant at or below normal operating temperature)

2.2.3 Natural circulation cooling

2.3 Cold shutdown (reactor sub-critical and coolant temperature < 93°C)

2.3.1 Cold shutdown with closed reactor vessel

2.3.2 Refueling or open vessel (for maintenance)

2.3.2.1 Refueling or open vessel – all or some fuel inside the core

2.3.2.2 Refueling or open vessel – all fuel out of the core

2.3.3 Mid-loop operation and other reduced primary coolant inventory conditions

2.3.4 Natural circulation cooling

2.4 Pre-operational

2.4.1 Construction, installation

2.4.2 Commissioning

2.5 Testing or maintenance being performed

2.6 Post-operational (decommissioning/dismantling/decontamination)

## 3. FAILED/AFFECTED SYSTEMS

### 3.A Primary systems

3.AA Reactor core (fuel assemblies, control and poison rods, guide thimbles, ...)

3.AB Systems for reactor control and protection e.g. control rod drive mechanism, accumulator…(motor, power supply, hydraulic system, other shutdown systems)

3.AC Reactor vessel (with core internals, PHWR or LWGR pressure tubes, ...)

3.AD Moderator and auxiliaries including neutron poison removal system (PHWR)

3.AE Primary coolant system (pumps and associated materials, loop piping, ...)

3.AF Pressure control (includes primary safety and relief valves)

3.AG Recirculating water system (BWR)

3.AH Steam generator, boiler, steam drum

3.AK At power fuel handling systems (PHWR, LWGR, GCR)

3.AL Annulus gas

**3.B Essential reactor auxiliary systems**

3.BA Reactor core isolation cooling (BWR)

3.BB Auxiliary and emergency feedwater

3.BC Emergency poisoning function (PWR mainly with the boron injection tank, chemical and volume control system participation)

3.BD Standby liquid control (BWR)

3.BE Residual heat removal (PWR and BWR except emergency core cooling functions)

3.BF Chemical and volume control (PWR with main pumps seal water, ...)

3.BG Emergency core cooling

3.BH Main steam pressure relief (reactors which have secondary loops)

3.BK Nuclear boiler overpressure protection (BWR)

3.BL Core flooding accumulator (PWR)

3.BP Failed fuel detection

3.BQ Gas cleanup system (LWGR, PHWR)

3.BR End shields and associated cooling system (PHWR)

**3.C Essential service systems**

3.CA Component cooling water (including reactor building closed cooling water)

3.CB Essential raw cooling or service water

3.CC Essential compressed air (e.g. instrument air…)

| | |
|---|---|
| 3.CD | Borated or refueling water storage (PWR) |
| 3.CE | Condensate storage |
| 3.CF | CO2 injection and storage (GCR) |

| | |
|---|---|
| **3.D** | **Essential auxiliary systems** |
| 3.DA | Spent fuel pool or refueling pool cooling and cleanup |
| 3.DB | Containment isolation (including penetrations and air lock door seals) |
| 3.DC | Main steam or feedwater isolation function |
| 3.DC.1 | Main steam isolation function |
| 3.DC.2 | Feedwater isolation function |
| 3.DD | Containment atmosphere clean up/treatment systems (e.g. spray, iodine removal…) |
| 3.DE | Containment pressure suppression |
| 3.DF | Containment combustible gas control |
| 3.DG | Essential auxiliary steam (GCR) |

| | |
|---|---|
| **3.E** | **Electrical systems** |
| 3.EA | High voltage AC (greater than 15kV) |
| 3.EA.1 | High voltage AC -- Onsite |
| 3.EA.2 | High voltage AC – Offsite (including Grid & Transmission Lines) |
| 3.EB | Medium voltage AC (600V to 15kV) |
| 3.EC | Low voltage AC (less than 600V – mainly 480V) |
| 3.ED | Vital instrumentation AC and control AC |
| 3.EE | DC power (e.g. UPS, batteries, rectifiers…) |
| 3.EF | Emergency power generation and associated auxiliaries (including fuel oil) |
| 3.EG | Security and access control |
| 3.EH | Communication and alarm annunciation |

| | |
|---|---|
| **3.F** | **Feedwater, steam and power conversion systems** |
| 3.FA | Main steam and auxiliaries (including auxiliary steam) |

| | |
|---|---|
| 3.FB | Turbines (main, feedwater and auxiliary feedwater turbines and associated auxiliaries) |
| 3.FC | Main condenser and auxiliaries (non-condensable gases extraction and treatment) |
| 3.FE | Turbine steam by-pass to condenser |
| 3.FG | Feedwater and condensate (including pumps, heat exchangers, tanks, etc.) |
| 3.FM | Condensate demineralizer |
| 3.FN | Circulating or condenser cooling water (including raw cooling and service water) |

**3.H Heating, ventilation and air conditioning systems (HVAC)**

| | |
|---|---|
| 3.HA | Primary reactor containment building HVAC |
| 3.HB | Primary containment vacuum and pressure relief |
| 3.HC | Secondary containment recirculation, exhaust and gas treatment (includes BWR standby gas treatment) |
| 3.HD | Drywell or wetwell HVAC and purge and inerting (BWR) |
| 3.HE | Reactor or nuclear auxiliary building HVAC |
| 3.HF | Control building HVAC (including main control room HVAC) |
| 3.HG | Fuel and spent fuel buildings HVAC |
| 3.HH | Turbine building HVAC |
| 3.HK | Waste management building HVAC |
| 3.HM | Miscellaneous structures HVAC (e.g. laboratories…) |
| 3.HN | Chilled water |
| 3.HP | Plant stack |
| 3.HQ | Emergency generator building HVAC |
| 3.HR | Seismic/Bunkered emergency control building HVAC |

**3.I Instrumentation and control systems**

**3.I.1 Analog I&C systems**

**3.I.2 Digital I&C systems**

| | |
|---|---|
| 3.IA | Plant/process computer (including main and auxiliary computers) |
| 3.IB | Fire detection |

| | |
|---|---|
| 3.IC | Environment monitoring |
| 3.ID | Turbine generator instrumentation and control |
| 3.IE | Plant & process monitoring (including the main and remote/supplementary control room equipment and various remote control functions) |
| 3.IF | In-core and ex-core neutron monitoring (including BWR reactor stability monitoring) |
| 3.IG | Leak monitoring (reactor coolant boundary, containment and auxiliary buildings) |
| 3.IH | Radiation monitoring |
| 3.IH.1 | Plant radiation monitoring |
| 3.IH.2 | Personnel monitoring (dosimetry & contamination detection) |
| 3.IK | Reactor power control (e.g. control rods & boration/dilution systems) |
| 3.IL | Recirculation flow control (BWR) |
| 3.IM | Feedwater control |
| 3.IN | Reactor protection |
| 3.IP | Engineered safety features actuation (including emergency systems actuation) |
| 3.IQ | Non-nuclear instrumentation |
| 3.IR | Meteorological instrumentation |
| 3.IS | Seismic instrumentation |
| 3.IT | Vibration monitoring |

**3.K Service auxiliary systems**

| | |
|---|---|
| 3.KB | Sampling (normal & accident conditions) |
| 3.KC | Control and service air (non-essential) and compressed gas |
| 3.KD | Demineralized water |
| 3.KE | Material and equipment handling |
| 3.KG | Nuclear fuel handling and storage (both fresh and spent fuel) |
| 3.KH | Fire protection |
| 3.KP | Chemical additive injection |

**3.S Structural systems**

3.SA Primary reactor containment building

3.SB Secondary reactor containment building or vacuum building (PHWR)

3.SC Reactor or nuclear auxiliary building

3.SD Control building

3.SE Emergency generator building

3.SF Fresh and spent fuel buildings

3.SG Turbine building

3.SH Waste management building

3.SK Pumping stations (e.g. cooling, make-up water, fire water…)

3.SL Backup ultimate heat sink building

3.SM Cooling towers &/or intake structure

3.SN Switchyard (enclosed/open)

3.SP Seismic/bunkered emergency control building

3.SQ Heavy water up-gradation building (PHWR)

**3.W Waste management systems**

3.WA Liquid radwaste processing, hold-up & discharge

3.WB Solid radwaste

3.WC Gaseous radwaste hold-up & discharge

3.WD Non-radioactive waste (liquid, solid and gaseous)

3.WE Steam generator blowdown

3.WF Plant drainage (floor, roof, ...)

3.WG Equipment drainage (including vents)

3.WH Suppression pool cleanup

3.WK Reactor water cleanup (BWR, PHWR, LWGR,...)

**3.Z No system involved**

3.ZA Other systems (specified in text of IRS report)

## 4. FAILED/AFFECTED COMPONENTS

**4.0 No specific component involved**

**4.1 Instrumentation (gauges, transmitters, sensors controllers, detectors, displays…)**

4.1.0 Other (specified in text of IRS report)

4.1.1 Pressure

4.1.2 Temperature

4.1.3 Level

4.1.4 Flow

4.1.5 Radiation/Contamination

4.1.6 Chemical concentration

4.1.7 Position

4.1.8 Dewpoint, moisture

4.1.9 Neutron flux (detectors, ion chambers and associated components)

4.1.10 Speed (e.g. rotational speed of equipment, wind speed…)

4.1.11 Fire (smoke, flames, heat,…)

4.1.12 Hydrogen concentration

4.1.13 Electrical (current, voltage, power, ...)

4.1.14 Vibration

4.1.15 Seismic motion

**4.2 Mechanical**

4.2.0 Other (specified in text of IRS report)

4.2.1 Pumps, compressors, fans

4.2.2 Turbines (steam, gas, hydro), engines (diesel, gasoline, ...)

4.2.2.1 Turbines (steam, gas, hydro)

4.2.2.2 Engines (diesel, gasoline, ...)

4.2.3 Valves (including safety/relief/check/solenoid valves) , valve operators, controllers, dampers and fire breakers, seals and packing

4.2.4 Steam generators and heat exchangers including internals

4.2.4.1 Steam generators including internals

4.2.4.2 Heat exchangers including internals

4.2.4.3 BWR vessel internals

4.2.5 Tanks, pressure vessels (e.g. reactor vessel and internals, accumulators)

4.2.6 Tubes, pipes, ducts

4.2.7 Fittings, couplings (including transmissions and gear boxes), hangers, supports, bearings, thermal sleeves, snubbers

4.2.8 Strainers, screens, filters, ion exchange columns

4.2.9 Penetration (personnel access, equipment access, fuel handling, ...)

4.2.10 Control or protective rods and associated components or mechanisms, fuel elements

4.2.11 Fuel storage racks, fuel storage casks and fuel transport containers

4.2.12 Nuclear assemblies (absorber, burnable, breeder, reflectors, neutron sources, shielding equipment)

**4.3 Electrical**

4.3.0 Other (specified in text of IRS report)

4.3.1 Switchyard equipment (switchgear, transformers, buses, line isolators, ...)

4.3.2 Circuit breakers, power breakers, fuses

4.3.3 Alarms

4.3.4 Motors (for pumps, fans, compressors, valves, motor generators, ...)

4.3.5 Generators of emergency and stand-by power

4.3.6 Main generator and auxiliaries

4.3.7 Relays, connectors, hand switches, push buttons, contacts

4.3.8 Wiring, controllers, starters, electrical cables

**4.4 Computers**

4.4.1 Computer hardware

4.4.2 Computer software

## 5. CAUSE OF THE EVENT

### 5.1 Direct cause

| | |
|---|---|
| 5.1.0 | Unknown |
| 5.1.0.1 | Other (specified in text of IRS report) |
| 5.1.1 | Mechanical failure |
| 5.1.1.0 | Other mechanical failure |
| 5.1.1.1 | Corrosion, erosion, fouling |
| 5.1.1.2 | Wear, fretting, lubrication problem |
| 5.1.1.3 | Fatigue |
| 5.1.1.4 | Overloading (including mechanical stress and overspeed) |
| 5.1.1.5 | Vibration |
| 5.1.1.6 | Leak |
| 5.1.1.7 | Break, rupture, crack, weld failure |
| 5.1.1.8 | Blockage, restriction, obstruction, binding, foreign material |
| 5.1.1.9 | Deformation, distortion, displacement, spurious movement, loosening, loose parts |
| 5.1.2 | Electrical failure |
| 5.1.2.0 | Other electrical failure |
| 5.1.2.1 | Short-circuit, arcing |
| 5.1.2.2 | Overheating |
| 5.1.2.3 | Overvoltage |
| 5.1.2.4 | Bad contact, disconnection |
| 5.1.2.5 | Circuit failure, open circuit |
| 5.1.2.6 | Ground fault |
| 5.1.2.7 | Undervoltage, voltage breakdown |
| 5.1.2.8 | Faulty insulation |
| 5.1.2.9 | Failure to change state |
| 5.1.3 | Chemical or core physics failure |
| 5.1.3.0 | Other chemical failure/problem (specified in text of IRS report) |
| 5.1.3.1 | Chemical contamination (including corrosion products, anionic impurities), deposition |

5.1.3.2 Uncontrolled chemical reaction

5.1.3.3 Core physics problems (operation outside core physics limits, e.g. shutdown margins, reduction in reactivity worth of reactivity devices…)

5.1.3.4 Poor chemistry or inadequate chemical control

5.1.4 Hydraulic/pneumatic failure

5.1.4.0 Other hydraulic/pneumatic failure (specified in text of IRS report)

5.1.4.1 Water hammer, pressure fluctuations, over pressure

5.1.4.1.1 Water hammer

5.1.4.1.2 Pressure fluctuations, over pressure

5.1.4.2 Loss of fluid flow

5.1.4.3 Loss of pressure

5.1.4.4 Cavitation

5.1.4.5 Gas binding and pressure locking

5.1.4.6 Moisture in air systems

5.1.4.7 Vibration due to fluid flow

5.1.5 Instrumentation and control failure

5.1.5.0 Other instrumentation and control failure (specified in text of IRS report)

5.1.5.2 False response, loss of signal, spurious signal

5.1.5.3 Oscillation

5.1.5.4 Set point drift, parameter drift

5.1.5.5 Computer hardware deficiency

5.1.5.6 Computer software deficiency

5.1.5.7 Electromagnetic and/or radiofrequency interference

5.1.6 Environmental (abnormal conditions inside plant)

5.1.6.0 Other internal environmental cause (specified in text of IRS report)

5.1.6.1 High temperature

5.1.6.2 Pressure

5.1.6.3 Humidity

5.1.6.4 Flooding, water ingress

5.1.6.5 Low temperature, freezing

5.1.6.6 Radiation, contamination, irradiation of parts

5.1.6.7 Dropped loads, missiles, high energy impacts

5.1.6.8 Fire, burning, smoke, explosion

5.1.7 Environmental (external to the plant)

5.1.7.0 Other external environmental cause (fire, toxic/explosive gasses,...)

5.1.7.1 Lightning strikes

5.1.7.2 Flooding

5.1.7.3 Storm, wind loading

5.1.7.4 Earthquake

5.1.7.5 Freezing

5.1.7.6 High ambient temperature

5.1.7.7 Heavy rain or snow

5.1.10 Human factors

5.1.10.1 Slip or lapse

5.1.10.2 Mistake

5.1.10.3 Violation

5.1.10.4 Sabotage

**5.3 Plant staff involved**

5.3.1 Maintenance

5.3.2 Operations

5.3.3 Technical and engineering

5.3.4 Management and administration

5.3.5 Control of contractor/sub-contractor

**5.4 Type of activity**

5.4.1 Not relevant

5.4.2 Normal operations

5.4.3 Shutdown operations

5.4.4 Equipment startup

5.4.5 Planned/preventive maintenance

5.4.6 Isolating/de-isolating (e.g. clearance & tagging of electrical & piping systems)

5.4.7 Repair (unplanned/breakdown maintenance)

5.4.8 Routine testing with existing procedures/documents

5.4.9 Special testing with one-off special procedure

5.4.10 Post-modification testing

5.4.11 Post-maintenance testing

5.4.12 Fault finding

5.4.13 Construction, installation and commissioning (of new equipment system, or complete plant)

5.4.14 Return to service (of existing equipment)

5.4.15 Decommissioning

5.4.16 Fuel handling/refueling operations

5.4.17 Inspection

5.4.18 Abnormal operation (due to external or internal constraints)

5.4.19 Engineering (design or field engineering) review

5.4.20 Modification implementation

5.4.21 Training

5.4.22 Actions taken under emergency conditions

5.4.23 Other activity (specified in text of IRS report)

5.4.24 Inspections, Tests, Analysis, Acceptance Criteria (ITAAC) – for new reactor construction

**5.5 Human performance related causal factors and root causes**

5.5.1 Verbal communications

5.5.2 Personnel work practices

5.5.2.0 Others (specified in text of IRS report)

5.5.2.1 Control of task/independent verification

5.5.2.2 Complacency/lack of motivation/inappropriate habits

5.5.2.3 Use of improper tools and equipment

5.5.2.4 Self-check practices (e.g. Stop, Think, Act, Review (STAR)…)

5.5.2.5 Questioning attitude, dealing with uncertainty (e.g. assumption of competence of more experienced personnel)

5.5.3 Personnel work scheduling (including workload, work time provided)

5.5.4 Environmental conditions

5.5.5 Man-machine interface

5.5.5.1 Alarm control & maintenance practices

5.5.5.2 Equipment/controls labeling

5.5.6 Training/qualification

5.5.7 Written procedures and documents

5.5.7.1 Procedure availability

5.5.7.2 Procedure completeness/accuracy

5.5.7.3 Procedure compliance

5.5.8 Supervisory methods (e.g. standard setting, emphasis of safe work practices & questioning attitude, self checks...)

5.5.9 Work organization

5.5.9.0 Others (specified in text of IRS report)

5.5.9.1 Shift/team size or composition

5.5.9.2 Planning/preparation of work (e.g. work package planning, pre-job briefings, shift turnover practices)

5.5.10 Personal factors

5.5.10.0 Others

5.5.10.1 Fitness for work (e.g. fatigue…)

5.5.10.2 Stress/perceived lack of time/boredom (including imposition of parallel &/or unexpected tasks)

5.5.10.3 Skill of the craft less than adequate/not familiar with job performance standards (including task difficulty)

5.5.11 Use of operating experience

## 5.6 Management related causal factors and root causes

5.6.0 Other (specified in text of IRS report)

5.6.1 Management direction

5.6.1.1 Existence of policies, standards, expectations

5.6.1.2 Communication/Enforcement of policies, standards, expectations

5.6.1.3 Production pressure/perceived pressure

5.6.1.4 Clarity of responsibility

5.6.2 Communication or co-ordination

5.6.3 Management involvement, monitoring and assessment

5.6.4 Decision process

5.6.5 Allocation of resources (e.g. planning & prioritization relative to safety…)

5.6.6 Change management

5.6.7 Safety culture

5.6.8 Management of contingencies (e.g. alternate plans of action…)

5.6.9 Management of contracted work (e.g. qualification, training, supervision and guidance…)

5.6.10 Management of staff training and qualification

5.6.11 Knowledge management

**5.7 Equipment related causal factors and root causes**

5.7.0 Others (specified in text of IRS report)

5.7.1 Design configuration and analysis

5.7.1.1 Design analysis quality

5.7.1.2 Materials selection

5.7.1.3 Modifications engineering quality

5.7.1.4 Modifications engineering review process

5.7.2 Equipment (procurement) specification, manufacture, storage and installation

5.7.2.1 Receipt inspection

5.7.2.2 Parts/consumables shelf life/storage controls

5.7.2.3 Installation and commissioning

5.7.3 Maintenance, testing or surveillance

5.7.3.1 Foreign material exclusion controls

5.7.3.2 Parts & consumables selection/use

5.7.4 Equipment environmental qualification

5.7.5 Equipment aging

## 6. EFFECTS ON OPERATION

| | |
|---|---|
| 6.0 | No significant effect on operation or not relevant |
| 6.1 | Reactor scram |
| 6.1.1 | Automatic reactor scram |
| 6.1.2 | Manual reactor scram |
| 6.2 | Controlled shutdown |
| 6.3 | Load reduction |
| 6.3.1 | Automatic load reduction |
| 6.3.2 | Manual load reduction |
| 6.4 | Activation of engineered safety features |
| 6.5 | Challenge to safety or relief valve |
| 6.5.1 | Challenge to safety or relief valve in the primary circuit |
| 6.5.2 | Challenge to safety or relief valve in the steam or condensate cycle |
| 6.6 | Unanticipated or significant release of radioactive materials |
| 6.6.1 | Unanticipated or significant release of radioactive materials outside the plant |
| 6.6.2 | Unanticipated or significant release of radioactive materials inside the plant |
| 6.7 | Unplanned or significant radiation exposure of personnel or public |
| 6.8 | Personnel or public injuries |
| 6.9 | Outage extension |
| 6.10 | Exceeding technical specification limits |
| 6.11 | House load operation (plant continues to operate supplying only its own loads) |

## 7. CHARACTERISTICS OF THE EVENT/ISSUE

| | |
|---|---|
| 7.0 | Other characteristics |
| 7.1 | Degraded fuel |
| 7.2 | Degraded reactor coolant boundary |
| 7.3 | Degraded reactor containment |
| 7.4 | Loss of safety function |
| 7.5 | Significant degradation of safety function |
| 7.6 | Failure or significant degradation of the reactivity control |
| 7.7 | Failure or significant degradation of plant control |
| 7.8 | Failure or significant degradation of heat removal capability |
| 7.9 | Loss of off-site power |

| | |
|---|---|
| 7.10 | Loss of on-site power |
| 7.11 | Transient |
| 7.11.0 | Other transient |
| 7.11.1 | Power transient |
| 7.11.2 | Temperature transient |
| 7.11.3 | Pressure transient |
| 7.11.4 | Flow transient |
| 7.12 | Physical hazards (internal or external to the plant) |
| 7.13 | Discovery of major condition not previously considered or analysed |
| 7.14 | Fuel handling event |
| 7.15 | Radwaste event |
| 7.16 | Security, safeguards, sabotage or tampering event |

## 8. NATURE OF FAILURE OR ERROR

| | |
|---|---|
| 8.0 | Not relevant |
| 8.1 | Single failure or single error |
| 8.2 | Multiple failure or multiple error |
| 8.2.1 | Independent multiple failures or errors |
| 8.2.2 | Dependent multiple failures or errors |
| 8.2.3 | Recurrent failure or error |
| 8.3 | Common cause failure (including potential for CCF) |
| 8.4 | Significant or unforeseen interaction between systems |

## 9. RECOVERY ACTIONS

| | |
|---|---|
| 9.0 | Not relevant |
| 9.1 | Recovery by human action |
| 9.1.1 | Recovery by foreseen human action (e.g. procedures and instructions/guidelines available and used, training prepared the operators to respond…) |
| 9.1.2 | Recovery by unforeseen human action (e.g. new actions or actions outside the procedures required, inadequate or non existent training) |
| 9.2 | Recovery by automatic plant action or by design |
| 9.3 | No recovery |

# APPENDIX D

## Glossary
## for the Joint IAEA/NEA
## IRS Guidelines

# 1. INTRODUCTION

The terminology below is intended for use in the International Reporting System for Operating Experience (IRS) activity and may not necessarily conform to terminology/definitions adopted elsewhere for international and national use.

The aim of this Glossary is to provide an opportunity for minimizing the proliferation of different terms describing the same thing and for harmonizing terms used for the preparation of national reports on nuclear power plant events, submitted to the IRS.

# 2. GLOSSARY

**Anticipated Operational Occurrences**

All operational processes deviating from Normal Operation which are expected to occur once or several times during the operating life of the plant and which, in view of appropriate design provisions, do not cause any significant damage to items important to Safety nor lead to accident conditions. (Note: Some Member States use the term ‘Abnormal Operation’, which means the same thing).

Examples of Anticipated Operational Occurrences are loss of normal electric power and faults such as a turbine trip, malfunction of individual items of a normally running plant, failure to function of individual items of control equipment, loss of power to main coolant pump.

**Barrier**

Anything that is used to protect a system or person from a hazard and includes physical barriers, natural barriers, administrative controls and human actions.

Also, administrative or physical controls designed to promote consistent performance that should inhibit an inappropriate action.

Barriers can be either administrative or physical in nature.

**Barrier function**. Those functions that contain, prevent, prescribe, monitor, and so forth.

**Barrier system**. A system that establishes barrier function systems such as material (physical barriers), functions (locks, passwords), symbolic (signs, postings, and procedures), and immaterial (self checking).

**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.

Also, a condition that shapes the outcome of a situation.

Also, 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

**Common Cause Failure**

The failure of two or more structures, systems and components (SSCs) to perform their functions as a result of a shared cause or specific occurrence. (Note: common mode failure, in which two or more SSCs fail in the same manner or mode due to a shared cause or specific occurrence, is a type of Common Cause Failure).

Multiple unavailability of identical component is of particular importance in this case, since the probability of several components being simultaneously and independently unavailable is rather small.

**Defence In Depth**

The application of more than a single protective measure for a given safety objective such that the objective is achieved even if one of the protective measures fails.

This basic safety principle is used to compensate for potential human and mechanical failures. The defence in depth concept is implemented based on several levels of protection, including successive barriers, preventing the release of radioactive material to the environment. The concept includes protection of the barriers by averting damage to the plant and to the barriers themselves. It includes further measures to protect the public and the environment from harm in case these barriers are not fully effective.

**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:

(1) Shared equipment dependencies.
(2) Functional dependencies.
(3) Common cause initiators.
(4) Physical interaction failures.
(5) Human interaction.
(6) Common cause failures.

**Direct/observed Cause**

The failure, action, omission or condition which immediately produced the event (i.e. the direct initiator of an effect or event).

**Diversity**

The existence of redundant SSCs to perform a defined function, where such SSCs collectively incorporate one or more different attributes.

Examples of such attributes are: different operating conditions, different sizes of equipment, different manufacturers, different operational or design principles, and types of equipment that use different physical methods.

**Event**

Any unintended (unusual) occurrence or a sequence of related occurrences, including human errors, equipment failures or other mishaps, the consequences or potential consequences of which are not negligible from the point of view of nuclear safety. An Event may also be cited by other terms, such as a deviation, an incident or an accident.

An action or happening that occurred during some activity.

Also, an unwanted, undesirable consequence for the safe operation of a plant (generally in terms of reduced safety margin).

Also, an undesirable consequence that challenges the safety of the reactor core.

Also, an undesirable occurrence.

Also in the context of the reporting and analysis of events, an event is any unintended occurrence, including operating error, equipment failure or other mishap, the consequences or potential consequences of which are not negligible from the point of view of radiological protection or nuclear safety.

**Failure**

Inability of a structure, system or component to function within acceptance criteria.

Note that the structure, system or component is considered to fail when it becomes incapable of functioning, whether or not this is needed at that time. A failure in, for example, a backup system may not be manifest until the system is called upon to function, either during testing or on failure of the system it is backing up.

**Human Errors**

Groups/families of attributes to characterize wrong human behaviour (understanding, intention and action).

Examples of such groups are: violation (the person has a good understanding, he develops an intention not in compliance with his understanding); mistake (the intention of the person is wrong because his understanding is not in compliance with the prescribed task); slip (the intention was good, 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.

**Normal Operation**

Operation of a nuclear power plant within specified Operational Limits and Conditions including shutdown, power operation, shutting down, starting up, maintenance, testing and refueling.

**Nuclear Safety (or simply Safety)**

The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of site personnel, the public and environment from undue radiation hazards.

**Operation**

All activities performed to achieve the purpose for which the plant was constructed, including maintenance, testing, refueling, in-service inspection and other associated activities.

**Operational Limits and Conditions**

A set of rules which set forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of the nuclear power plant.

**Operating experience**

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 reporting thresholds for events occurring at nuclear installations during commissioning, operation, surveillance and maintenance activities and decommissioning, and on deviations from normal performance by systems and by personnel, which could be precursors of events.

**Precursor**

An event that has the potential, under other circumstances, to lead to a core damage event.

**Prescribed Limits**

Limits established or accepted by the regulatory body.

The term ‘authorized limits’ is sometimes used for this term in IAEA documents.

**Protection System**

A system that monitors the operation of a reactor plant and which, on sensing an abnormal condition, automatically initiates actions to prevent an unsafe or potentially unsafe condition.

Such a system would include all monitoring & control instrumentation, electrical and mechanical devices and circuitry, from sensors to actuation devices involved in the protective function.

**Quality Assurance**

All those planned and systematic actions necessary to provide adequate confidence that an item or service will satisfy given requirements for quality.

**Recovery Actions**

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

**Redundancy**

Provision of more than the minimum number of (identical or diverse) elements or systems, so that the loss of any one does not result in the loss of the required function of the whole.

**Regulatory Body**

A national authority or a system of authorities designated by a Member State, assisted by technical and other advisory bodies, and having the legal authority for conducting the licensing process, for issuing licences and thereby for regulating nuclear power plant Siting, Design, Construction, Commissioning, Operation and Decommissioning or specified aspects thereof. This national authority could be either the government itself, or one or more departments of the government, or a body or bodies especially vested with appropriate legal authority.

**Residual Heat**

The sum of the heat originating from radioactive decay and shutdown fission and the heat stored in reactor related structures and in heat transport media.

**Root Cause**

The fundamental cause(s) that if corrected will prevent recurrence of an event or adverse condition.

Also, the most basic reason(s) for an event that can be reasonably identified and that over which management has control to remedy.

The fundamental cause of an initiating event which, if corrected, will prevent its recurrence, namely, the failure to detect and correct the relevant latent weakness(es) and the reasons for that failure

**Safety Culture**

The assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance.

**Safety Functions**

Safe operation of nuclear power plants is maintained by the following fundamental safety functions: (a) controlling reactivity; (b) cooling of the radioactive material; (c) confinement of the radioactive material.

**Safety Management System**

A system to achieve and enhance safety by bringing together requirements for managing the organization, including planned and systematic actions providing confidence that the requirements are satisfied. The safety management system ensures that health, environmental, security, quality and economic requirements are integrated with safety requirements.

**Safety Systems**

Systems important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences or accident conditions.

Safety systems consist of the protection system, the safety actuation systems, and the safety system support features. Components of safety systems may be provided solely to perform safety functions or may perform safety functions in some plant operational states and non-safety functions in other plant operational states.

**Screening**

Reviewing operating experience information to determine what information is valuable for and applicable to a particular plant or more generically applicable to a number of plants of similar type or which use the same equipment.

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