GUIDE ON

INCIDENT REPORTING SYSTEM FOR RESEARCH REACTORS

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1. INTRODUCTION

The IAEA has included in its current programme on research reactor safety an Incident Reporting System for Research Reactors (IRSRR) with the objective to improve the safety of research reactors through the exchange of safety-related information on unusual events.

The systematic collection and evaluation of operational experience with unusual events[1] is a very useful way to improve operational safety. A proper analysis of unusual events can identify root causes and provide valuable lessons to be learned by, for example, reactor operators or reactor designers, etc. Reporting of all incidents occurring in the research reactor, as it is defined in the licence for operation, or as otherwise required by the regulatory body, or as described in the safety report, should be considered. Unusual events involving experimental devices and irradiation targets for isotope production should also be considered.

The Incident Reporting System for Research Reactors will collect, maintain and disseminate reports on unusual events which are received from Member States of the IAEA participating in the system (this could include reports on unusual events that occurred before the IRSRR came into effect). All reports will be stored in a database available for IAEA Member States.

The IRSRR will make use of the experience gained through the use of the IAEA/NEA Incident Reporting System for nuclear power plants (NPPs) [1] and of the information stored in the Agency's Research Reactor Database (Directory of IAEA databases, IAEA, December 1992).

The IRSRR should not be confused with the International Nuclear Event Scale (INES) reporting system.[2] [2]

This document provides guidance for the establishment of the IRSRR and the channels of communication within the system. It also describes the format and content of the information that participants should report.

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[1] The term `unusual event' in this document means safety-related unusual event including incidents and accidents.

[2] The IRSRR and the INES systems differ from each other fundamentally. The INES reporting system promptly provides the Media and the Public with General Information and an Authoritative Rating on events in Nuclear Facilities, on the basis of a (potential or real) Consequence Analysis. The IRSRR will provide in a timely manner the Technical Nuclear Community with Technical Information and Lessons Learned on events in research reactors, using the method of Cause Analysis.

2. PARTICIPATION OF MEMBER STATES IN THE IRSRR

Participation in the IRSRR will be voluntary and open to Member States which have embarked on a research reactor programme.

The IRSRR is intended as an international forum for the sharing of operational experience thus forming a feedback loop in an international context.

Member States will determine how information will be sent to and received from the Agency. The IAEA recommends that each Member State appoint a national co-ordinator (preferably from the regulatory body) and a local co-ordinator (from the operating organization). Each co-ordinator should be a professional, knowledgeable with research reactors or should be assisted by such a professional.

The IRSRR is based on the principle that each participant will provide timely information on its experience with unusual events in research reactors so that the information is available to all other participants. A Member State participating in the IRSRR shall therefore send event reports to the IAEA in accordance with the arrangements set out in this document.

3. RECEIPT AND DISTRIBUTION OF INFORMATION

Unusual events with safety significance or of general interest to the research reactor community should be identified by the national or local co-ordinators and selected for transmission to the IAEA. If a national or local co-ordinator considers an unusual event to be highly significant to safety, a preliminary report should be sent to the IAEA as soon as practicable. Reports can be prepared by national or local co-ordinators but to ensure uniform quality, the national co-ordinator should perform a quality check before sending the information to the IAEA. Reports should be provided as written documents and may also be provided electronically for computer retrieval.

The IAEA will nominate a person knowledgeable with research reactors and responsible for the operation of the IRSRR. Before storing information on incidents or sending them to participants, the IAEA will carry out a quality check of the information received. This quality check is intended to ensure that all requirements regarding format and content are met in accordance with the checklists given in Section 5 and the Appendices. Where major modifications are proposed by the IAEA, prior approval of the national co-ordinator shall be obtained before the IAEA transmits the reports to other participants in the IRSRR.

Upon receipt of a report, the IAEA will send it to all IRSRR co-ordinators of the participating Member States. If a participant asks the IAEA for additional information to supplement a report already received, or the Agency itself makes such a request, then the IAEA will pass this request on to the appropriate co-ordinator. All information sent to the IAEA under the IRSRR should be distributed through the co-ordinators only

The IAEA will process reports for appropriate computer storage and retrieval. Computer programs being developed for this purpose will permit searches for all the required information to be done easily.

The information stored in the IRSRR database will be made available to the participants in accordance with the confidentiality requirements of the national co-ordinator who supplied the reports. Without specific requirements, all event reports are restricted, i.e. for official use only within the research reactor community.

Besides receiving, storing and distributing information, the IAEA will prepare periodic reports on IRSRR activity and will organize periodic meetings to review and evaluate the material available on unusual events. The proceedings of such meetings will be distributed to all participants in the IRSRR.

4. REPORTING

4.1. EVENTS TO BE REPORTED

Unusual events that meet one or more of the following criteria could be considered as appropriate for reporting to the IRSRR:.

(a) The unusual event identifies important lessons learned that allow the international research reactor community to prevent a recurrence of a similar event or to avoid the occurrence of a more serious unusual event in terms of safety; or
(b) The unusual event is itself (potentially) important or serious in terms of its safety implications or whether it (potentially) reduces the defense in depth significantly; or
(c) The unusual event is a repercussion of similar events previously reported to IRSRR, but which identifies new lessons learned.

4.2. REPORTING CATEGORIES

Unusual events caused by any of the following categories should be reported to the IRSRR:

(a) Unanticipated releases of radioactive material or exposure to radiation
(b) Degradation of barriers and safety related systems
(c) Deficiencies in design, construction, operation (including maintenance and periodic testing), quality assurance or safety evaluation, including experimental devices and isotope production facilities
(d) Generic problems of safety interest.
(e) Consequential actions.
(f) Events of potential safety significance.
(g) Effects of unusual external events of either man-made or natural origin.

More details of the above reporting categories are given in Appendix I.

5. REPORTING FORMAT

5.1. GENERAL REQUIREMENTS

The reports sent to the IAEA should follow a standard format including the coded watchlist for computer storage. The IRSRR report should consist of the following six parts:

(a) Cover sheet
(b) Narrative description
(c) Investigation of the unusual event and safety assessment
(d) Observed causes and corrective actions
(e) Lessons learned
(f) Coded watchlist.

The reports should be written in English. Each report should be precise and "stand alone". If necessary, clear drawings may be incorporated. The use of abbreviations and symbols should be avoided. The use of SI units is recommended.

The remainder of this section gives detailed information and guidance for preparing an IRSRR report in the standard format.

5.2. COVER SHEET

5.2.1. Format

The cover sheet of each report should follow the format shown below:

INTERNATIONAL ATOMIC ENERGY AGENCY
INCIDENT REPORTING SYSTEM FOR RESEARCH REACTORS (IRSRR)

IRSRR number *	Date of receipt *
Title:	
Country:	Date of incident: Type of report: (circle one) Preliminary Main Follow-up Follow-up expected: Yes/No
Research Reactor Name:	Power: kW
Research Reactor Code:	Designer:
Reactor Type:	Start of operation:

* To be completed by IAEA.

Abstract:

5.2.2. Checklist for filling out the cover sheet

IRSRR number and date of receipt

The IAEA assigns this number upon receipt of the report.

Title

The selected title should allude to the initiating event and the final consequences, and identify the most important safety related system or component whose failure was the principal cause of the event. If several unrelated events occurred, a separate report should be prepared for each event.

Research reactor name, research reactor code, power, designer, reactor type, start of operation

These details should be filled out using the data given in the IAEA Reference Data Series on Research Reactors in the World. [3]

Date of incident

Year/month/day. If there is no specific date, as for generic reports, the year of the investigation should be given.

Type of report

An indication should be given whether the report is a preliminary, main or a follow-up.

Follow-up expected

An indication should be given whether a follow-up report can be expected or not. A follow-up report should also be a "stand alone" report rather than a supplementary report. A follow-up report should refer to the initial report in its abstract section.

Abstract

The abstract should include a brief description of the relevant information presented in the four main sections of the report (i.e. narrative description, investigation and safety assessment of the incident, root causes and lessons learned.) The abstract should contain no more than 300 words.

5.3. NARRATIVE DESCRIPTION

This section should describe the sequence of events from the initiating event to the final stabilized condition of the reactor.

The following aspects or considerations should be included in the description of the incident, when relevant:

(a) Status of the research reactor prior to the incident;
(b) How the operators became aware of the incident;
(c) Operator actions related to the incident;
(d) Systems, components or experimental devices involved in the incident;
(e) Direct consequences of the incident, including information on the affected reactor equipment;
(f) Pertinent diagrams or drawings which may help the understanding of the incident; and
(g) Other occurrences which may be related to the incident.

Because of the great variation in design and configuration of research reactors around the world, it may be useful to provide a brief description of and/or background information on the affected systems.

When a safety-related deficiency is reported, the description should indicate how this was detected.

5.4. INVESTIGATION OF THE UNUSUAL EVENT AND SAFETY ASSESSMENT

This section should describe the investigation performed to determine the causes of the incident and the assessment of the safety consequences and implications of the event. In particular, the following aspects and observations should be addressed, if relevant:

(a) Any violation of safety limits, safety system settings or limiting conditions for safe operation.
(b) Any violation of periodic inspection requirements or administrative requirements.
(c) Analysis, including calculations, to determine the causes of the event.
(d) Review of procedures.
(e) Assessment of the status of the affected items and systems important to safety.
(f) Assessment of the safety significance of the event and whether the event would have been more severe under reasonable and credible alternative conditions.
(g) Assessment of the contribution of human factors to the event.
(h) Summary of conclusions and recommendations.

5.5. OBSERVED CAUSES AND CORRECTIVE ACTIONS

The following points, if relevant, should be included in this section:

(a) Results of the observed cause investigation;
(b) Corrective actions regarding the failed/affected equipment;
(c) Actions to prevent occurrence of similar future events (e.g. modification of design, procedures or periodic testing programme, training of personnel, change of Operational Limits and Conditions.)
(d) Regulatory actions.

5.6. LESSONS LEARNED

This section should identify the lessons learned. It is recommended that the lessons learned be presented according to the following classification:

(a) Design and construction
(b) Safety analysis
(c) Operational Limits and Conditions
(d) Maintenance and Periodic Testing
(e) Procedures
(f) Utilization
(g) Radiation protection
(h) Emergency planning
(I) Quality Assurance
(j) Personnel training and qualification
(k) Equipment

5.7. CODED WATCHLIST

The IRSRR coded watch list is a simplified means to search and retrieve computerized information on events. This is achieved by assigning numerical codes to the typical systems, root causes, consequences, etc., which generally characterize research reactor incidents.

This means that the national co-ordinator should select from the pertinent “Glossary” the appropriate codes which best describe the main parameters of the reported event. If more than one code applies to the description of a particular parameter (as is often the case) all appropriate codes should be selected.

For the purpose of codification the event parameters or descriptions have been divided into nine main groups. For each of these groups all applicable codes should be filled in.

The nine event parameters are described below. The associated Glossary of Codes is provided in Appendix II.

EVENT PARAMETER		**CODES**	
1. Reporting category	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]
2. Plant status prior to the event	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]
3. Failed/affected systems	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]
4. Failed/affected components	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]
5. Cause of the event	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]
6. Effects on operation	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]
7. Characteristics of the incident	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]
8. Nature of failure or error	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]
9. Nature of recovery actions	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]	[ ][ ][ ][ ][ ]

(1) Reporting category

This field identifies the category (or categories) into which the event falls.

(2) Plant status prior to the event

This field identifies the reactor status prior to the event. Sometimes this has no relation to the incident. Even in such cases, the appropriate code(s) should be indicated as precisely as possible.

(3) Failed/affected systems

This field identifies:

(a) The systems which failed or lost their normal function, thereby initiating or triggering further steps of the event.
(b) The systems which functioned as designed but induced further steps in the development of the event.
(c) The systems which were damaged as a result of the event.
(d) The important systems which lost their normal function as a result of the event.
(e) Other systems that played a role in the development of the event.

The codes of this field are based on the mechanical/physical constitution of the research reactor. However, sometimes one part/component of a system has two or more functions. In such cases both should be indicated.

(4) Failed/affected components

This field should identify which component failed or was affected. If multiple components are affected then they should also be indicated.

(5) Cause of the event

This field identifies the observed or direct cause of the event. For a sequential event, all the observed causes of each stage should be selected. ‘Observed cause’ is a cause that is the direct initiator of the event or is the direct trigger of the next steps.

(6) Effects on operation

This field will indicate which effect on the operation of the reactor has been observed.

(7) Characteristics of the incident

This field identifies the nature of the event. In comparison with ‘reporting category’, this field describes the type of event that started the incident, or its triggering mechanism, whereas ‘reporting category’ is more related to the outcome of the event. There will typically be only one or a small number of characteristics of the event.

(8) Nature of failure or error

This field identifies the type of event. ‘Failure’ includes both physical impairment and function loss. In this coding system, ‘common cause failure’ means the multiple failures that are the result of one common cause (not limited only to the failure of redundant equipment).

(9) Nature of recovery actions

This field identifies the “first” observed recovery action of the incident.

APPENDIX I: UNUSUAL EVENT REPORTING CATEGORIES

This appendix identifies categories for reporting unusual events, giving some background information on their establishment and examples of events that may fall into them. The categories provide a basis for identifying safety-related unusual events that are expected to be reported through the IRSRR. It is important to note that a report may be prepared not only because an event has occurred, but also because a significant safety-related action has been taken as a result of findings during maintenance, periodic testing, in-service inspection, safety audits, etc. The examples given here are expected to be useful for an understanding of the categories, but do not necessarily cover every aspect of them.

PART 1: REPORTING CATEGORIES

The categories are:

1.1	Unanticipated releases of radioactive material or exposure to radiation
1.1.1	Unanticipated releases of radioactive material
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 (including experimental devices and isotope production facilities important to safety)
1.2.1	Fuel cladding failure or fuel damage
1.2.2	Degradation of primary coolant boundary
1.2.3	Degradation of containment/confinement function or integrity
1.2.4	Degradation of systems required to control reactivity and shutdown
1.2.5	Degradation of systems required to assure primary coolant inventory and core cooling
1.2.6	Degradation of essential support systems
1.2.7	Degradation of experimental devices or isotope production facilities
1.3	Deficiencies in design, construction, operation (including maintenance and periodic testing), quality assurance or safety evaluation, including experimental devices and isotope production facilities
1.3.1	Deficiencies in design
1.3.2	Deficiencies in construction
1.3.3	Deficiencies in operation (including maintenance and periodic testing)
1.3.4	Deficiencies in quality assurance
1.3.5	Deficiencies in safety evaluation
1.4	Generic problems of safety interest
1.5	Consequential actions
1.6	Events of potential safety significance
1.7	Effects of unusual external events of either man-made or natural origin

PART 2: DISCUSSION AND EXAMPLES

Category 1.1: Unanticipated releases of radioactive material or exposure to radiation.

1.1.1. Unanticipated releases of radioactive material

Discussion: Unanticipated releases of radioactive material may occur as a result of design deficiencies, exceeding safety limits or deficiencies in conduct of operations.

Examples:

(a) Release from a damaged fuel element in the core or in spent fuel storage
(b) Release from liquid or solid waste storage facility
(c) Release from irradiated capsules or from experimental devices
(d) Release of irradiated gas from beam tubes or other experimental facilities

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.

Examples:

(a) Exposure of personnel due to poor planning of maintenance tasks or fuel management or manipulation of experimental devices.
(b) Exposure due to non-compliance with operating procedures (e.g., access control procedure)
(c) Exposure following failure of fuel cladding, irradiation capsule, transfer container, etc.

Category 1.2: Degradation of barriers and safety related systems (including experimental devices and isotope production facilities important to safety)

1.2.1. Fuel cladding failure or fuel damage

1.2.2. Degradation of the primary coolant boundary

1.2.3. Degradation of confinement/containment function or integrity.

Discussion: The aim is to collect information on degradation of any of the three barriers against release of radioactive materials.

Examples:

(a) An unacceptable rate or level of fuel cladding failure in the reactor or in the storage pool that is caused by exceeding safety limits, poor water quality or design and manufacturing deficiencies. Mechanical damage of fuel elements during fuel handling should also be taken into account.

(b) Cracks and breaks in piping, in the reactor vessel or in major components of the primary coolant circuit that have safety relevance (reactor coolant pumps, valves, pool, etc.).
(c) Significant defects in welds or materials used in the primary coolant system.
(d) Loss of coolant.
(e) Loss of coolant flow.
(f) Unavailability of residual heat removal system on demand.
(g) Degradation of coolant quality (significant change in pH, conductivity, cleanliness, concentration of impurities).
(h) Loss of containment/confinement function or integrity, including leakage rates exceeding authorized limits.

1.2.4. Degradation of systems required to control reactivity and shutdown

Examples:

(a) Failure of the reactor protection system to produce a signal.
(b) Bypass or incorrect safety system setting.
(c) Failure of the reactivity control mechanism.
(d) Reduction of the shut-down margin.
(e) Failure of manual scram.

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

Discussion: The aim is to collect information on anomalies in systems which assure in normal and transient operations sufficient means to remove core power and decay heat

Examples:

(a) Degradation or failure of the emergency core cooling system.
(b) Degradation or failure of the emergency ventilation or clean-up system.
(c) Degradation or failure of containment isolation system.
(d) Degradation or failure of flap valves.

1.2.6. Degradation of essential support system

Discussion: The aim is to collect information on anomalies in safety related systems that could lead to degradation of principle barriers.

Examples:

(a) Degradation of the reactor power regulation system.
(b) Failure of the radiation monitoring system.
(c) Loss of electrical power associated with safety-related systems (e.g. loss of emergency power (Diesel generator), or DC power to instrumentation).
(d) Degradation of the water treatment systems.
(e) Loss of compressed-air for safety-related systems.

1.2.7. Degradation of experimental devices or isotope production facilities.

Examples:

(a) Degradation of experimental devices' components or their protective system.
(b) Any degradation of the integrity of experimental devices leading to significant contamination or affecting the safety of the reactor.

Category 1.3: Deficiencies in design, construction, operation (including maintenance and periodic testing), quality assurance, or safety evaluation of reactor systems, experimental devices and radioisotope production facilities.

1.3.1. Deficiencies in design

1.3.2. Deficiencies in construction

Discussion: The aim is to collect information on deficiencies in design or construction, including experimental devices, that, if uncorrected, could result in loss of a required safety function.

Examples:

(a) Degradation of materials under environmental conditions not sufficiently considered in the design stage or because in the design stage the influence was not yet known or not clearly understood.
(b) Despite proper design, errors were made during construction or installation that could influence the performance of the systems or components if not detected during testing, maintenance or otherwise.

1.3.3. Deficiencies in operation (including maintenance and periodic testing)

Discussion: The safe operation of a research reactor or its experimental devices relies to a large extent upon the skill and proper actions of the reactor personnel. As a result of deficiencies in this area, a simple event could escalate into an incident. Information in this area is important for the feedback of operational experience.

Examples:

(a) Inadvertent criticality, e.g., during in-core fuel management.
(b) Personnel errors or procedural deficiencies or shortcomings in man-machine interface resulting in loss of reactor capability to perform safety functions.
(c) Violation of licence conditions, such as operational limits and conditions, periodic testing requirements or administrative requirements.

1.3.4. Deficiencies in quality assurance

Discussion: A quality assurance programme provides a disciplined approach to all activities affecting quality, including verification of task performance and implementation of corrective actions where required. Quality assurance is an aspect of good management and related to design and construction as well as to operation. Deficiencies in the quality assurance programme might influence the safe operation of research reactors.

Examples:

(a) Wrong drawings are used for maintenance.
(b) A component was not constructed as intended in the design.
(c) Insufficient verification of accomplished work owing to deficiency in task description or indicated responsibility.

1.3.5. Deficiencies in safety evaluation

Discussion: Unanalyzed or insufficiently analyzed events might confront the operators with unexpected situations should such events arise.

Examples:

(a) Any event caused by a failure, condition or action that demonstrates an insufficient independence of safety systems and components. Safety systems and components are those needed to:

    (I) shut down the reactor and maintain it in a safe shutdown condition, or
    (ii) remove residual heat, or
    (iii) control release of radioactive material.

(b) Any event that results in the reactor not being in a controlled condition or that results in an unanalyzed condition that significantly compromises reactor safety.

(c) Any incorrect analyses of possible chemical reaction of irradiated materials.

Category 1.4: Generic problems of safety interest.

Discussion: This category is intended to include those events that individually seem not to be significant, but after recurrence indicate that a problem of safety significance could exist.

Examples:

(a) Recurring events.
(b) Events with implications for similar reactor designs.

Category 1.5: Consequential actions

Discussion: This category is intended to include significant consequential actions resulting from reported events. This includes actions taken by organizations on the basis of lessons learned from elsewhere.

Examples:

(a) Important modifications to the design basis.
(b) Changes to emergency planning.
(c) Changes to design assessment requirements.
(d) Changes to accidents analysis and evaluation.
(e) Important changes to the requirements for construction, commissioning and operation.

Category 1.6: Events of potential safety significance

Discussion: This category is intended to include events that, under different circumstances or of greater intensity, could have had safety significance.

Examples:

(a) Any event that occurs at shutdown or low power operation that would become significant for safety if it occurred at full power.
(b) Events that have no significant consequences but are considered to approach a near miss situation or are precursors of more serious events.
(c) An event that identifies a significant common cause failure.

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

Discussion: This category includes those events (acts or conditions) which might challenge the safety of the reactor.

Examples:

(a) A tornado or cyclone that affects the site.
(b) An earthquake that approaches design basis limits.
(c) Chemical explosion, fire, aircraft crash or other man-made event that affects the site.

APPENDIX II: GLOSSARY OF CODES FOR THE CODED WATCHLIST

1. REPORTING CATEGORIES

CODE	DESCRIPTION
1.1	Unanticipated releases of radioactive material or exposure to radiation
1.1.1	Unanticipated releases of radioactive material
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 (including experimental devices and isotope production facilities important to safety)
1.2.1	Fuel cladding failure or fuel damage
1.2.2	Degradation of primary coolant boundary
1.2.3	Degradation of containment/confinement function or integrity
1.2.4	Degradation of systems required to control reactivity and shutdown
1.2.5	Degradation of systems required to assure primary coolant inventory and core cooling
1.2.6	Degradation of essential support systems
1.2.7	Degradation of experimental devices or isotope production facilities
1.3	Deficiencies in design, construction, operation (including maintenance and periodic testing), quality assurance or safety evaluation, including experimental devices and isotope production facilities
1.3.1	Deficiencies in design
1.3.2	Deficiencies in construction
1.3.3	Deficiencies in operation (including maintenance and periodic testing)
1.3.4	Deficiencies in quality assurance
1.3.5	Deficiencies in safety evaluation
1.4	Generic problems of safety interest
1.5	Consequential actions
1.6	Events of potential safety significance
1.7	Effects of unusual external events of either man-made or natural origin

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
2.1.6	Pulse operation

2.1.7	Handling of experimental devices with reactor on power
2.2	Subcritical
2.2.1	Reactor in subcritical state
2.2.2	Handling of experimental devices during the subcritical state
2.3	Shutdown
2.3.1	Normal shutdown
2.3.2	Shutdown and refuelling
2.3.3	Handling of experimental devices during shutdown
2.3.4	Extended shutdown
2.4	Pre-operational
2.4.1	Construction
2.4.2	Commissioning
2.5	Testing or maintenance was being performed
2.6	Decommissioning

3. FAILED/AFFECTED SYSTEMS

CODE	DESCRIPTION
3.1	Primary reactor systems
3.1.1	Reactor core/fuel assemblies/control and shutdown rods/guide thimbles
3.1.2	Control rod drive (mechanism, motor, power supply, hydraulic system, other shutdown system
3.1.3	Reflector
3.1.4	Reactor vessel
3.1.5	Moderator
3.1.6	Core support structure
3.2	Reactor coolant systems
3.2.1	Primary coolant system
3.2.1.1	Primary cooling system (pumps and associated materials, piping, ……)
3.2.1.2	Primary system pressure control (includes primary safety relief valves)
3.2.2	Secondary cooling system (pumps and associated materials, piping, ……)
3.2.3	Emergency core cooling system
3.2.4	Residual heat removal system (including natural convection system)
3.2.5	Pool cooling
3.2.5.1	Pool cooling system (including pumps, valves, piping)
3.2.5.2	Pool integrity
3.3	Containment/confinement systems
3.3.0	Other
3.3.1	Containment/confinement integrity
3.3.2	Heating, Ventilation and air-conditioning system (HVAC)
3.3.3	Emergency isolation
3.4	Instrumentation and control systems
3.4.0	Other monitoring and control systems
3.4.1	Reactor protection
3.4.2	Reactor power control
3.4.3	Neutron flux monitoring channels

3.4.4	Process monitoring (temperature, flow, pressure, level, leak detection, ….)
3.4.5	Plant/process computer
3.4.6	Engineered safety features actuation
3.4.7	Radiation monitoring systems
3.4.8	Environmental monitoring
3.4.9	Communication and alarm systems
3.4.10	Fire detection
3.5	Electrical systems
3.5.0	Other
3.5.1	AC supply system
3.5.2	DC supply system
3.5.3	Emergency power supply system
3.6	Auxiliary systems
3.6.0	Other
3.6.1	Water purification (including coolant)
3.6.2	Compressed air
3.6.3	Demineralized water supply/make-up
3.6.4	Fuel handling and storage
3.6.5	Fire Protection
3.6.6	Sampling system
3.6.7	Spent fuel pool and/or refuelling pool cooling, including clean-up system
3.6.8	Cranes and lifting devices
3.7	Waste management
3.7.0	Other
3.7.1	Liquid radwaste
3.7.2	Solid radwaste
3.7.3	Gaseous radwaste
3.7.4	Non-radioactive waste (solid, liquid, gaseous)
3.8	Heating, ventilation and air-conditioning systems (HVAC)
3.8.0	Other
3.8.1	Containment/confinement (HVAC)
3.8.2	Control room (HVAC)
3.8.3	Spent fuel building (HVAC)
3.8.4	Waste management building (HVAC)
3.9	Structural systems
3.9.0	Other
3.9.1	Fuel storage building
3.9.2	Waste management building
3.9.3	Cooling tower
3.9.4	Pump building
3.9.5	Plant stack
3.10	Experimental devices and isotope production facilities
3.10.0	Other
3.10.1	Experimental devices
3.10.2	Isotope production facilities
3.10.3	Beam tubes
3.10.4	Hot cells

4. FAILED/AFFECTED COMPONENTS

4.0	No specific component involved
4.1	Instrumentation (gauges, transmitters, sensors)
4.1.0	Other
4.1.1	Pressure
4.1.2	Temperature
4.1.3	Level
4.1.4	Flow
4.1.5	Radiation/Contamination
4.1.6	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 measuring
4.1.11	Fire detectors
4.1.12	Hydrogen detectors
4.1.13	Electrical (current, voltage, power, ...)
4.2	Mechanical
4.2.0	Other
4.2.1	Pumps, compressors, fans
4.2.2	Generators (diesel, gasoline, ...)
4.2.3	Valves (including safety/relief/check/solenoid/natural convection valves) , valve operators, controllers, dampers and fire breakers, seals and packing
4.2.4	Heat exchangers (heaters, coolers, condensers, boilers, air dryer, ...), heat exchanger tube plugs
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
4.2.8	Strainers, screens, filters, ion exchange columns
4.2.9	Penetration (e.g., hot cells, reactor building, fuel handling, etc.)
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
4.3	Electrical
4.3.0	Other
4.3.1	Switchyard equipment (switchgear, transformers, buses, substations)
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.	(coding not to be used)
4.3.7	Relays, connectors, hand switches, push buttons, contacts
4.3.8	Wiring, logic circuitry, controllers, starters, electrical cables
4.4	Computers
4.4.1	Computer hardware
4.4.2	Computer software
4.4.3	Other computer devices (printers, etc.)

5. CAUSE OF THE EVENT

5.1 Cause

5.1.0	Unknown or other
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)
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 or core physics failure
5.1.3.1	Chemical contamination, deposition
5.1.3.2	Uncontrolled chemical reaction
5.1.3.3	Core physics problems
5.1.3.4	Poor chemistry or inadequate chemical control

5.1.3.5 Fuel metallurgy problems
5.1.3.6 Unexpected material behaviour

5.1.4 Hydraulic/pneumatic failure

5.1.4.0 Other hydraulic/pneumatic failure

5.1.4.1 Water hammer, abnormal pressure, 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

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
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, PLC (Programmable Logic Controller) and printed circuit boards
5.1.5.6 Computer software deficiency
5.1.5.7 Data input error or calibration error
5.1.5.8 Data acquisition (retrieval system)

5.1.6 Environmental (abnormal conditions inside plant)

5.1.6.0 Other internal environmental cause
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 gases, ...)
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/high humidity
5.1.7.7 Heavy rain or snow
5.1.7.8 Heavy sand stroms

5.1.8 (coding not to be used)

5.1.9 (coding not to be used)

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

5.2 (coding not to be used)

5.3 Inadequate human action – plant staff/contractor/experimentator involved
5.3.1 Maintenance
5.3.2 Operations
5.3.3 Technical and engineering
5.3.4 Management and administration
5.3.5 Experiments

5.4 Inadequate human action – 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
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 Commissioning (of new equipment)
5.4.14 Recommissioning (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 review
5.4.20 Modification implementation

5.4.21	Training
5.4.22	Actions taken under emergency conditions
5.4.23	Other activity
5.4.24	Research reactor typical operation
5.4.24.0	Others
5.4.24.1	Routine operation of experimental devices
5.4.24.2	Non- Routine operation of experimental devices
5.4.24.3	Routine installation or removing of experimental devices
5.4.24.4	Non Routine installation or removing of experimental devices
5.4.24.5	Handling of experimental devices
5.4.24.6	Operation of standard experimental devices eg., beam tubes, in-core/out of core devices, etc.
5.4.24.7	Pulse operation
5.4.24.8	Other operational modes e.g., square waves
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
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
5.5.3	Personnel work scheduling
5.5.4	Environmental conditions
5.5.5	Man-machine interface
5.5.6	Training/qualification
5.5.7	Written procedures and documents
5.5.8	Supervisory methods
5.5.9	Work organization
5.5.9.0	Others
5.5.9.1	Shift/team size or composition
5.5.9.2	Planning/preparation of work
5.5.10	Personal factors
5.5.10.0	Others
5.5.10.1	Fatigue
5.5.10.2	Stress/perceived lack of time/boredom
5.5.10.3	Skill of the craft less than adequate/not familiar with job performance standards
5.6	Management related causal factors and root causes
5.6.0	Others
5.6.1	Management direction
5.6.2	Communication or co-ordination
5.6.3	Management monitoring and assessment
5.6.4	Decision process
5.6.5	Allocation of resources

5.6.6 Change management
5.6.7 Organizational/safety culture
5.6.8 Management of contingencies
5.7 Equipment related causal factors and root causes

5.7.0 Others
5.7.1 Design configuration and analysis
5.7.2 Equipment specification, manufacture and construction
5.7.3 Maintenance, testing or surveillance
5.7.4 Ageing

6. EFFECTS ON OPERATION

6.0 Unidentified or 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
6.4 Activation of engineered safety features
6.5 Challenge to safety or relief valve
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

7. CHARACTERISTICS OF THE INCIDENT

7.0 Other characteristics
7.1 Degraded fuel
7.2 Degraded reactor coolant boundary
7.3 Degraded reactor containment/confinement
7.4 Loss of safety function
7.5 Significant degradation of safety function
7.6 Failure or significant degradation of 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 or emergency 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 incident
7.15 Radwaste incident
7.16 Security, safeguards, sabotage or tampering incident
7.17 Degradation or malfunctioning of experimental devices

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. NATURE OF RECOVERY ACTIONS

9.0 Not relevant
9.1 Recovery by human action
9.1.1 Recovery by foreseen human action
9.1.2 Recovery by unforeseen human action
9.2 Recovery by automatic plant action or by design
9.3 No recovery

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