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Operational safety experience feedback by means of unusual event reports

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FOREWORD

Operational experience of nuclear power plants can be used to great advantage to enhance safety performance provided adequate measures are in place to collect and analyse it and to ensure that the conclusions drawn are acted upon.

Feedback of operating experience is thus an extremely important tool to ensure high standards of safety in operational nuclear power plants and to improve the capability to prevent serious accidents and to learn from minor deviations and equipment failures — which can serve as early warnings — to prevent even minor events from occurring.

Mechanisms also need to be developed to ensure that operating experience is shared both nationally as well as internationally. The operating experience feedback process needs to be fully and effectively established within the nuclear power plant, the utility, the regulatory organization as well as in other institutions — such as technical support organizations and designers.

The main purpose of this publication is to reflect the international consensus as to the general principles and practices in the operational safety experience feedback process. The examples of national practices for the whole or for particular parts of the process are given in annexes.

The publication complements the IAEA Safety Series No. 93 "Systems for Reporting Unusual Events in Nuclear Power Plants" (1989) and may also give a general guidance for Member States in fulfilling their obligations stipulated in the Nuclear Safety Convention.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

In September 1994 the Nuclear Safety Convention was opened for signature by the Member States of the IAEA. Article 19, Sections (vi) and (vii) are very pertinent to the field of Operational Safety Experience Feedback (OSEF), in text of this document. These sections are reproduced here, and state that:

- incidents significant to safety are reported in a timely manner by the holder of the relevant license to the regulatory body;
- programmes to collect and analyse operating experience are established, results obtained and 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.

The IAEA Safety Fundamentals, Safety Series No. 110, The Safety of Nuclear Installations, describes in Section 6, Verification of Safety, that safety verification also means that the operating organization has the responsibility to ensure that events important to safety are reviewed in depth and that, when necessary, equipment is modified, procedures are revised and training is given to prevent recurrence. Access to information and relevant experience from similar installations worldwide is essential in such reviews.

The role of OSEF for the safety of NPPs is discussed in INSAG Technical Note No. 2 "The importance for nuclear safety of efficient feedback of operational experience". The document states that the objective of the feedback system is to ensure that all available information is properly used to improve safety.

Several meetings, symposia and other national or international actions underlined the importance of OSEF in the last years. In the framework of the IAEA Incident Reporting System (IRS) it was decided to summarize national practices in the OSEF domain.

The IAEA convened from 21 to 25 June 1993 a Consultants meeting with the aim of analysing various steps of the OSEF process and to develop a specific questionnaire for Member States to obtain information on national models and practices in this area.

The next step undertaken by the IAEA to summarize national practices in the OSEF area was a Technical Committee Meeting on Compilation of National Practices on Operational Safety Experience Feedback which was held from 16 to 19 May 1994. The meeting provided a broad forum for in-depth presentation and discussion of various models and approaches in the OSEF area in Member States. The meeting recommended broader promotion of OSEF in other IAEA activities including missions and seminars.

The Consultants Meeting on Operational Safety Experience Feedback based on the use of unusual event reports held from 7 to 10 November 1994 evaluated the compiled information and prepared the first draft of this publication.

1.2. OBJECTIVES

The primary objective of this publication is to provide advice on what constitutes practices in the field of Operational Safety Experience Feedback as applied to nuclear power plants, using information from national systems supplied from Member States. It is acknowledged that OSEF systems can, and do, vary in different member countries depending on, amongst other things, the level of involvement of the utility or regulator. The objective is, therefore, not to be prescriptive, but to draw together common general elements of the Operational Safety Experience Feedback process.

The OSEF process is undertaken by different organizations (licensees, regulators, designers, international organizations) who, by co-operating, ensure that the total process is efficient and effective.

A secondary objective is to identify the various organizations, their roles, responsibilities, and the timing of their involvement in the whole process.

1.3. SCOPE

It is intended that all aspects important to Operational Safety Experience Feedback will be covered in this publication. The objectives above, have set the broad boundaries of the scope of this work. The scope will therefore include:

(a) The main elements of the OSEF process as applied to nuclear power plants;
(b) The different organizations involved and their roles, responsibilities and the timing of when they become involved.

To be effective, OSEF needs to draw on the experience of the international community. There exist international incident reporting systems, for instance, such as IRS and the WANO for NPPs operating experience information exchange programmes whose aim is to ensure the exchange of operational safety experience among participating members. The scope of this publication will also cover links between national and international systems and the reporting systems mentioned above that are used to convey information on root causes, lessons learned and corrective actions taken. For OSEF to be efficient and effective it needs to be monitored. The regulatory body has an obvious role in this process but the utility or any other organization with responsibilities in certain parts of the OSEF process may also want to audit it for their own purposes.

1.4. STRUCTURE

Operational Safety Experience Feedback (OSEF) is a process made up of a number of constituent parts. The eight general elements that make up this process are described in Section 2.

The nuclear industry takes in a large number of countries throughout the world. The lessons learned from plants abroad can be as important as incidents occurring in ones own country. Therefore to be effective, worldwide, the OSEF process needs to have methods of transmitting information efficiently between different countries. Section 3 describes the links between national and international OSEF systems which exist to make the international OSEF process effective. At the national level a number of bodies are involved to varying degrees in OSEF. The regulator, in particular, may not be heavily involved in the detail but requires to be assured that the process is working efficiently. The ability of the system to be monitored by the regulatory body, the utility, or even at plant level is therefore important. Section 4 outlines how this might be achieved.

2. GENERAL ELEMENTS IN OSEF

The overall objective of OSEF is ultimately to enhance safe and reliable nuclear power plant operations. Although the details of the national systems vary from country to country, in all cases the systems are established in order to ensure that both national and international safety related events are systematically analysed and appropriate actions are taken to reduce and hopefully prevent the occurrence of similar events. An effective system for the operational safety experience feedback covers the following:

(1) Reporting of unusual events from plants.
(2) Storage, tracking and documentation system.
(3) Screening of unusual reports to select those for detailed evaluation especially with regard to safety significance.
(4) Analysis of safety significant events.
(5) Recommended actions resulting from the assessment.
(6) Approval of recommended actions depending on significance and national practice.
(7) Implementation of actions and evaluation of benefits in operation.
(8) Dissemination and exchange of information including international systems.

The above eight elements generally describe the important components that need to be addressed in the development and implementation of an OSEF programme.

However, in order to be efficient and effective there needs to a resource commitment from management in the various participating organizations that make up the programme as well as a desire to ensure that it remains efficient and effective.

Within this process there are many organizations that have various responsibilities in the eight elements. These include: the plant; utility; research organizations; plant designers; regulatory bodies; and international co-ordinators.

Figure 1 shows a simplified flow diagram for the transmission of information from one organization to the next.

The incident starts at plant level and is communicated to the utility and then if necessary on to the regulator, other utilities/plants, research organizations and designers all within the national boundary where the incident occurred. If the incident has lessons to be learned for other countries it is sent to the international body (the IAEA/OECD for IRS reports) for worldwide distribution down to plant level.

Therefore the flow of incident information can run its full course from one country to another via the international co-ordinating agencies.

At each of these steps in the process of the dissemination of information a number of the eight elements are involved. Screening and analysis for example are two important elements which are used many times in the information flow. This ensures that the incident reports passed on have relevance and use.

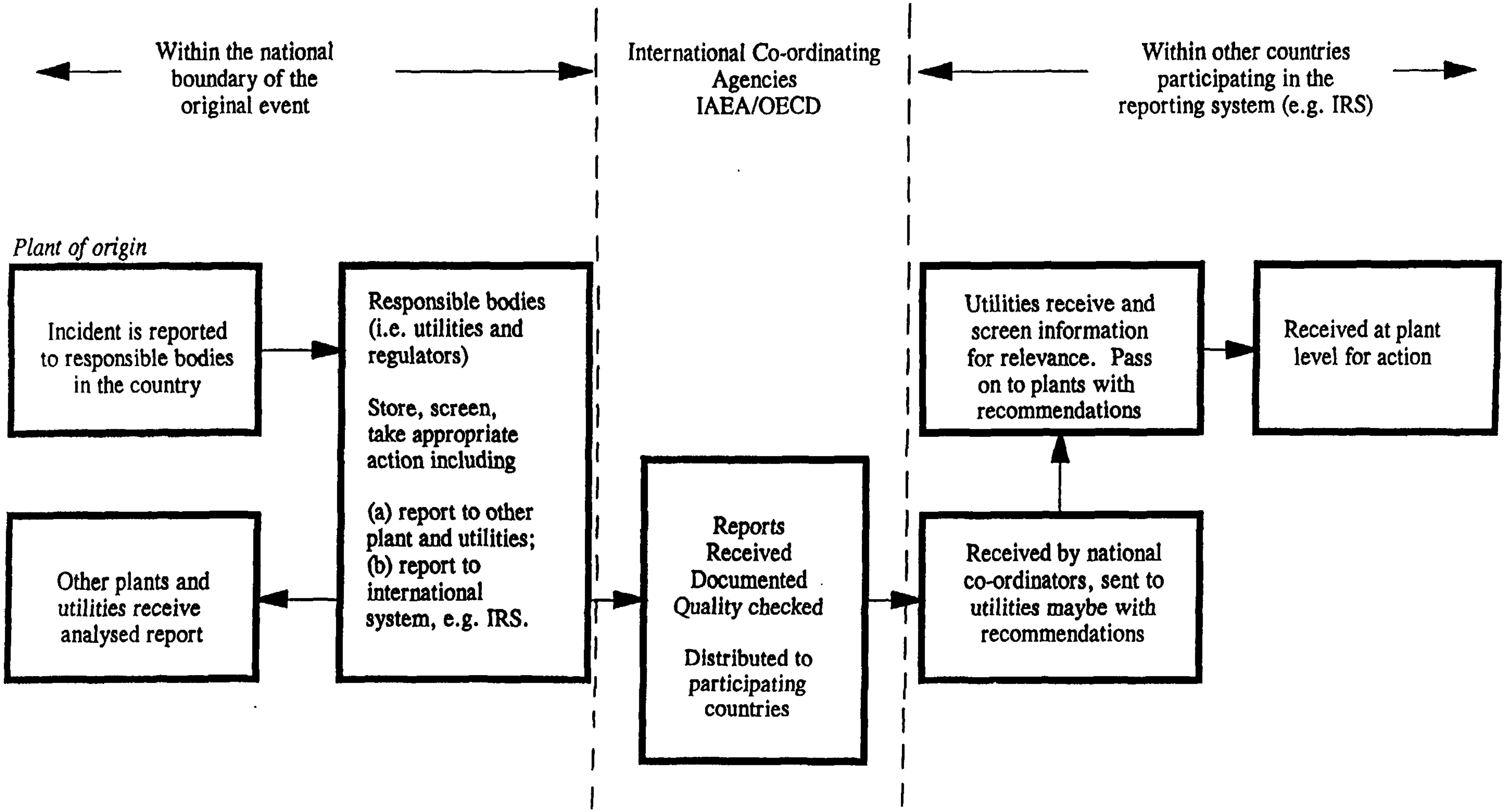


FIG. 1. Incident information flow diagram.

2.1. THE REPORTING OF UNUSUAL EVENTS

The reporting process is initiated when the reporting criteria are met. These criteria may be defined on different levels according to the requirements of the regulatory body, the utilities and vendors, etc. The reports to the regulatory body in compliance with regulatory requirements meet three basic attributes:

- reporting criteria,
- reporting format,
- reporting timing.

Within this report they are referred to as Unusual Event Reports (UERs).

The reporting to the regulatory body is carried out by each plant either directly or through the headquarters of the utility. Various reporting criteria, procedures and categories can be used.

However, the reporting criteria thresholds are defined in such a way that no safety relevant event is lost.

The following reporting categories (1-7) are recommended by the IAEA Safety Guide No. 93 on Systems for Reporting Unusual Events in Nuclear Power Plants, to identify events having an actual or potential safety significance.

(1) Release of radioactive material or exposure to radiation

    (a) Release of radioactive material that exceeds prescribed limits whether it is confined to the site or extends beyond it, or
    (b) Exposure to radiation that exceeds prescribed dose limits for site personnel or members of the public.

(2) Degradation of items important to safety

    (a) Fuel cladding failure, or
    (b) Degradation of the primary coolant pressure boundary, main steam or feedwater line, or
    (c) Degradation of containment function or integrity, or
    (d) Degradation of systems required to control reactivity, or
    (e) Degradation of systems required to control the system pressure or temperature, or
    (f) Degradation of essential support systems.

(3) Deficiencies in design, construction, operation (including maintenance and surveillance), quality assurance or safety evaluation.

(4) Events indicating generic problems of design, construction, operation, quality assurance or safety evaluation.

(5) Events leading to important modifications of design, construction, operation (including maintenance and surveillance) and quality assurance as a consequence of events that have occurred in other plants.

(6) Events of potential safety significance.

(7) Unusual events of either man-made or natural origin that directly or indirectly affect the safe operation of the plant.

Events that do not reach the threshold of reporting criteria, but still are safety significant, such as safety systems unavailabilities, or the finding of a defective component of safety systems upon performing preventive maintenance, etc., are also recorded and analysed at the plant level. These records are loaded into appropriate databases for statistical analysis and consulting.

In some countries, such records are also stored in a centralized databank accessible to both regulators and licensees. Such are the cases of Nuclear Reliability and Performance Data System (NRPDS) in the USA and Nuclear Component Databank (DACNE) in Spain.

Annexes I-IV present examples of national practices (Bulgaria, Spain, Finland, Russia and the Ukraine) regarding reporting criteria for UERs. Annexes V-X present examples of national overview of national legal framework and practices in OSEF (Czech Republic, France, Germany, Japan, USA, Argentina).

Detailed requirements concerning the content and format of an unusual event report are given in the IAEA Safety Guide No. 93. In accordance with this document the report on an unusual event should include the following:

(1) NARRATIVE DESCRIPTION

Operational state prior to the event including status of relevant systems, event description in chronological order, component or system faults, findings during maintenance and surveillance, operator actions/procedural controls, short term actions, previous occurrences and consequences.

(2) SAFETY ASSESSMENT

This should contain an assessment of the safety consequences and implications of the event. The primary aim of the safety assessment is to ascertain why the event occurred and whether the event would have been more severe under reasonable and credible alternative conditions, such as different power levels or operating modes; in addition, the safety significance should be pointed out.

(3) ROOT CAUSES AND CORRECTIVE ACTIONS

This should analyse the root cause of the event down to human factors (including procedures), man-machine interfaces or design and manufacturing; actions taken or planned, including those to reduce the probability of similar future events.

(4) LESSONS LEARNED

The importance of the event with respect to the lessons learned and their classification identified with each lesson learned.

It is important that written UERs contain sufficient technical details for an understanding of the event.

For an easy retrieval the reports have a unified cover sheet with basic information such as title of event, dates, name of plant, reactor type, power, designer, start of operation and coded watchlist such as the system used for IRS report (Safety Series No. 93).

The prompt UER format usually contains the same information as the front page of the full UER plus a short abstract of the event.

Timing of reporting has generally two phases: prompt UER based on preliminary assessment of safety significance of the events (by phone or fax within 24 hours) and full UER submission (usually within one month but within a margin of 10 days - 3 months). Well-defined reporting provides a better guarantee that all generic concerns of safety related events are reported to the nuclear community.

The number of UERs reported usually depends not only on the frequency of such events occurring but also on such specific plant factors as design, age, operating practices, etc., which may affect the number of reports. Experience in some Member States has shown that an average of 10 to 30 events per year per operating unit are reported.

2.2. STORAGE, TRACKING AND DOCUMENTATION SYSTEM

The UERs are stored centrally by both the regulatory body, and the utilities or by independent organizations supporting the regulatory body or the utilities.

Different techniques are used for storage, tracking and documentation of national unusual event reports from simple storage of hard copies to computerized full text databases with the keywords and coded information enabling quick search and retrieval of information.

Computerized information management is usually utilized when the amount of information justifies it. Such a system should ideally be adapted to enable finely structured searches to be made for information needed to support in-depth safety investigations, or to perform trend and pattern analysis and generic studies.

To permit searches for information, a typical list of uniquely retrievable fields for each reported event are recommended in the Safety Guide:

(1) Name or unique identification of the plant.
(2) Main plant features.
(3) The date when operation commenced.
(4) Unusual event number (the report number and a revision number that uniquely identifies each report).
(5) Event date.
(6) Unusual event report abstract.
(7) Operational state (e.g. power level, hot shutdown, refuelling).
(8) The systems involved and their status.
(9) The components involved - type, manufacturer and number of the same type that faulted.
(10) The fault mode (e.g. valve failed to open, operator performed an undesirable action).
(11) Cause of the system or component fault, or personnel error, or the shortcomings in the man-machine interface.
(12) Whether the system or component fault was immediately evident, had existed for some period, or was incipient.

(13) The type of fault - total or partial failure.
(14) The method by which the system or component fault or personnel action was discovered (e.g. operational event, inspection, testing, alarm, etc.).
(15) The effect of the system or component fault, or the personnel action, on the systems and plant (e.g. produced a trip, reduced power, etc.).
(16) Type and characteristics of personnel error.
(17) Characteristics of any radioactive releases or personnel exposures (e.g. radioactive releases to containment only or to the environment, external or internal exposure of personnel).

An example of an effective national computerized database of UERs is given in Annex XI (Sweden).

2.3. SCREENING

Screening is undertaken to ensure that no safety relevant experience is overlooked and that no applicable lesson learnt is disregarded. The main goal of the screening process is to identify those events that are selected for further detailed investigation and analysis. This would also include prioritization according to safety significance. The quality of screening depends on engineering judgement, therefore highly experienced and knowledgeable engineers are devoted to this task.

Within the national system screening takes place at the plant/utility level and also is carried out by the regulatory body/authority. The focus of screening is different at the plant/utility level and at the regulatory body/authority level.

2.3.1. Screening at plant level

At the plant level, two types of screening are carried out: in-house and outside experience.

In-house events are those that occur at the plant. External are the experience coming from outside, either from the same country or a foreign one, in nuclear power plants with similar or different technology.

This first screening of internal events is carried out on line, i.e. as they occur. So that, the shift supervisor is responsible for the screening, until someone else takes over, if needed. Therefore, the shift supervisor is trained appropriately on reportability criteria.

Screening of internal events is performed by a multidisciplinary group of plant engineers regularly. This group reviews every safety significant event occurring at the plant and discusses whether or not the causes have been clearly identified and the corrective actions taken or planned and are commensurate with the causes.

Screening of external events at the plant level may be either on-line or undertaken periodically as long as the intervals between screenings are not too long.

As for the inputs to be analysed the sources of them are identified, as well as the type of inputs, the way they follow inside the plant organization until they arrive to the OSEF engineer(s) in charge of the screening.

A compromise needs to be established whereby the effort devoted to analysing information needs to be considered against the effort and cost of undertaking it.

The screening of external operating experience consists of deciding whether the considered item is applicable or not to the plant. Those inputs considered applicable are distributed to the specific branches (operations, maintenance, etc.) for information.

2.3.2. Screening at national level

At the national level, the screening may be carried out by the regulatory body or other authority. In countries with a large nuclear industry, this screening is made in parallel by several organizations: (a) the regulatory body that focuses on safety concerns, (b) utilities associations, that also pay attention to economic aspects, (c) vendors and designers, that use operating experience to improve designs.

The regulatory body makes two types of screening: national events and foreign experiences.

In some countries, the regulatory body has responsibility for conducting the operating experience feedback. In this case, the regulatory body undertakes the steps described above.

In other countries, the regulatory body just monitors that the OSEF programme operates the functions as intended with the support of utilities or vendors' organizations. In both cases some screening has to be made to ensure that the OSEF is working as expected.

A multidisciplinary group made of experienced engineers of different specialities (human factors, PSA, safety systems, root cause analyses) meets regularly to discuss every UER in order to make sure that no safety significant aspect of any event has been overlooked, and that no applicable generic lessons have been disregarded. This group is referred to as the Events Assessment Group (EAG). It can also consider unfavourable trends, or events that considered individually are not very important, but whose repetition may show a pattern that has to be corrected. The frequency of the meeting of the EAG varies from weekly to monthly depending on the workload.

Annex XI contains information on how the screening is carried out in Sweden.

2.4. ANALYSIS

An assessment should include detailed investigation into the probable causes of the unusual event and any contributing factor as well as identification of corrective actions needed to minimize the likelihood of the same or similar events occurring in the future.

Depending on the nature of the event, the following type of specific assessment may be performed:

(1) Detailed investigation of operator actions during the unusual event and the adequacy of operating procedures.
(2) Determination of the adequacy of predictive methods (e.g. fault tree analysis) by comparing new or past event simulations with the actual event.
(3) Detailed investigation of the consequences, using applicable structural analyses (e.g. stress analysis of pressure retaining components).
(4) Determination of the consequences of the event including, or associated with, secondary

faults (e.g. faults caused by associated environmental conditions).

(5) Evaluation of whether the event would be significantly more serious under different permitted operating conditions (e.g. higher or lower power levels) or whether a precursor to a significantly more serious fault would have occurred if other personnel or equipment faults were involved.

(6) Identification of corrective actions needed to minimize the likelihood of the same or similar events occurring in the future.

Those who undertake these assessments shall have full access to information concerning the unusual event. During preparation of the assessment report other appropriate organizations may be contacted for comments and discussion.

Events identified for analysis are assessed using appropriate root case investigation techniques. Some of the methodologies available are the Management Oversight and Risk Tree (MORT) analysis, the Human Performance Enhancement System (HPES) of the Institute of Nuclear Power Operations (INPO), and the Analysis of Safety Significant Events Teams (ASSET) of the IAEA, and there are many others.

Most of these methodologies make use, in some degree, of more elemental techniques such as barrier analysis, change analysis, event and causal factor analysis, and fault tree analysis.

Depending on the nature of the event or the investigation scope, some methodologies can be more appropriate in a case by case basis. So for example, MORT is especially appropriate for investigation of events in which organizational and/or management weaknesses are part of the causes. HPES is more human factor oriented and ASSET is especially appropriate for events showing weaknesses of the plant surveillance programme. It is recognized that any of those methodologies and others, if applied in due manner by well trained and experienced persons, produce beneficial results.

An investigation team should include at least one professional well-trained in root cause analysis techniques.

During the investigation, all similar events are identified and assessed using available databases. When equipment failures are included, equipment failure databases or maintenance records for similar components are reviewed. As part of the assessment, the effectiveness of corrective actions from previous events are evaluated and weaknesses identified. Where the evaluation of previous events has not resulted in effective corrective actions, weaknesses in the OSEF programme are identified and corrected.

Typical in-house investigation reports should include an event description, a description of performance anomalies, a discussion of root causes, a discussion of the lessons learned, and a statement of the specifically identified corrective actions that are recommended.

When interviewing personnel to obtain information, the investigator stresses that he/she is looking for facts and a blame-free atmosphere is established; if needed, confidentiality is guaranteed to interviewees.

As described above, the conduct of event analysis is applicable to both plant and regulatory body investigations.

Plants are responsible for analysing their own events. The regulatory body usually relies on plant analyses, but it can perform its own analyses if needed for safety significance and/or potential generic implications of an event or if the plant analysis is not adequate.

At plant level, external reports identified as applicable to the plant are subsequently analysed. The result of the analysis is plant specific actions.

The lessons learned from the event are identified, reviewed and addressed by the OSEF programme. The review process following the analysis consists of identifying lessons learned, root causes, and plant specific actions necessary to enhance the safety and reliability of the plant. Applicable safety significant outside reports on events with high risk of having adverse consequences at the plant are prioritized in the analysis and implementation process.

Once the corrective actions derived from outside events analysis are decided, they have the same consideration as those derived from an in-house event, i.e. the priority depends on their safety significance rather than their origin.

Annexes XII-XIII contain examples of national practices (Finland, UK) of incident investigation methods and practices.

2.5. RECOMMENDED ACTIONS

Recommended actions follow from the analysis of the event. The organization/authority proposing the recommendation would be that which undertook the analysis. These could consist of changes to any of the following:

- plant improvements, modifications, or design changes;
- procedures for operation, maintenance or testing;
- personnel, i.e. deployment or the necessity for training.

Detailed categories of corrective actions that are identified as a result of a detailed analysis of an unusual event are taken from the Safety Guide and added to. These recommended actions can result in changes to:

(1) Operation, testing, calibration, maintenance, surveillance or inspection procedures.
(2) Operating margins.
(3) Quality assurance programme.
(4) Component design or location.
(5) System configuration or location.
(6) System or component reliability.
(7) Safety analysis methods or assumptions.
(8) Safety design standards.
(9) Regulatory processes.
(10) Design methods.
(11) Construction methods.
(12) Commissioning methods.
(13) Radiological protection.
(14) On-site emergency planning.
(15) Training programmes and implementation.

The corrective actions apply in the first instance to the affected plant but these may also be applicable generically to other operating plants including plants under construction. This may result in changes to technical specifications, to the design or require a research programme to investigate the feasibility of various design improvements.

Three important factors to be taken into account while deciding about corrective actions are:

- cost against perceived benefit,
- changes in existing procedures,
- the need for training with regard to the new procedures.

Corrective actions may need to be undertaken quickly, or even as a matter of urgency, to bring the plant back to a safe condition. It may be necessary, in certain circumstances, to prevent the plant from operating until such matters have been dealt with, to the satisfaction of the utility and the regulatory body.

Corrective actions identified in OSEF may have a different impact on an operating plant, plant under construction and plant still in the design stage. It may not be cost effective to implement the whole package of corrective actions on a plant already built, as against a plant which is still in the design stage. In the latter case it may be cost effective to take the corrective actions but not for other cases.

Annex XIV presents list of actions taken in response to events reported to the IRS in areas of emergency core cooling system and human performance in maintenance and testing in Spain.

2.6. APPROVAL OF RECOMMENDED ACTIONS

The level of recommended actions may be different depending on the safety potential and national practice.

In some countries a categorization system exists for plant modifications for different levels of safety significance which requires different levels of documentation, approval and quality assurance procedures. The higher the level of safety significance the higher the level of approval which will need to be sought. These levels of approval exist at:

(a) plant level - not likely to impact safety;
(b) utility level - not likely to have a significant effect on safety;
(c) regulatory body - likely to have a significant effect on safety.

Documents are required to be prepared consisting of:

- detailed description of proposed corrective action (including drawings, schemes, etc.);
- safety analysis proving that corrective action improves safety and has no adverse effects;
- quality plans justifying materials used, design standards
- plans for implementing the corrective actions;
- procedures for undertaking a safe method of working.

At each level of approval, either at plant level, utility level or at the regulatory body level, specific conditions may be placed on how the process may be undertaken.

2.7. IMPLEMENTATION

Implementation of corrective actions is the responsibility of individual plants. Quality assurance review processes may need to be applied to ensure that corrective actions are implemented according to approved conditions. A quality assurance review process should be applied by the utility to ensure that corrective actions are implemented as approved and overseen, if necessary, by the regulatory body. Any actions that were deemed necessary by any of the approving bodies need to be undertaken and, if necessary, documented. A later evaluation should sometimes be done to check the effectiveness of the actions implemented. In this way, the direct feedback loop is closed by the implementation of specific corrective measures.

Decision making persons are heavily involved in the feedback process at the steps 6 and 7 of the OSEF to facilitate an understanding of the decision making process.

If the implementation of recommended actions is needed urgently, assessment and detailed evaluation could be done quickly, if necessary, and approved corrective actions could be immediately implemented.

On a longer timescale the regulatory body may want to monitor the situation in relation to the implementation of recommended actions. This may be done by requiring plants/utilities to provide periodic progress reports relating to the status of outstanding recommended actions.

2.8. DISSEMINATION AND EXCHANGE OF INFORMATION

Dissemination and exchange of information lies at the heart of OSEF. To have the maximum impact and benefit information needs to be disseminated to many relevant bodies that can make use of it. This needs to occur at the following levels:

- plant level,
- utility level,
- national level, and also at
- international level.

Within this framework there will exist other bodies at national and international levels that could benefit from OSEF including:

- research and development organizations, and
- design companies.

It is expected that the organization providing the detailed information to other interested bodies, described above, will be that which performed the assessment, and recommended the corrective actions. Normally, these organizations are the utilities. In this way, various organizations can benefit from the general feedback process in different ways. Operating personnel can learn to recognize events more easily and to be more sensitive to operating conditions and plant response characteristics. Vendor companies should be able to improve their design and manufacture by incorporating the lessons learned. Research organizations should receive additional guidance for establishing research goals and programmes, and technical universities should obtain an additional means of improving their knowledge. In this way, dissemination should lead to a more broadly based effort to improve safety by using operating safety experience and it may constitute an input to decision making and to the effort of the regulatory body in assuring the public on the safe operation of nuclear power plants. Reporting to the international systems of those unusual events with safety significance which may be of interest to the international nuclear community should be done by the appropriate organizations.

Annexes XV-XVII present examples of national practices (Spain, Slovenia, South Africa) in dissemination and exchange of information on UERs.

3. LINKS BETWEEN NATIONAL AND INTERNATIONAL OPERATIONAL SAFETY EXPERIENCE FEEDBACK

Effectiveness of the national operational safety experience feedback systems can be significantly strengthened by linking with international systems.

The primary objective of such international operational safety experience feedback programmes is to use lessons learned at other nuclear power plants in different countries. Such linking is extremely important for countries with a small number of operating units and for such types of reactors which are in use in several countries. The links between national and international operational safety experience feedback systems broadens the sources of information on safety significant events, the related lessons learned, and corrective measures taken on the plant level or regulatory level.

Participants in such international systems can benefit from each other's experience in reducing the risk of duplication of effort and optimizing the use of limited resources for running the programmes in the OSEF domain.

There are many different sources of information on international OSEF. The sources can be grouped into the following categories:

- Organizations formed by utilities to exchange information on international operational experience, such as the World Association of Nuclear Operators (WANO) and the Institute of Nuclear Power Operations (INPO).

  WANO is formed by nearly all worldwide utilities and uses four types of report: the Event Notification Report (ENR), the Event Analysis Report (EAR), the Event Topical Report (ETR) and the Miscellaneous Event Report (MER).

  INPO is formed by nuclear utilities from the USA and more than a dozen other countries. The most significant types of report issued by INPO are the Significant Operating Experience Report (SOER), the Significant Event Report (SER) and the Significant Event Notification (SEN).

- Systems formed by Member States of the IAEA and OECD: the most significant source is the IAEA/NEA Incident Reporting System (IRS).

- Organizations formed from vendor and owners groups (WOG, COG, BWROG, WWER).

- Bilateral agreements between government agencies/regulatory bodies on exchange of operating experience.

Despite the differences in existing international systems even generalized mutual awareness of the reported events could be of high interest for national regulatory bodies or for utilities. Free exchange of information and completeness of international databases (i.e. all significant events are reported) is an important factor to ensure the effectiveness of the international OSEF. Therefore, it is important in ensuring that the threshold of reporting criteria guarantees that all significant events are collected.

Participation in international OSEF systems requires the establishment and harmonization of relevant parts of national OSEF systems. National OSEF systems need to have procedures to deal with international information from receipt to dissemination. Since the numbers of international reports are high, the initial screening could sometimes be carried out efficiently by national utility organizations or, for larger utilities, by corporate office organizations responsible for dissemination of external operational experience to facilitate the use of international OSEF.

The reports, screened for applicability in the initial screening process and forwarded to the plant, should be screened again at utility level and yet again at plant level. This screening consists of evaluating the specific applicability and the possible effects on the plant, and of estimating the potential for the event to occur at the plant.

Reports identified as applicable to the plant should be subsequently analysed. The result of the analysis would be plant specific actions. The report should be considered together with other similar in-house and outside of operational experience. This is to ensure that all aspects of events and event trends are considered.

The lessons learned from the event should be identified, reviewed and addressed by the OSEF programme. The review process consists of identifying lessons learned, root causes, and plant specific actions necessary to enhance the safety and reliability of the plant. Applicable international OSEF reports on events with high risks of having adverse consequences at the plant should be prioritized with regard to the review and implementation process.

4. MONITORING OF OSEF PROGRAMME

The organizations that are interested in monitoring the OSEF programme are the regulatory body and the plant/utility.

The reason that the OSEF programme should be monitored is to ensure that it remains efficient and effective. There are a number of ways that this can be achieved and some of these are:

- to audit the OSEF programme arrangements at plant and utility level;
- to apply trend and pattern analysis for measurable indicators such as:
    - performance indicators;
    - repeat events.

The plant/utility reviews periodically the effectiveness of the OSEF programme. The purpose of this review is to provide feedback to the utility management on the overall programme effectiveness and to recommend remedial measures to resolve weaknesses identified. Indicators of programme effectiveness include the number, severity, recurrence rate of events, common of root causes for different events. This review also checks out that corrective actions derived from OSEF are implemented in a timely manner and are effective to solve the original problems.

For this to take place, it is necessary that the licensee has procedures defining how every element of Section 2 is carried out, from reporting to dissemination of information, and who is responsible for every step of the OSEF.

The whole documentary history of each step of the OSEF programme should be available to be audited/monitored in this process.

A periodical report, usually annual, summarizes the activities performed over the interval considered in the framework of OSEF, i.e. it lists the in-house events and outside experience that have been analysed, the corrective actions approved and the status of their implementation. For those corrective actions not yet closed out, a foreseen deadline for completion is planned.

The OSEF system is audited by the plant/utility at regular intervals, usually annually, by an experienced group not directly involved in OSEF on that plant. This audit team is usually made up of quality assurance people belonging to the same licensee and at least one member of a different plant. The independent audit team acts on behalf of the senior management of the utility, to whom the audit conclusions are reported.

Some criteria universally used for OSEF effectiveness are: (a) all applicable outside experience is analysed, (b) all in-house safety significant events are used in the OSEF process, (c) in the case of in-house events no single root cause is pervasive, (d) corrective actions are fully and timely implemented, and (e) the performance of the plant, for safety significant events, deviations or anomalies, response to challenges to safety systems (e.g. electrical grid disturbances), unavailability of safety systems trains, show no negative trend over the period considered.

Problems or deficiencies noted in the audit report about the overall administration or function of the operating experience programme are identified and discussed with plant/utility senior management. Failure to fully satisfy some of the audit criteria may not be indicative of an unsatisfactory programme, however, identified weaknesses are assessed to determine their impact on the overall effectiveness of the programme. After discussion with the senior management remedial measures are proposed to solve identified weaknesses. Based on this, senior management makes decisions for improvement of the OSEF process. The implementation of approved recommended measures to address identified weaknesses are undertaken to ensure timely completion.

The regulatory body makes similar audits to the plants, but usually at longer intervals. It also supervises that all OSEF participants in the country, such as plants, utilities, vendors, perform their job as intended. From a national standpoint, it is easier to assess global trends of plant performance, because statistics are more representative in greater numbers. Comparisons of broad trends with other countries are used as valuable references.

As for the plant specific approach, the regulatory body compares performance patterns of the country's plants or plants belonging to the plants of the same generation, designed by the same vendor and having comparable electrical output. The differences so noted are examined closer, because sometimes they are based on differences of design, age of the plant, etc. Once an adverse plant specific pattern is identified (e.g. design flaws are outstandingly higher in one specific plant), this is discussed with the plant/utility, investigated in depth and the appropriate actions taken.

On the part of the IAEA, it surveys the overall IRS report rate production, reviews every report for quality and compliance with the format and spells out its conclusions in the annual Technical Committee Meetings of IRS co-ordinators, where they are discussed and the appropriate remedial actions agreed upon.

The IAEA can help in evaluating, always upon request, plant OSEF or national OSEF system, through its Assessment of Safety Significant Events Team (ASSET), Operational Safety Review Team (OSART), Assessment of Safety Culture in Organizations Team (ASCOT) services and others that can be tailored to the interested organization or specific request.

Annex XIX presents example of supervision and documentation of OSEF process in Spain.

Annex I
REGULATORY BODY REPORTING CRITERIA FOR UERs IN BULGARIA

PROMPT COMMUNICATION

Report within one hour on the occurrence of any nuclear or radiation emergency, which may cause a spread of radioactive material beyond the prescribed zone.

INITIAL NOTIFICATION WITHIN 24 HOURS

1. Events, connected with the storage and operations with nuclear fuel

2. Operational events

   2.1. All disturbances related to the primary circuit integrity or connected with the reactor power control, reactivity control, pressurizer level or disturbances of pressure and temperature in the reactor core.

   2.2. Failure of an emergency protection system or another safety system to begin the execution of or to complete the function provided in the project, regardless of the case.

   2.3. Reactivity changes as a result of an unplanned situation non-planned reactivity increase to a subcritical condition or non-compensated reactivity for the different conditions and modes of operation.

   2.4. Equipment failure, faulty actions or omission by personnel which have led to or could lead to disturbances in the function provided for in a safety system.

   2.5. Other violations of limits or conditions for safe operation, regardless of the cause.

   2.6. Natural or other events (earthquakes, terrain collapses, floods, aviation accidents, fires, explosions, etc.) which have set or could set safety systems out of operation or could prevent the personnel from performing their duties.

   2.7. Gas-aerosol or liquid discharges within the containment confinement or out of it, violation of dose limits for the personnel or population, contamination of the environment over the prescribed limits.

   2.8. Lack of possibility for evaluation of the radiation condition of the facility or the environment condition because of faults in the technical means for radiation control or inadequate redundancy of the means of control.

   2.9. Activation of reactor protection system (except of the cases of turbine system actuation).

   2.10. Manual or automatic operation of the systems for primary circuit emergency cooling, spray system, emergency feedwater system, emergency diesel generators and other safety systems, except for testing these systems in accordance with the time schedule.

2.11. Other cases connected with nuclear safety, at the discretion of the management of the plant.

FULL REPORT

On all occurrences listed in item 2, any additional information arising from the occurrence needs to be reported and a full final report which includes all relevant information should be submitted to the regulatory body within 30 days.

Annex II
REPORTING CRITERIA FOR UERs IN SPAIN
(CSN Safety Guide 1.6)

ABNORMAL EVENTS

Abnormal events are those that require activation of the emergency plan, either on site or outside the site. They have to be reported to the regulatory body within 30 minutes of their occurring.

(1) Automatic and required actuation of emergency core cooling systems.

(2) Events evolving that may affect the safety barriers and which control is not assured at some point in time (indication of fuel failure; leakages; or abnormal transients of coolant temperature or pressure, when they do not fulfill the Technical Specifications - TS).

(3) Unexpected behaviour of the plant (surpassing safety limits; loss of redundancy in a safety system during a transient; surpassing the instantaneous release limits of the TS; relief/safety or safety valves in safety systems remaining open at a pressure lower than required; inadvertent criticality).

(4) Degradation of a safety function during operation or shutdown (loss of all a.c. electrical power, internal and external; fast and uncontrolled depressurization of the secondary system (PWR); low level or low flow in the ultimate heat sink when still is higher than the minimum designed for; loss of containment integrity or loss of redundancy in the core cooling function that according to the TS will require a shutdown).

(5) Internal event which control is not assured at some point in time and that still does not affect the safety systems but endangers the safety of the plant (fire in the plant with a duration of more than 10 minutes from its detection; flooding in areas close to safety system locations, or release of toxic or explosive substances inside the plant.

(6) Natural phenomena or external event that threatens the safety of the plant (damages in water dams; intensity of winds or raining above those expected with a return period of 10 years; uncontrolled fire near the plant; release of toxic substances when their expected concentrations at the site are higher that the authorized limits; or explosions close to or inside the site, earthquake detected by the surveillance instrumentation of the plant; crash of an airplane in the site, or abnormal air traffic).

(7) Threat to the physical security of the plant (intrusion or sabotage attempt; intentional degradation of the Physical Security Plan; access blocking; credible bomb threat).

(8) Disappearance of radioactive material.

(9) Significant loss of communications capacity with outside the plant.

OTHER EVENTS

Other events are reportable events not requiring activation of emergency plans. Depending on the reporting criteria applicable they have different reporting timing, in any case within 24 hours of the occurrence.

(1) Any event that requires performance of activities related to safety and not foreseen in the normal or emergency operating procedures. This class of events should be notified before performing the recovery operations, except when the urgency of the actions to take, advices the opposite to assure public, personnel or installation safety.

(2) Any unscheduled shutdown of the plant, and any shutdown or power reduction required by the "action" of the Technical Specifications.

An unscheduled shutdown is a disconnection from the grid required before the end of the weekend following the day of discovery of the problem causing the shutdown.

(3) Any unscheduled actuation of the reactor trip system, with one or more control rods out of the core, and any unscheduled actuation of the reactor trip systems caused by a valid signal when all control rods are inserted in the core.

(4) Any unscheduled (unexpected) release or any uncontrolled (through abnormal paths) release of radioactive material with one or more of the following characteristics:

(a) surpassing the release limits in the TS;
(b) causing an uncontrolled release to outside the plant buildings;
(c) causing a release inside the plant buildings surpassing the capacity of the treatment systems or producing dose rates higher than required in the radiological zoning of the plant.

(5) Any case where one person has received, or it is estimated that he may have received, one dose due to external irradiation and/or due to contamination that surpasses, in a single exposure, the dose limits established in the Spanish legislation.

(6) Unfulfillment of a Technical Specification (TS). When the TS refers to a Limiting Condition for Operation (LCO) there is an unfulfillment when the requirements of the LCO and the associated action are not satisfied in the time and mode established in TS 3/4.0.

(7) Unfulfillment of a Surveillance Requirement (SR). There is an unfulfillment of a SR when this is not performed in the time and mode established in TS 3/4.0, except when it has been declared the unfulfillment of the LCO requirements before the time expiration.

(8) Surpassing the value of a variable covered by an LCO that affects the safety barriers overall, i.e. reactivity control, power distribution, reactor coolant system, containment and fuel pool.

(9) Discovery of deficiencies in design, construction, installation, operation or maintenance whenever it has been determined that they may have impaired the safety function of structures or systems required to:

(a) obtain and maintain the safe shutdown of the reactor;
(b) control the release of radioactive materials;
(c) mitigate the consequences of an accident included in the Final Safety Analysis Report.

(10) Discovery of deficiencies in the actuation of the plant personnel, or in the operating procedures whenever it has been determined that they could have impaired the safety functions defined in the previous paragraph.

(11) Any automatic or manual actuation of safety systems, except when part of maintenance testing or when required by the test and mode changes requirements of the TS.

(12) Any event or internal condition of the plant, that in the judgement of the licensee, causes a potential impact on the safety of the plant or decreases the capacity of personnel to operate the plant in a safe manner and it does not reach the level of emergency, including: common mode failures in equipment; fires; releases of toxic, explosive or radioactive substances; floods; riots; strikes and decreases in communications or operators information systems capacity.

(13) Any natural phenomena or external condition to the plant that, in the judgement of the licensee, causes a potential impact on the safety of the plant or decreases the capacity of personnel to operate the plant in a safe manner and it does not reach the level of emergency, including: fires; releases of toxic, explosive or radioactive substances; storms; floods; earthquakes and riots.

(14) Any other event not covered in the previous paragraphs that, in the judgement of the licensee, may be significant for safety or radiological protection, including: disappearance of radioactive material and events or situations that causes a significantly degraded condition of the plant, including its safety barriers, or places the plant in a situation, not previously analysed that threatens safety, or in a condition outside the design basis.

Annex III
EVENT REPORTING IN FINLAND
(YVL-GUIDE 1.5, STUK)

EVENT REPORTS

This annex deals with event reports and notifications. An event may also be such that, according to instructions, two or more different reports should be submitted on it (e.g. an event including a scram and pressure vessel damage of which a special report shall be submitted). In such a case only one report may be drawn up which is provided with the identifying mark of each type of report.

Special events

Special events are incidents, failures, observations, shortcomings and problems (later in the text jointly called events) which have special significance to the plant's nuclear safety, safety of the plant personnel or radiation safety in the plant's vicinity.

The list below gives examples of events considered special events.

Emergencies

(a) A plant or general emergency situation has been declared at the plant.

Special events related to the Technical Specifications

(b) The plant has been operated in a manner which is against the Technical Specifications.

(c) Power operation has had to be interrupted owing to a requirement in the Technical Specifications.

(d) A limit defined in the Technical Specifications, which serves to secure the integrity of fuel cladding or that of the primary circuit pressure boundary, has been exceeded.

Events relating to the actuation of safety functions

(e) The reactor emergency core cooling system or containment building isolation has been actuated. Isolation of certain process systems, which takes place after a scram, usually is not considered containment building isolation as referred to here.

(f) The automatic protection function has not been actuated even though some parameter has exceeded the protection limit defined in the Technical Specifications or the protective function has not been carried out as designed.

Damage to and failure of systems and components

(g) One of the following conditions has been discovered: an increase in reactor coolant activity indicative of the failure of several fuel rods, an exceptional leak or degradation of the primary circuit or degradation of the reactor containment building so that it no longer meets the requirements set for leak tightness and strength.

(h) A component failure, functional deficiency, erroneous process or electrical connection, incorrect instruction or some other reason has been detected which could prevent a safety function as postulated in some accident analysis or some other basis for the Technical Specifications.

(i) Recurrent faults have been detected in an important component type relating to some safety function and a decision has been made to carry out corrective measures to enhance safety.

(j) Faulty or deficient functioning of a safety or pressure relief valve in the primary or secondary circuit has been discovered.

(k) An indoors gas or liquid leak has occurred at the plant. Circumstances thus created compromise or may compromise the performance of a safety function.

Deficiencies in safety assessment

(l) The reactor multiplication factor ascertained in the steady state has deviated by more than one per cent from the value anticipated for that state or the possibility of unexpected criticality in the reactor or outside it has been ascertained.

(m) An error in some accident analysis or in some method of analysis or some other erroneous basis for the Technical Specifications has been discovered and there is reason to believe that plant operation during some events is not as safe as previously assessed.

Events related to radiation safety

(n) An uncontrolled release of radioactive material indoors has occurred at the plant resulting in an essential increase in air or surface contamination or in the radiation dose rate in the rooms concerned.

(o) Some individual has received a radiation dose possibly resulting in overexposure (see Guide YVL 7.10).

(p) Releases of radioactive material off-site have exceeded the limit for corrective measures (see Guide YVL 7.1).

External events

(q) An exceptional natural phenomenon or other external threat against the plant has brought about a situation where power restriction or some other protective measure has been considered necessary.

(r) A fire or an explosion has occurred at the plant.

(s) Loss of external electricity grid has occurred as the consequence of which the plant has been supplied AC power by own internal power units.

Other events

(t) A fuel assembly has or may have become damaged during handling or its safety has been threatened in consequence of some other abnormal event.

(u) A safety threat or an attempt to deliberately damage the plant has been noted or a significant shortcoming in physical protection has been discovered.

(v) Unsolved deficiencies have been discovered in nuclear material inventory or there is other reason to believe that nuclear material has been lost.

STUK reports safety significant events to the IAEA and OECD/NEA (the IRS systems) in the extent it deems necessary. In classifying the events STUK complies with the guide referred to in Ref. [1].

Special report

A report of special events (a special report) is submitted in two weeks from the event. Where a lengthier investigation is required however, parts of the report which contain only event description and preliminary safety assessment are first submitted. The rest are submitted when investigations relating to the event have been accomplished and the decision about possible corrective measures has been taken. The report presents the following information, where appropriate:

Summary

- operational condition and power level of the plant during the initiating event;
- discovery of the event;
- brief chronological description of the event;
- personal injuries and the radiation doses received;
- releases of radioactive material;
- root causes;
- measures to ensure safety and to avoid recurrence.

Event description

1. The operational condition of the plant during the initiating event:
   - operational state and power level of the plant;
   - status and functioning of systems and components associated with the event;
   - operational and maintenance work associated with the event in progress at the time;
   - alarms prior to the event or other deviations from normal operation.

2. Event discover

3. Chronological event progression:
   - failure or malfunction which initiated the event;
   - automatic control and protection functions;
   - actions by the operators and other personnel to ensure safety;
   - failures and malfunctions which affected event progression.

4. Consequences of the event (e.g. changes in plant operational state, personal injuries, radiation doses, releases of radioactive material off-site)

5. Diagrams of process behaviour (e.g. pressure, temperature, flow):
   - plant status prior to the event;
   - changes in parameters;
   - flow charts, electrical diagrams, logic diagrams, etc. of system associated with the event.

Safety assessment

1. An overall assessment of the event's safety significance.

2. The event's impact on ensuring the most important safety functions:
   - reactor shutdown;
   - reactor cooling;
   - removal of residual heat from the reactor and spent fuel;
   - isolation of radioactive materials from the environment.

3. Potential consequences of the event under some other operational conditions.

4. Reference to similar events which have occurred in the same nuclear power plant earlier.

Causes of the event

1. direct causes;
2. root causes.

Measures to prevent recurrence

1. Summary of the reports and investigations accomplished.

2. Event review by the safety and quality assurance organizations.

3. Structural improvements at the plant (both direct corrective measures as well as those to be implemented later).

4. Improvements in the Technical Specifications, procedures, instructions and training, etc.

Notification of a special event

In case of an emergency, STUK is alerted according to instructions contained in the utility's emergency plan. During office hours, the alarm is given by the alarm phone and outside office hours STUK's duty officer is notified. An emergency is a situation in which the plant's safety is in jeopardy of deteriorating, or deteriorates, considerably.

Safety significant transients shall be communicated promptly to STUK according to instructions issued by STUK. It is characteristic of events which require immediate notification that:

- emergency operating procedures have been devised to provide against the event;
- transient has actuated a safety system;
- there are unanticipated component failures or personnel errors associated with the transient.

Immediate notification shall also be made if breach of radiation dose limits is ascertained or suspected.

All special events are, according to STUK's instructions, promptly communicated to STUK by telephone during office hours and also in the next daily report.

Reactor scrams

Scram report

Scram reports, with the exception of scheduled scram tests at low power, are submitted to STUK not later than four weeks from the scram. In the report such information is presented, where appropriate, as, according to subsection 4.1, is required for inclusion in a special report.

Notification of a scram

STUK is notified of a scram by telephone during office hours according to STUK's instructions and also in the next daily report.

Operational transients

Report on an operational transient

Significant operational transients which have lead to a forced reduction of reactor or generator power are reported as well as other significant transients which have occurred when operating the plant or its systems. The report is submitted to STUK not later than four weeks from the event. In the report such information is presented, where appropriate, as, according to subsection 4.1, is required for inclusion in a special report.

Notification of an operational transient

An operational transient which is not classified as a special event and which does not result in a scram is notified in the next daily report.

Pressure vessel damage

Report on pressure vessel damage

A report on damage is submitted to STUK in the form of a special report, if so required under subsection 4.1.

Notifying pressure vessel damage

STUK is notified by telex or telefax if damage to a pressure vessel or its auxiliaries occurs during operation. Notification is required in the following cases:

- pressure vessel is damaged or otherwise ascertained as deviating from the construction plan (e.g. leaks through the construction material or joints, collapse of supports, non-conformance resulting in repair by welding, events which have compromised the safe use of the structure);
- pressure vessel has been used in an incorrect manner (e.g. non-compliance to operational parameters);
- pressure vessel safety valve has not operated in the manner designed.

The following information shall be included in the notification:

- damaged item and method of detecting the damage;
- description of the damage;
- preliminary plans for measures (method of repairing the damage, additional inspections, schedule for measures, etc.).

REFERENCE

[1] Systems for Reporting Unusual Events in Nuclear Power Plants, IAEA Safety Series No. 93, IAEA Safety Guides, 1989

This guide is valid as of 1.11.1989 until further notice.

This guide replaces Guide YVL 1.5 issued on 24.4.1981.

Annex IV
REPORTING CRITERIA USED IN THE RUSSIAN FEDERATION AND THE UKRAINE

The criteria for reporting operational events at nuclear power plants are stated in the document ПН АЭ Г-12-005-91 and are applicable to all power plants of the former Soviet Union.

Operational events are defined by this document as events which cause significant deviation from the operational limits and/or conditions.

The reporting criteria concern two operational event categories:

- accidents;
- incidents.

Accidents

External release of a large fraction of the radioactive material accumulated in the reactor core and resulting in excessive dose limits for beyond design basis accidents, i.e. public internal dose must not exceed 10 rem for the first year after the accident and 30 rem to thyroid gland of children resulting from inhalation at a distance of 25 km from NPP.

External release of a large fraction of the radioactive material accumulated in the reactor core and resulting in excessive dose limits for design basis accidents, i.e. dose limits at the border and beyond the controlled area must not exceed 10 rem for the whole body for the first year after the accident and 30 rem for thyroid gland of children resulting from inhalation. Implementation of protective measures covered by local emergency plans within the area of 25 km padius including evacuation.

External release of radioactive fission products in quantities leading to insignificant excess of dose limits for design basis accidents. Significant damage to a large fraction of the core caused by mechanical effects or melting and resulting in excess of the maximum permissible design limit for fuel element damage in conformity with ПЪЯ РУ AC-89. Partial implementation of countermeasures covered by emergency plans (i.e. local iodine prophylactic and/or evacuation) to lessen the likelihood of health effects.

External release of the radioactive material in quantities exceeding the values for П 01 code incidents but not for the design basis accidents. Damage to the core where the safe operation limit for fuel element damage in conformity with ПЪЯ РУ AC-89 is violated but the maximum design is not. Irradiation of the personnel resulting in an overexposure of the order of 1 Sv leading to acute health effects.

Incidents

External release of the radioactive material and external radioactive effluents above the maximum permissible values without violating the operational limits and conditions. Contamination of NPP process rooms and equipment above the levels specified by the design for normal operation. Irradiation of the personnel by doses exceeding the dose limit for the personnel.

Deviation from the NPP operational limits and conditions in any operation mode without developing into an accident (except for П 03 and П 04 code incidents).

Inoperability of the safety system channels (including redundancy ones) in any operation ode of the unit.

Inoperability of several safety system channels (except for redundancy ones) in any operation mode of the unit for a period longer than specified by technical specifications.

Reactor trip or power unit switch-off from the electric power network in any operation mode of the unit caused either by equipment failure and/or human error or by external effects of natural or artificial origin.

Damage to fuel assemblies and fuel elements during fuel handling operations without developing into an accident or П 01 and П 02 code incidents.

Damage to safety significant equipment and pipelines related to A and B groups in conformity with ПН АЭ Г-7-008-89 and to control rods and servomotors detected or caused during maintenance or equipment inspection.

Power reduction to 25% and more for a period longer than 3 hours caused either by equipment failure or human error or by external effect of natural or artificial origin, except for cases when unit is unloaded for requested maintenance.

False initiation of any of the safety system channels in a mode which is not associated with providing safety function.

Inoperability of several safety system channels in any operation mode of the unit for a period not exceeding the permitted by technical specifications (except for cases when they are taken from the stand by mode for carrying out testing or maintenance where their availability is not required).

Annex V
LEGAL FRAMEWORK, REPORTING CRITERIA AND RULES IN THE CZECH REPUBLIC

INTRODUCTION

The operational experience feedback (OSEF) is based on the law Act No. 28/1984 and Regulation No. 6/1980 of the former Czechoslovakia, which are in force. Specific reporting criteria are included in the limits and conditions (Technical Specifications) of the NPP Dukovany (within 72 hours, within 24 hours, immediately not later than 8 hours). These rules are in agreement with the IAEA Safety Series No. 93. The nuclear power plant Dukovany (EDU), Utility (Czech Energy Board - ČEZ), regulatory body (State Office for Nuclear Safety - SONS) and an independent organization the Nuclear Research Institute (NRI) in Řež are participants of this process.

The hard copy and computerized system for storage of domestic unusual events records is used by EDU, SONS and NRI. Screening of events is usually prepared on the engineering judgement.

All unusual events are analysed by Failure Committee (NPP, SONS, ČEZ and NRI). It also recommends and checks corrective actions resulting from the assessment during meetings, which are held monthly and in case of necessity as soon as possible after an incident.

Selected safety significant events are additionally inspected by the SONS as part of its activities. Corrective actions are approved by the SONS (selected equipment, technical specifications).

The SONS is responsible for dissemination and exchange of information including international systems (INES, IRS, bilateral agreements).

The analysis, summarizing and statistics of all events is done by independent organization (NRI) in quarterly and annual reports. The PSA model is also used for an assessment.

GENERAL REPORTING REQUIREMENTS AND PROCEDURES

NATIONAL SYSTEM OF UNUSUAL EVENTS REPORTING SYSTEM PROVIDES THE NECESSARY BASIS FOR OPERATING THE IAEA-IRS IN ACCORDANCE WITH RECOMMENDATIONS OF IAEA SAFETY SERIES NO. 93.

ASSIGNMENTS OF RESPONSIBILITIES

<u>ACT No. 28/1984 ON STATE SUPERVISION OF NUCLEAR SAFETY OF NUCLEAR FACILITIES</u>

§14 - THE RESPONSIBLE ORGANIZATION IS OBLIGED:

d) TO INFORM WITHOUT ANY DELAY RESPONSIBLE NUCLEAR SAFETY INSPECTORS AND REGULATORY OFFICE ITSELF ABOUT IMPORTANT FACTS, ESPECIALLY ABOUT UNUSUAL EVENTS AFFECTING THE NUCLEAR SAFETY OF THE NUCLEAR FACILITY.

§13
IN THE CASE OF A DEVIATION FROM THE LIMITS AND CONDITIONS (L&C) THE RESPONSIBLE PERSONNEL WILL TAKE IMMEDIATE ACTION FOR THE FASTEST POSSIBLE RESTORATION OF COMPLIANCE. IN CASE SUCH RESTORATION OF COMPLIANCE IS NOT POSSIBLE AND THE POSSIBLE CONSEQUENCES OF THE DEVIATION ARE SIGNIFICANT FROM THE NUCLEAR SAFETY POINT OF VIEW, A REACTOR SHUTDOWN AND CORE COOLING SHALL BE SECURED. IN ALL CASES AN ANALYSIS OF THE VIOLATION OF L&C SHALL ALWAYS BE CARRIED OUT, MEASURES SHALL BE PROPOSED FOR EXCLUDING ANY REPETITION OF SUCH VIOLATION AND A REPORT SHALL BE SUBMITTED, UNDER PRINCIPLES SET IN ADVANCE BY THE CZAEC, TO THE STATE OFFICE FOR NUCLEAR SAFETY.

REGULATION No. 6
of the Czechoslovak Atomic Energy Commission
of January 23, 1980
on Ensuring Nuclear Safety in Nuclear Power Plants during
Commissioning and Operation

Under Article 54, paragraph 3, letter c of the Act of Law on the Authority of Federal Ministries and in agreement with the Federal Ministry of Fuel and Power and other participating bodies, the Czechoslovak Atomic Energy Commission rules

PART 4
Common and Final Provisions
Common Provisions
Article 38

§2
THE STAFF OF THE STATE NUCLEAR SAFETY REGULATORY BODY OF THE CZECHOSLOVAK ATOMIC ENERGY COMMISSION IS ENTITLED TO PARTICIPATE IN THE ANALYSES OF ACCIDENT SITUATIONS SUCH AS ARE RELATED TO NUCLEAR SAFETY, E.G. THE VIOLATION OF LIMITS AND CONDITIONS, EMERGENCY SHUTDOWN, ETC.

REGULATION No. 6
of the Czechoslovak Atomic Energy Commission
of January 23, 1980
on Ensuring Nuclear Safety in Nuclear Power Plants during
Commissioning and Operation

Under Article 54, paragraph 3, letter c of the Act of Law on the Authority of Federal Ministries and in agreement with the Federal Ministry of Fuel and Power and other participating bodies, the Czechoslovak Atomic Energy Commission rules

PART 3
Operation of Nuclear Power Plant
Section 2
Principles of Operation
Article 31

§10 - THE OPERATING ORGANIZATION SHALL ENSURE THE IMMEDIATE TRANSMISSION OF INFORMATION TO THE STATE NUCLEAR SAFETY REGULATORY BODY OF THE CZECHOSLOVAK ATOMIC ENERGY COMMISSION

- THE OCCURRENCE OF MODES DANGEROUS FROM VIEW-POINT OF NUCLEAR SAFETY,

- ON REACTOR SHUTDOWN INITIATED BY PROTECTION SYSTEMS.

§11 - THE OPERATING ORGANIZATION SHALL REGULARLY CARRY OUT ANALYSES OF OPERATION AND FAILURES AND SUBMIT THEM TO THE STATE NUCLEAR SAFETY REGULATORY BODY OF THE CZECHOSLOVAK ATOMIC ENERGY COMMISSION TOGETHER WITH MEASURES TAKEN; THE INTERVALS BETWEEN REPORTS SHALL NOT EXCEED ONE MONTH.

REQUIREMENTS STIPULATED IN LIMITS AND CONDITIONS FOR SAFE OPERATION IN NPP DUKOVANY

REPORTED OCCURRENCE

Reported occurrence shall be reported in accordance with the agreed upon rules, to the regulatory (inspection) body:

(A) Within 72 hours from the moment of occurrence detection:

(1) Non-planned decrease of unit power without emergency protection actuation, i.e. outage of turbo-generator, outage of a main circulation pump (ROM-limiting power regulator), action of MEZ I, II, III, protection system mode-3, protection system mode-4 upon the drop of control rod assembly.
(2) Presence of foreign objects in the primary circuit.
(3) Radiation set-up which exceeds the intervention levels determined by the regional radiation protection inspector.
(4) Occurrence of nuclear related dangerous situations in shutdown reactor during handling of the fuel.
(5) Loss of normal and emergency lighting in the reactor hall for longer than 10 minutes.
(6) Non-tightness of the primary circuit's main components - VA, VB, VC, VD, VP, TC.
(7) Automatic actions of ESFAS and stepwise startup automatics.
(8) Non-planned actuation of steam generator safety (relief) valve and that of pressurizer.

(B) Within 24 hours from the detection of an occurrence

(1) Any fires within the N PP fenced area.

(C) Immediately (not later than within 8 hours from the moment of occurrence detection)

(1) Engagement of emergency protection 1 and 2.
(2) Pressure loss of primary coolant ($\geq$ 2t/h, or radioactivity according to limiting condition 3.4.2.4.) from the system VA, VB, VC, VD, TC, TC (10, 50), TK, TY, YP.
(3) Violations of **LIMITS AND CONDITIONS FOR NORMAL OPERATION**.
(4) Loss of natural circulation and impossibility of its restoration within 1 hour.
(5) All occurrences evaluated by a shift personnel as of level 2 and higher on the INES scale.

VIOLATION OF LIMITS AND CONDITIONS

In the case of violation of allowable parameters, requirements to equipment SERVICEABILITY, PROTECTION SYSTEMS SET-POINTS, basic conditions and actions of personnel during certain operational states and organizational measures (hereafter only LIMITS AND CONDITIONS) the following steps shall be taken:

(a) The fulfilment of LIMITS AND CONDITIONS shall be restored as soon as possible, if the restoration of the fulfilment is impossible and conceivable impacts from the nuclear safety viewpoint are serious, the reactor shall be shut down and cooled down.
(b) Each violation of LIMITS AND CONDITIONS shall be entered in the operational log book of the corresponding unit supervisor who must immediately inform the shift engineer.
(c) Immediately by phone shall be informed the head of shift operation division, deputy director for production, deputy director for nuclear safety and technology, as well as the SONS (during working hours - the inspector general, after working hours - resident inspector at home address).
(d) Within 72 hours shall be written the preliminary report on the violation of LIMITS AND CONDITIONS which includes:
- situation which preceded the LIMITS AND CONDITIONS violation;
- impact of limiting conditions violation on equipment, systems and structures;
- immediate corrective measures to prevent the repetition of such a violation.
(e) Report on non-fulfillment of LIMITS AND CONDITIONS together with the analysis approved by the "failure committee" (the SONS representative should be present at the committee meeting) shall be submitted to the SONS within 30 days following the occurrence.

REPORTING OF OPERATIONAL OCCURRENCES AND RECORDING

The rules applied for the analysis and reporting of occurrences are given by the ČEZ (Czech Power Board) Guideline on the reviewing of failures and reliability of nuclear power plants equipment.

The guideline includes frequency, the format of reporting and the legal position of the records.

(a) Operating organization's duty is to ensure the immediate passing of information to the State Office for Nuclear Safety (according to para 4.6. c) on:
- occurrences (events) dangerous from the nuclear safety standpoint;
- reactor shutdown by the protection system.

(b) Chairman of the office - inspector general, and if he is absent - his appointed deputy orders, in the case when there is danger in delay, upon arising safety related facts, the necessary measures, including the power decrease or unit shutdown.

(c) Operating organization's duty is to perform regular analyses of operation and of operational failures. The reports on these analyses together with proposed corrective measures are submitted to the State Office for Nuclear Safety as a minimum once a month.

(d) Requirements for the activities related to the quality assurance and to in-service inspections of the specified safety related equipment are included in the in-service inspection programme.

(e) Operational records which should be kept by the Operating Organization over the NPP service life include data related, first of all, to:
- the nuclear power plant operation from the viewpoint of LIMITS AND CONDITIONS observance;
- maintenance, tests, inspections and repairs of equipment and systems;
- quality assurance programmes;
- qualification of the specified equipment, NPP personnel health examination and training;
- individual dose rates, specific radioactivity of effluents and radioactive wastes, as well as radiation level within the NPP.

(f) Records of the recording instruments which serve to check LIMITS AND CONDITIONS shall be filed over the period of two cycles between refueling or during two years, with the exception of the records related to failures and accidents. These records are being attached to the materials on investigation of failures and accidents and are filed together with them during the NPP operation life.

(g) Records are liquidated on the basis of the report which shall include a concise description of the normal operation violation, and references to the reports on the inquiry on accidents.

REPORTING PROCEDURES AND CHANNELS OF COMMUNICATION

(A) Reporting from NPP to the SONS

Legal document "Limits and Conditions for Operation of NPP" set:

Type of events which must be reported:
- within 72 hours from the moment of occurrence detection;
- within 24 hours from the moment of detection.

Fires in NPP site:
- immediately (not later than within 8 hours from the moment of detection) among others:
  - violations of limits and conditions;

- loss of natural circulation and non-possibility of its restoration within 1 hour;
- all occurrences preliminary evaluated by a shift personnel as level 2 and higher of INES scale.

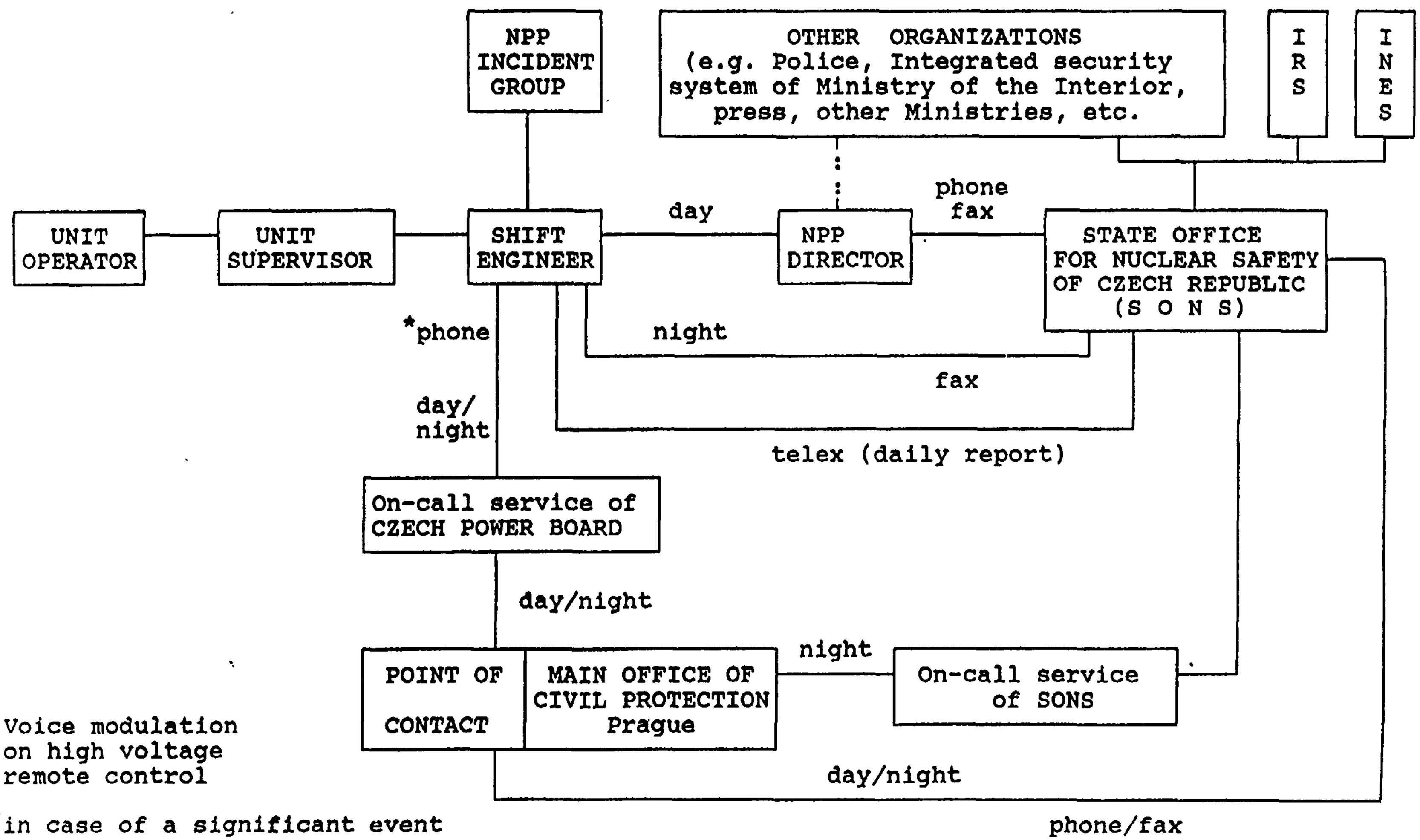
Operating organization is obliged to perform on the regular basis evaluation of all events and failures (monthly), inspector of the SONS shall be present on the meeting.

Evaluation is performed by "failure evaluation committee" of operating organization, results of the evaluation must be sent to the SONS and must include corrective measures, if there is need of any.

(B) Reporting to the international organizations and other countries

The SONS is delegated to inform the international organizations about the evaluation in accordance with IRS and INES rules and other countries in accordance with mutual agreements.

INFORMATION CHANNELS OF UNUSUAL EVENTS **



* Voice modulation on high voltage remote control

**in case of a significant event

Annex VI
FRENCH OPERATIONAL SAFETY EXPERIENCE FEEDBACK

1. GENERAL

It is the ministry in charge of the environment and the ministry in charge of industry who are responsible, within the governmental authorities, for matters relating to the technical safety of nuclear installations.

Within the ministry in charge of industry, at the disposal of the ministry in charge of the environment, there is a department named the "Nuclear Installations Safety Directorate" (DSIN).

In making its decisions, the DSIN is assisted by the Institute of Nuclear Safety and Protection (IPSN) which is its technical support and which carries the technical safety analysis. One of the important tasks of the IPSN is to collect all relevant information about technical safety problems and the measures taken in France and in other countries.

The DSIN also relies on the opinion and recommendations of the standing group of experts concerning nuclear reactors. This standing group bases its judgements on analysis reports made by IPSN and makes recommendations accompanied by proposals for technical requirements and is consulted by the head of the DSIN.

Improving safety by collecting and disseminating information from safety related events, has been in practice since the beginning of the French nuclear programme.

Importance of safety feedback has been recognized officially in a ministerial letter (CAB No. 900MZ 03/09/79) and reinforced by an order of August 10th 1984, relative to the quality of the design, construction and operation of basic nuclear facilities. Corrective actions resulting from an anomaly or incident are considered as quality relevant activities, and are defined in Articles 12 and 13 of the circular relative to enforcement of the basic nuclear facility design, construction and operating quality regulations. According to this order, the owner (operating organization) is responsible for performing safety experience feedback, and the head of DSIN is responsible for enforcing it.

2. REPORTING, STORAGE, SCREENING AND ANALYSIS

The events which occur in an operating facility are numerous. The important point is to be sure that an efficient selection keeps for examination all those that can directly provide safety lessons, while avoiding that other events be lost.

Two families can be distinguished:

- significant incidents candidate for being precursors incidents;
- events, i.e. incidents which individually may not have a significant for safety, but could reveal important trends if they occur repeatedly. Any occurrence that leads to an action statement of the Technical Specifications is considered as an event.

Reporting criteria for significant incidents have been established by the DSIN, and are defined in a letter (SIN No. 1732/82), those for significant events have been established by EdF and submitted to the DSIN. Both types of criteria are defined in two separate directives which give more details of the terms and conditions of reporting.

Reporting criteria for significant incidents

The utility has the obligation to report any incident referring to the ten following criteria:

(1) Automatic and manual reactor trip except reactor trips due to turbine trip.
(2) Safeguard system actuation:
- safety injection;
- containment isolation;
- containment spray;
- auxiliary feedwater.

(3) Loss of safety function:
- total loss of safety system;
- partial loss of a safety system that required or could have required a reactor shutdown according to technical specifications;
- safety limit violation;
- common cause failure that resulted or could have resulted in several failures in one or several safety system;

(4) Problem identified in design, fabrication or operation that results in an operating condition not previously analysed or that could exceed design basis conditions.
(5) Any release of activity exceeding regulatory limits.
(6) Staff irradiation exceeding regulatory limits.
(7) Nuclear or non-nuclear incident resulting in death or severe wound.
(8) External hazard (natural or man made) that could affect the safety of the plant.
(9) Sabotage (actual or attempted) that could affect the safety of the plant.
(10) Any incident deemed significant by the plant supervisor.

The significant incidents have to be formally notified to DSIN, within 24 hours, by a phone call or a telex; a detailed report must be sent within one month.

Reporting criteria for significant events

Criteria complementary to those of significant incidents

- Unplanned unavailability of components included in Technical Specifications with no exceeding allowed outage times (AOT).
- Unavailability of components beyond AOT for preventive maintenance.
- Service loading of the pressurizer discharge valves following a transient with scram.

Criteria related to non-compliance with periodical tests

- Radioactive releases by the normal path and not exceeding the reglementary radioactive thresholds.
- Shift of safety related systems sensors.
- Shift of automatic action setpoints (excluding protection and safeguard actions).
- Failure of alarms involved in the application of incidental or accidental procedures.
- Fortuitous discovery of anomalies.

General criteria

Any other event that may be considered as safety related.

For these events, no formal notification is required. These events are inserted in a national computerized data file according to a coding established by the utility Electricité de France, EdF.

Storage

The events such gathered, are normally stored as soon as they occur in the plants with their main characteristics in a national computerized file called the "Event File". All EdF engineering and design departments and nuclear plants are connected to this computerized network as well as the DSIN and IPSN.

Manufacturer, such as Framatome, has the accessibility to some events of the "Event File" selected according to its specific supplying.

An average of 450 significant incident reports is issued a year and about 9000 events are stored in the Event File.

The coding established by EdF enables searches for all the features deemed necessary (plant states, component and system involved, testing or maintenance in progress).

Incident analysis by EdF

■ A first level analysis is performed by people of the plant directly involved in the incident.

This analysis may lead to temporary corrective actions such as:

- modification of operating procedures;
- restoring operating practices;
- improving the training of operators;
- improving in the quality assurance programme, etc.;
- modification of maintenance or testing programmes.

This first level analysis may lead also to a fast experience feedback form sent to all other concerned plants.

Standard format of incident analysis report established by EdF is shown in Appendix 1.

■ A second level analysis is performed by the utility central services when the preliminary assessment based on information from the concerned plant indicates:

- pronounced safety significant consequences;
- generic aspects concerning the design or operating methods.

The objective assigned to this second level analysis is to validate the conclusions of the 1st level analysis when these requestion the components design or standardize operations methods.

When necessary a follow up sheet is established including the description of the event, the causes analysis, the proposed actions, the actions taken, the experts' opinion and the Directory decision.

These follow up sheets are stored in the "Event File".

These analysis are submitted to specialized committees made up of the various EdF department representatives (designers and operator) who give an advice on the 2nd level analysis and the corresponding correctives actions proposed by the utility central services in charge of the analysis.

- Final decision to implement corrective actions is assigned to the EdF Directory

Besides, results of experience feedback are taken into account by the Nuclear Safety Committee for Operation who is in charge of elaborating the Safety doctrine of operation.

Assessment of incidents occurring in plant before commissioning tests is performed by the Engineering Department of EdF which has its own experts group which draws lessons for the next generation plant design.

Screening and incident analysis by IPSN

Incident reports are examined in detail every week during a meeting gathering all the engineers in charge of a site with a representative of the head of the service in charge of the PWR's assessment.

During this meeting, incident reports received during the previous week are commented by the responsible engineer. These comments may lead to ask additional information from the licencee or when these reports concern special items like mechanics, electricity, core physics or human factor they are dispatched to the corresponding services.

There is no explicit criteria to determine which incident has to be selected for an in depth examination, this decision is taken by the representative of the head of the service according to his judgement and to the results of previous analysis.

During the same meeting, IRS reports received are commented by the engineer in charge of dispatching the IRS reports, besides incident reports which need to be reported to NEA are identified.

Finally, each engineer in charge of a site comments the most significant safety related events he has selected in the national computerized data file (Event File). Many times, this examination leads to ask further information from the licencee and sometimes to ask him to declare a significant event as a significant incident.

Abstracts of incident reports are stored in computer file (SAPIDE) with reference to all documents related to incidents: analysis report, recommendations of the standing group of experts, requests from IPSN, requirements from DSIN, additional information.

A computer program permit searches for all the features deemed necessary.

Incident reports and the referenced documents are also stored in a paper version accessible to every one.

Once an incident has been selected for detailed analysis, the responsible engineer has to gather and co-ordinate the work of the different specialists involved in the analysis.

Different meetings are generally needed in order to appreciate the progress of the analysis to gather the different assessments and to come to a mutually agreed position.

The engineer in charge of the incident analysis has the responsibility of the final report and to draw the main conclusions.

A draft of this report is sent for comment to the utility before being addressed to the DSIN.

Every week, a group of experienced engineers in reactor operation and probabilistic safety assessment is performing a qualitative screening of significant events described in the incident telex sent by EdF and received during the previous week. This screening aims at selecting those of the incidents, that may be considered as precursors of incidents leading to core melt and necessitating an analysis according to a methodology similar to the accident sequence precursor programme (ASP).

3. RECOMMENDATIONS AND DECISION PROCESS ON ACTIONS

Once corrective actions have been decided by EdF Directory, a complete modification file is established by EdF Central Services and sent to the plants.

Information of DSIN is mandatory concerning safety related equipment. The process of this information has changed over the years and has become now simpler; it is made now through information notices which identify the reference number of the modification and precise the first plant where the modification implementation is scheduled and the next ones.

These information notices give a brief summary of:

- the origin of the modifications;
- a description of the modifications;
- the incidence on safety;
- the requalification required;
- the repercussion on the reference and operating documents such as incidental and accidental procedures, periodical testing, preventive maintenance programme.

Besides, important modifications that have an impact on the regulation are examined by IPSN and submitted to the authorization of the DSIN (example: modification of the setpoint of steam generator relief valves).

Verification of the implementation of the modifications may be checked through regulatory inspections or during the regular DRIRE/IPSN/EdF meetings scheduled systematically before and after refueling outages under the responsibility of inspectors of the Regional Directorates for Industry, Research and the Environment (DRIRE).

4. DISSEMINATION AND EXCHANGE OF INFORMATION

Information of feedback activities may be found on different documents such as IPSN assessment reports and DSIN letters.

A general experienced feedback review report is issued by IPSN every two or three years. This report reviews the overall operating experience of all the French nuclear plants and is examined by the standing group of experts concerning nuclear reactors during two or three days meetings which make recommendations about corrective actions that have to be taken into account by EdF through DSIN requirements. Corrective actions issued from foreign experience feedback are examined too.

This group of experts has at the same time the opportunity to review the effectiveness of corrective actions and modifications implemented by EdF and resulting from the previous DSIN requirements.

The final safety analysis report assessment of a plant by IPSN gives also the opportunity to the group of experts to examine the effectiveness of corrective actions implemented in this plant through the IPSN assessment and DSIN requirements. Conclusions from inspections by DRIRE inspectors are reviewed too. The final safety analysis report assessment may give the opportunity to make, according to the circumstances, recommendations concerning a specific item which presents an urgency degree and a generic aspect.

Incident report analysis carried by IPSN include specific corrective actions to be recommended following the occurrence of a significant incident. This incident report generally takes into account the results of the 2nd level analysis performed by EdF.

Short term corrective actions or verifications to be implemented on all plants, not necessitating long delays or important modifications are required when an anomaly or event which is supposed to be generic is occurring in a plant. These simple and evident checks are required to be extended to all concerned plants by letters addressed by DSIN at each refueling outage programme and whenever needed.

A computerized follow up file is established for every recommendation of IPSN or the standing group of experts including respective requirements from DSIN and corresponding answers from EdF.

There is non-special criteria for screening external information.

Emphasis is put on events occurring in similar plants:

- first of the kind incidents;
- events with implication for French similar plants;
- checking whether an incident having occurred in a unit is likely to happen in another type of unit.

Foreign information used in the feedback process relies mainly on IRS reports, generic letters and Information Notices from NRC, NUREG reports.

INPO and WANO reports are used by the licencee (EdF).

APPENDIX 1. FORMAT CONTENT FOR SIGNIFICANT INCIDENT REPORT

A. GENERAL INFORMATION

1. GENERALITIES
2. INCIDENT TITLE
3. ANALYSIS STATE
4. REACTOR AND COMPONENTS STATE
5. IMMEDIATE OR POTENTIAL CONSEQUENCES
   - 5.1. On plant availability
   - 5.2. On component
   - 5.3. On performances
   - 5.4. On safety
   - 5.5. On environment
   - 5.6. On radioprotection
6. COMPLEMENTARY INFORMATION
7. COMPLEMENTARY DOCUMENTS ESTABLISHED
8. ANNEX DOCUMENTS

B. OPERATION PART

1. PRECISE DESCRIPTION OF THE INCIDENT
   - 1.1. Incident chronology
   - 1.2. Final state
   - 1.3. Restart chronology
2. COMMENTS
   - 2.1. Root causes
   - 2.2. Comments on observed anomalies
3. ACTIONS TAKEN OR CONSIDERED
   - 3.1. On human level
   - 3.2. On technical level

C. MAINTENANCE PART

1. PRECISE DESCRIPTION OF THE FAILURE OR ANOMALY
2. COMPONENT INVOLVED
   - 2.1. Component designation
   - 2.2. Manufacturer
   - 2.3. Type of component
   - 2.4. Main characteristics
3. STATISTICAL INFORMATION
   - 3.1. Hours in operation
   - 3.2. Number of starts
   - 3.3. Unavailability duration
4. FAILURE OR ANOMALY CAUSES ANALYSIS
5. WORK PERFORMED
6. ANOMALY OR FAILURE ASSESSMENT
7. ACTIONS TAKEN
   - 7.1. Temporary actions
   - 7.2. Final actions
8. COMMENTS

Annex VII
GERMAN PRACTICE IN OPERATIONAL SAFETY EXPERIENCE FEEDBACK

1. GERMAN INCIDENT REPORTING SYSTEM

Based on the "Ordinance on the Nuclear Safety Representative and Reporting of Incidents and other Events (AtMSV)"

Implemented by:

- reporting criteria (app. of AtMSV);
- standardized reporting form;
- database for the incidents reported.

The reporting criteria contain:

- 4 reporting categories with reporting deadlines;
- definition of the incidents to be reported.

The reporting category of an incident mainly depends on the urgency for the information of the licensing authority to take corrective measures in time, if necessary.

2. HISTORY OF DEVELOPMENT

1973: Decision of the federal and state authorities for a standardized reporting procedure with standardized reporting criteria.

1975: Publication of the reporting criteria for incidents in NPPs.

1985: Publication of revised reporting criteria.

1989: Second revision of the reporting criteria, in particular with regard to the AtMSV:
- addendum with additional guidance;
- standardization of the radiological part of the reporting criteria and the reporting form for NPPs and other nuclear installations.

1991: Publication of the revised reporting criteria for incidents in NPPs to be reported to the licensing authorities.

1991: Utilities agree to supplement INES-classification of the incidents reported to the licensing authorities.

1992: Publication of AtMSV with reporting criteria as appendix.

3. REPORTING CRITERIA

Category S
- Incidents to be reported immediately to carry out inspections or take corrective actions at once, e.g. incidents showing safety significant deficiencies.

- Reporting deadline/period:
  - immediately by phone or fax/telex;
  - within 5 workdays by reporting form.

Category E
- Incidents to be reported during a period of 24h to carry out inspections or take corrective actions without delay, e.g. potentially safety significant incidents.
- Reporting deadline/period:
  - within 24h by phone or fax/telex;
  - within 5 workdays by reporting form.

Category N
- Incidents to be reported to identify possible weak points at the earliest possible stage; incidents of low safety significance.
- Reporting deadline/period:
  - within 5 workdays by reporting form.

Category V
- Incidents before commissioning with potential significance for plant operation.
- Reporting deadline/period:
  - within 10 workdays by reporting form.

4. REPORTING FORM

The reporting form consists of the following 4 parts:

- general information concerning the plant, equipment and the incident itself;
- details concerning radiological consequences of the incident;
- texts for descriptions (e.g. event description, root causes, actions taken, etc.);
- code catalogue for indicating different aspects with regard to the incident.

4.1. EXPERIENCES CONCERNING THE REPORTING FORM

- The reporting form is accepted by the utilities.
- The "description" could be more detailed for many cases.
- Often, the "root case" description refers to further investigations. The results are not always presented in the final report.
- Often, the "code catalogue" is not filled in completely.

5. ASSESSMENT AND FEEDBACK

Assessment by 2 levels:

First level: State level
For the particular incidents in German NPPs:
- assessment by the NPP;
- check by the state licensing authority and their consultant (TÜV).

Second level: Federal level
On a generic basis for all incidents reported (national and international):
- assessment by BMU/BfS and their consultant GRS.

5.1. OVERVIEW OF THE STEPS OF ASSESSMENT AND FEEDBACK (FEDERAL LEVEL)

1. Screening process
   - first assessment of German incidents reported;
   - systematic in-depth analysis of German incidents;
   - analysis of the applicability of incidents reported by IRS on German plant situations.

2. Special investigations
   - detailed investigation of particular incidents;
   - generic investigations concerning technical questions identified by the screening process.

3. Actions due to the results of investigations
   - "information notices" with recommendations;
   - discussion in the German RSK;
   - reports to IRS.

4. Dissemination and exchange of information
   - periodic reports on national and international incidents (German reporting system/IRS);
   - information notices;
   - reports of the results of special investigations;
   - reports to IRS.

5.2. FIRST ASSESSMENT OF GERMAN INCIDENTS REPORTED

Goals

- To ensure standardized application of reporting procedures by all plants.
- First examination of the incidents with regard to their safety significance.

Tasks

- Check of the reporting procedure including classification with regard to the German reporting criteria.
- Check and completion of the reporting form codes.
- First assessment of the safety significance (report monthly for the BMU).

6. SYSTEMATIC IN-DEPTH ANALYSIS

Goals

- Identification of incidents or safety related questions:
  - with significance for other NPP;
  - for which further investigations are necessary.

Tasks

- Screening by engineering judgement by 3 steps:
  - by responsible experts using the incident form, the incident database and additional detailed information given by utility and TÜV;
  - by weekly discussion of the reported incidents in the department responsible using supplementing information;
  - by feedback from other departments involved.
- Screening by supplementary analyse methods:
  - precursor identification;
  - trend analysis using the database;
  - identification and classification of incidents with human factors.
- Check of INES classification.

Main criteria for the assessment

- Violation of deterministic criteria with regard to plant design and operation.
- Violation of the redundancy principle concerning plant equipment and administration.
- Violation of the diversity principle concerning plant equipment and administration.
- Multiple failure of components.
- Unexpected behaviour of systems or components.
- Effects due to missing (or insufficient consideration in design or operating instructions.
- Increase in failure frequencies for components.

7. SPECIAL INVESTIGATIONS

The investigations contain:

- Detailed investigations of particular incidents.
- Generic investigations concerning technical questions identified by the screening process on an engineering and scientific basis.

Methods

- In-depth investigation, if necessary, supported by e.g.:
  - PSA methods;
  - simulation models;
  - codes (e.g. for the thermo-hydraulic behaviour).
- Transients are checked under assistance of an "Analysesimulator" (analysis simulation programme) for special cases - planed -.

8. DISSEMINATION AND EXCHANGE OF INFORMATION

- Quarterly and annual reports on all incidents in German reported.
- Information notices (Weiterleitungsnachrichten) concerning incidents with significance for other German NPPs.
- Reports of the results of special investigations.
- Comprehensive reports on the systematic in-depth analyses of reported incidents for the German RSK.

- Ad-hoc expertise on special issues for the BMU.
- Periodic German reports on incidents in foreign NPPs reported by IRS with comments concerning the applicability on German NPPs.
- Annual report on selected foreign incidents with significance for German NPPs.
- Reports including assessment on selected German incidents to IRS.

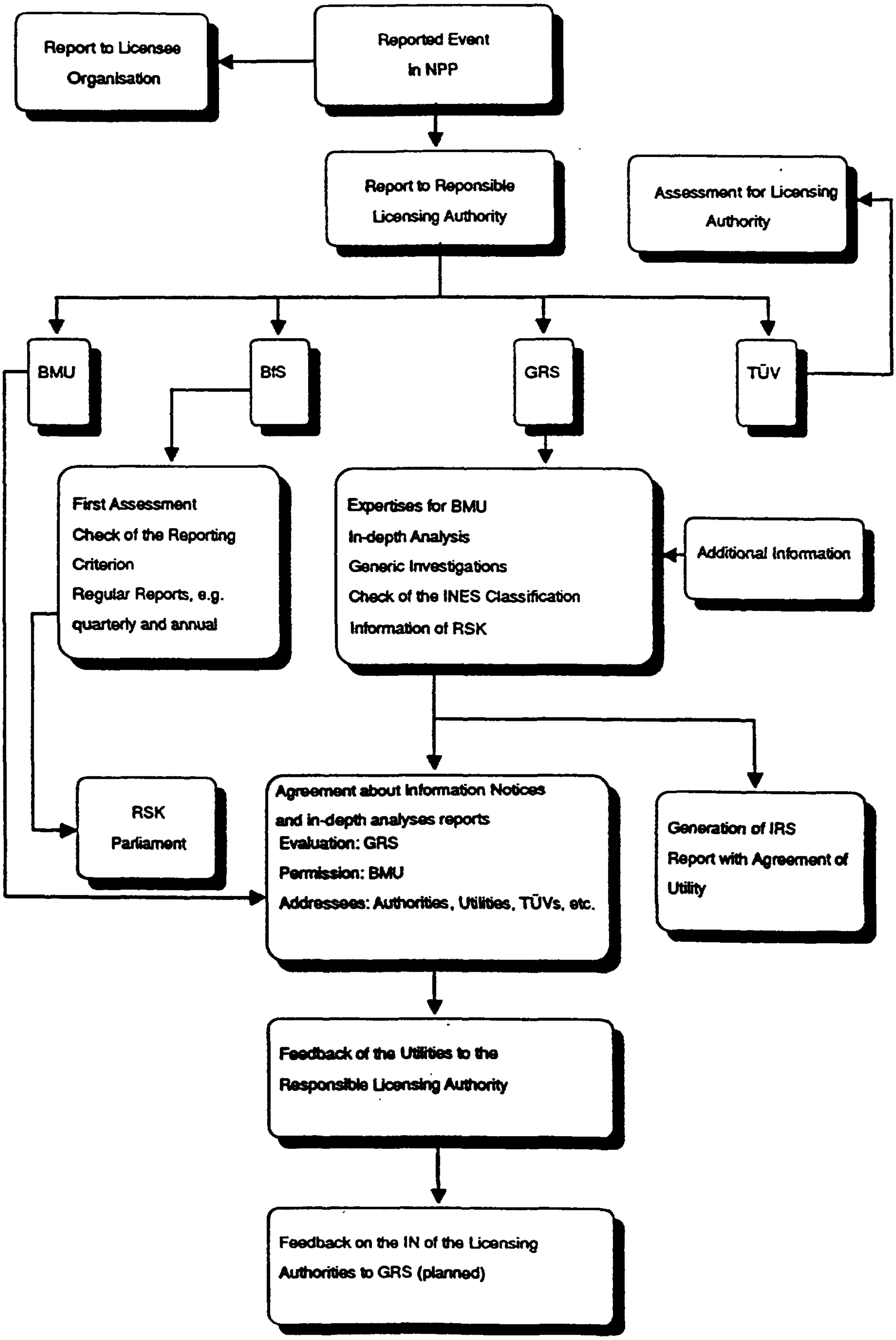
9. INFORMATION NOTICE

Structure in general

- Description of the incident and the problem involved.
- Description of root causes and results of investigations.
- Evaluation of safety significance.
- Actions taken in the plant involved.
- Recommended actions for other German NPPs.

Addressees

- NPPs
- Utilities
- Federal Ministry (BMU)
- State licensing authorities
- RSK
- BfS
- TÜV
- Manufacturer
- Special organizations (in some cases)



German practice in operational safety experience feedback

Annex VIII

INCIDENT REPORTING AND FEEDBACK OF OPERATING EXPERIENCE OF NUCLEAR POWER PLANTS IN JAPAN

1. INTRODUCTION

Ministry of International Trade and Industry (MITI), the regulatory body for the operation of commercial nuclear power plants in Japan, has been paying close attention to incidents and failures occurred at domestic plants that are required to be reported to MITI under the provisions of laws and notifications, as well as those occurred at overseas plants, the details of which are obtained through various channels including OECD/NEA and IAEA Incident Reporting System.

As MITI understands that the best way to promote nuclear power generation, in the sense of public acceptance, is to keep safe operation of nuclear power plants as much as possible, or to keep incidents or failures as less as possible, safety administration of the domestic nuclear power plants is most essential.

As of the end of March 1993, 42 domestic nuclear power plants are in operation and incident and failure reporting rate at the end of Fiscal Year 1992 was 0.5/reactor-year including minor incidents.

Such a low rate is attributed to the voluntary effort of individual electrical utility and the strict administration of the regulatory body.

2. PROCEDURES FOR EVENT REPORTING

2.1. PROCEDURES FOR SUBMITTAL AND REVIEW

Basically, reports relating to EUIL are submitted to the Electric Power Technology Division, the Agency of Natural Resources and Energy (ANRE), while Nuclear Power Operating Administration Office (NOA) of the same Agency receives report relating to LRNR.

For the incidents and failures occurred in the nuclear power plant, NOA generally administrates and respective organizations within MITI are to review relating items of their own scope of responsibility and give comments to NOA (for example, electrical facility will be reviewed by Electric Power Technology Division and turbine facility by Electric Power Generation Division) and NOA will issue overall instruction or comments to the utility.

If the event is of more important nature, MITI Technical Advisory Committee on Nuclear Power Generation is asked their opinion which will be referred on the necessary commitment of MITI.

The first press release is made soon after the survey based on the quick report mainly on the description of the event and the cause of the event if available.

After the identification of the cause and the counter measures are established, MITI will make the second press release and clarify the event on the cause, the counter measures to prevent recurrence, etc.

2.2. SAFETY ADMINISTRATION OF NUCLEAR POWER PLANT OPERATION - RESPONSIBILITIES AND IMPLEMENTATION OF LESSONS LEARNED

(1) Activities of regulatory side

Regulatory side (MITI) gives the general principle to the utilities regarding the procedures of reporting the event upon receiving the report from the utility company.

- (i) MITI usually holds hearing of the content of the event in detail from the utility.
- (ii) If necessary, MITI dispatches the staff or survey team to the site and investigate the event in more detail.
- (iii) Request the utility to submit more specific data such as analysis data, test data in addition to their detailed report.
- (iv) Based on the investigation of all the facts obtained in the above, MITI makes identification of the cause and appropriate counter measures referring proposals submitted by the utility.
- (v) During the above processes, MITI asks its Technical Advisory Committee to survey the event and respond its opinion.
- (vi) Any further action is necessary, MITI requires the utility for such action.

(2) Activities of utility side

Utility is considered to have a responsibility of so-called self-imposed safety administration.

When unusual event occurred in the nuclear power plant, the utility company would make actions as follows:

- (i) The utility company reports the event to the regulatory body according to the criteria as mentioned in 2 above.
- (ii) The utility company organizes ad-hoc task force within the company to investigate the cause and counter measures of the event, or if necessary, to nominate responsible organization to manage the case.
- (iii) The utility company reports the identified cause(s) and proposes necessary counter measures based on its investigation.

If necessary, the utility may perform shop tests, Lab tests or large scale mockup tests to simulate the event.

Details of countermeasures, for example, need further discussion with the regulatory side and regulatory side agrees to implement such counter measures by the utility.

For all the above activities both by the regulatory and the utility, detailed survey of similar events is performed and if applicable, past experiences or lessons learnt from domestic and overseas event records should be fully referred and implemented.

2.3. MANAGEMENT OF INCIDENT REPORTS

The Safety Information Research Center (NUSIRC) of the Nuclear Power Engineering Corporation (NUPEC) has been established in 1984 to manage large amount of information accumulated in the regulatory side.

Database of safety administration (Nuclear Power Information System) has been first established in 1985 and addition to files and accumulation of data made it a large database.

All the domestic incident reports including both quick and detailed reports are input into the incident/failure file composed of magnetic tape and optical disc systems and available for retrieval of the data by some 1,700 key words.

For every occurrence of incident/failure of the domestic nuclear power plant, similar event data is provided upon request through the retrieval of the data file.

Overseas incident information such as IRS (OECD/NEA and IAEA) and LER (USNRC) is also input to the separate file and retrieved for supplemental information to the domestic files.

As one of the functions of NUSIRC, statistical analysis of the event data is widely performed and results of those analyses are also used for various fields of activities such as national and international conferences or meetings.

3. NATIONAL PRACTICE FOR THE BACKFITTING OF LESSONS LEARNED FROM THE NUCLEAR POWER PLANTS OPERATING EXPERIENCES

It is considered that, in Japan, backfitting of the operating experience is the sole responsibility of electric utilities as the securing of safety by self-imposed safety system is the basic philosophy.

However, backfitting of operating experience is practically performed on a case by case basis, actions for such backfitting has some distinction in the government and private sectors.

(1) Backfitting of lessons learned in the regulatory level

(a) For the minor incidents defined by the Minister's Notification, are annually released as a judgement of the individual electric utility.
(b) Practically, MITI requires to confirm integrity of the similar type of plant owned and operated by the same counter measures. In case of extremely important, MITI may order to stop the operation of the plants of similar type to check the integrity of the equipment of the same type.
(c) Main portion of the actual execution of the counter measures is based on direct guidance of the government.
(d) For the investigation of cause incident, etc., government will give guidance based on laws and regulations to the utility upon utility's report. In case of any difficulty encountered for the investigation, government may commit temporary inspection based on laws and regulations. Confirmation of the completion of execution of counter measures for the items required by the government, will be performed by PSI is such countermeasure relates to the licensing for construction plan.

For the items other than above, the senior specialist for nuclear power operation stationed at the nuclear power plant site may confirm at the site.

Soft related items, such as security administration organization, is to be confirmed through site inspection by overall security administration inspection which is periodically performed.

(e) The system of senior specialist for nuclear power operation was instituted in April of 1980. Under this system, specialists are dispatched to nuclear power stations and permanently stationed there to monitor and consult plant regarding operating administration.

Their primary duties are to ensure compliance with safety regulations, perform inspections of the reactor installations, be present at the surveillance tests out carried voluntarily by electric utilities, and to report to MITI headquarters and serve as liaison, etc., in the event of incidents or failures and feedback of such experiences to the plant.

(f) MITI Technical Advisory Committee has been reorganized this year and Comprehensive Preventive Maintenance Committee was newly established.

This new Committee will handle the matter relating to:

(i) severe accident counter measures;
(ii) periodic safety review; and
(iii) ageing counter measures and is expected that backfitting of lessons learned will also be discussed in this Committee.

(2) Feedback of lessons learned in the utility level

Utilities of Japan has their information exchange networks in the various levels of management and have periodical meetings for their own references.

As the basic philosophy for securing safety of the nuclear power plants is self-imposed safety systems, utilities voluntarily performs feedback of the operating experiences, usually before MITI's requirement, such as inspection of integrity of the equipment similar to that which caused any incident or failure.

Periodical inspections are also made as the part of preventive maintenance which is also corresponding to the operating experiences.

(3) Examples of backfitting or lessons learned from operating experiences

As mentioned above, backfitting of lessons learned from incidents/failures of the nuclear power plants to the operating plants are made by the electric utilities themselves.

Some of the operating experiences and their status of backfitting are shown in Attachment 1. Reporting criteria are provided in Attachment 2 and procedures for event reporting in Attachment 3.

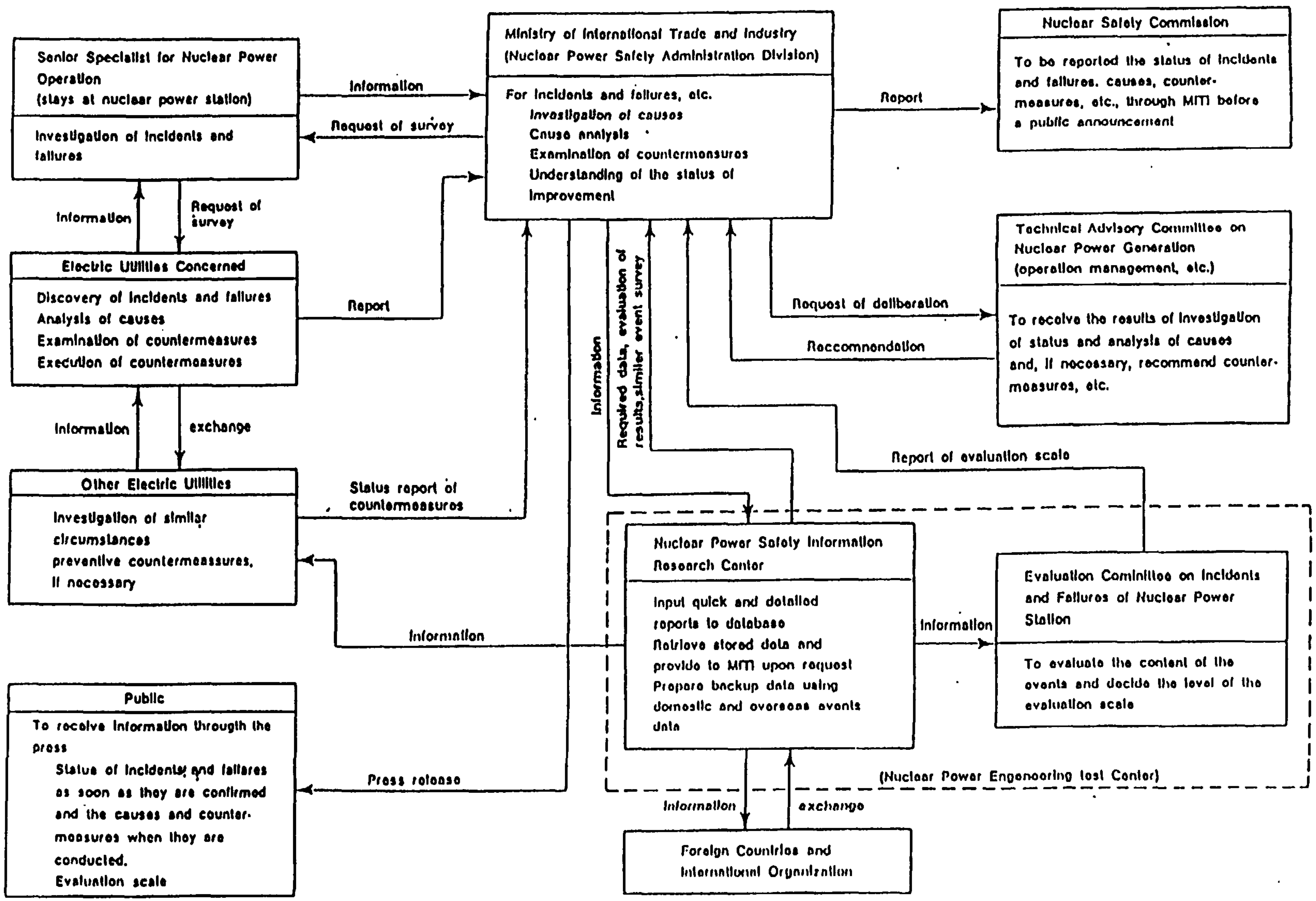


Figure 1 Flow diagram of the reporting and processing system for incidents and failures

Backfitting of Operating Experiences (BWR reactors)

Related IRS Report			Backfitting in Japan	
Report Number	Title	Abstract	Contents of Backfilling	Status of Backfilling
857 00	Cracks in shroud support access hole cover welds (Peach Bottom-3, 1988 1 21)	By ultrasonic testing, intergranular stress corrosion cracking were found in the weld heat affected zone around the entire circumference of the access hole covers of the shroud support within the reactor vessel	The access hole cover has been changed from welded type to bolt fitted type	Some BWR units completed the backfitting and the other units will be backfitted during plant shutdown such as periodical inspection
877 00	Loss of recirculation pumps accompanied by severe power oscillation (Lasalle 2, 1984 10 19)	During 88% power operation, two recirculation pumps tripped causing power oscillations The reactor was tripped due to "Neutron Flux High' signal	(1) The response procedure for the recirculation pump tripping is added to the operating procedures (2) When the APRM indicated oscillations (entering unstable region) the reactor is to be manually shut down immediately (3) A curriculum of the response procedure of the neutron flux oscillation is introduced in the training for operators (4) SRI is introduced for insuring further safety margin	Items (1) ~ (3) were backfitted at all BWR units For item (4) some BWR units were introduced and the other units will be introduced during periodical inspection
952 00	Reactor manual shutdown due to increase in PCY floor drain (Fukushima Daiichi-3, 1988 7 27)	During 99% power operation a slight increase of the PCY floor drain was detected, the reactor was manually shutdown As a result of inspection, a crack was found at a welded joint elbow of rent piping which was connected with primary loop recirculation (PLR) pump outlet valve	From the viewpoint of prevention of fatigue cracking, the socket weld joint design has been changed to the bull weld type	The socket welding joints of the primary system in the PCY are being replaced with the bull welding joints one after another units
959 02	Damage of the recirculation pump (Fukusha Daini-3, 1989 1 9)	During 94% power operation, the vibration of the recirculation pump (B) increased, the reactor was manually shutdown As a result of the disassembling inspection of the pump, damage of the hydrostatic bearing ring and impeller was found The cause of the event was lack of the welding penetration on the hydrostatic bearing ring	The hydrostatic bearing ring design has been changed from the weld type to the one block centrifuged casting or full penetration welding type	Backfittted for all concerned BWR units

Backfitting of Operating Experiences (PWR reactors)

Related IRS Report			Backfitting in Japan	
Report Number	Title	Abstract	Contents of Backfilling	Status of Backfilling
598 00	Rupture of a feedwater pipe (Mulhein Karlich, 1985 9 27)	While the feedwater pumps were operating in minimum flow mode as a pre operational test, a rupture of one feedwater line occurred because of pressure rising by heating up the water enclosed between the closed check and gate valves	As only Mihama 1 has a possibility of occurrence of the similar event, countermeasures such as addition of a balance holes have been taken	Backfitted for Mihama 1
700 00	Feedwater line break due to severe pipe wall thinning causes fatalities (Surry-2, 1986 12 9)	During full power operation, the 18-inch suction line to the main feedwater pump A failed catastrophically The cause of the event was thinning of the pipe due to corrosion/erosion	In Japan, the measurement of wall thickness of the secondary piping has already been performing up to that time The portions which tendency of the wall thinning was found are replaced with ones made from corrosion/erosion resistant material	By way of precaution, measurements of wall thickness of the feedwater and condensate water system piping are intentionally performing in order to secure detailed tendency of the wall thinning relate to configuration of the piping, fluid condition, etc
924 00	Broken retaining block studs on Anchor Darling check valves (Diablo Canyon-2, 1988 10)	Failures of retaining block studs Anchor Darling check valves were found	Tough no damage was discovered in the previous periodical inspections, to prevent the occurrence of the similar event, the check valves using similar material to the Anchor Darling check valves, have been replaced with corrosion resistant material	Backfitted for all PWR units
1027 00	Primary-to-secondary steam generator tube leak due to defective Westinghouse-designed mechanical plugs (North Anna-1, 1989 2 25)	A steam generator tube damaged due to broken piece of Westinghouse designed mechanical plugs	All mechanical plugs at hot leg side and mechanical plugs using heats NX-4523 NX-5222 at cold leg side have been replaced with plugs made from TT690	Backfitted for all PWR units
1172 00	Automatic shutdown of Mihama-2 (1991 2 14)	During rated power operation, a steam generator tube break occurred The reactor was automatically shutdown and the emergency core cooling system actuated	The following countermeasures are taken in order to prevent reoccurrence (1) Intensification of the quality assurance efforts, (2) Improvement of integrity of the steam generator tube, (3) Improvement of maintenance and management procedure, etc , (4) Improvement of operations manual, (5) Improvement of monitoring systems and instrumentation and control systems	Being conducted

INCIDENT REPORT FORM

Date ____________

Name of Utility ____________

Title							
Date of Incident		Weather					
Location of Incident							
Affected Electric Article		Voltage					
Event Description							
Cause of Event							
Protection System and Suitability of Its Operation							
Outline of Affected Electric Article							
Other Consequence							
Interrupted Electric Power To Be Supplied and Its Time Interval		Interrupted Electric Power To Be Generated and Its Time Interval					
Date of Restoration		Cost for Restoration					
Countermeasures to Prevent Recurrence							
Personal Injury	Organization	Name	Sex	Age	Experienced Years of Work	Suffered Description	
Outline of Privately Owned Electric Article	Type of Industry		Generated Power		Generated Power Voltage		
			Received Power		Received Power Voltage		
Qualification and Nomination of Chief Engineer							
Name and Organization of Chief Engineer							

ATTACHMENT 2

Criteria for the reporting of events

There are two categories involved in the criteria for reporting incidents and failures occurred at domestic nuclear power plants.

Incidents and failures reportable to MITI under the provisions of laws

Certain incidents and failures must be reported to MITI by the concerned electric utility company under the provisions of the Law for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (LRNR) and the Electric Utility Industry Law (EUIL).

The events which must be reported according to LRNR are:

(1) Reactor events which necessitate a reactor shut down (the decision to shut down a reactor being based on the restriction items of the safety regulations which are approved in accordance with LRNR).
(2) Release of gaseous and liquid radioactive materials which exceeds the permitted level of radiation.
(3) Leakage of substances contaminated by nuclear fuel material or nuclear source material outside of the radiation controlled area.
(4) Execution of preventive measures for leakage of substances contaminated by nuclear fuel material or nuclear source material inside the radiation controlled area.
(5) Radiation exposure of the employee which has exceeded or has the possibility of exceeding the allowable level of radiation.
(6) Occurrence of personnel hazard or the possibility of such a hazard in a nuclear facility.

The following incidents or failures at the nuclear power plant should be reported to MITI according to EUIL:

(1) Death of personnel caused by electric shock.
(2) Fire in an electrical system.
(3) Death of personnel or destruction of structures caused by the failure or destruction of articles.
(4) A radiation accident.
(5) Destruction of principal electrical equipment.
(6) Incidents or failures disturbing electric generation.
(7) Natural hazards.
(8) Incidents occurred during construction or installation of electrical equipment.

All of the items reportable in accordance with EUIL, except a radiation accident, is mainly concerned with electrical systems and electricity generation. These are some items which must be reported not only for nuclear power generation, but also for hydro and thermal power generation and electricity transmission and distribution.

Requirements for the time of submittal of the report is:

- for LRNR: Formal report within 10 days after occurrence of the event.

- for EUIL: Quick report within 48 hours and Detailed report within 30 days after occurrence of the event.

In both cases, however, regulatory body requires immediate report of the event by any possible communication means.

Reportable reactor events under EUIL are quite similar to those required under LRNR. In case the report is submitted under the provisions of two laws, exactly the same report is submitted to the authority.

ATTACHMENT 3

Procedures for event reporting

1. Format of the reports

An incident or failure occurred in a domestic nuclear power plant is required to be reported to MITI according to the reporting criteria.

There is no specific reporting format required by LRNR. However, the format of the report under EUIL is defined by law. So, the similar format is used for both cases.

The content of the report under LRNR should include:

(1) Date of occurrence of the incident
(2) Name of the nuclear power plant
(3) Status
(4) Causes
(5) Effects
(6) Counter measures taken
(7) Date of restart of the plant

The EUIL requires a quick report to be made within 48 hours and a detailed report within 30 days. Items to be reported according to EUIL are:

(1) Date of occurrence of the incident
(2) Name of the nuclear plant facility
(3) Status
(4) Causes
(5) Types of protection system and status of their operation
(6) Brief explanation of ailed electrical equipment
(7) Effect of the incidents
(8) Power generation loss and period of unavailability owing to the incident
(9) Counter measures taken
(10) Date of recovery

The detailed report according to EUIL includes following items in clear and precise manner:

(1) Detailed description of the event.
(2) Detailed report on the investigation of the causes (analysis reports and laboratory test reports such as reproduction of the event or verification, etc. shall be attached).
(3) Countermeasures taken to prevent recurrence.
(4) Other necessary information.

2. Procedures for submittal and review

Basically, reports relating to EUIL are submitted to the Electric Power Technology Division, the Agency of Natural Resources and Energy (ANRE), while Nuclear Power Operating Administration Office (NOA) of the same Agency receives report relating to LRNR.

For the incidents and failures occurred in the nuclear power plant, NOA generally administrates and respective organizations within MITI are to review relating items of their own scope of responsibility and give comments to NOA (for example, electrical facility will be reviewed by Electric Power Technology Division and turbine facility by Electric Power Generation Division) and NOA will issue overall instruction or comments to the utility.

If the event is of more important nature, MITI Technical Advisory Committee on Nuclear Power Generation is asked their opinion which will be referred to the necessary commitment of MITI.

The first press release is made soon after the survey based on the quick report mainly on the description of the event and the cause of the event if available.

After the identification of the cause and the countermeasures are established, MITI will make the second press release and clarify the event on the cause, the counter measures to prevent recurrence, etc.

Annex IX
REVIEW OF OPERATING SAFETY EXPERIENCE FEEDBACK PROCESS IN THE USA

ABBREVIATIONS

AEOD	USNRCs Office for Analysis and Evaluation of Operational Data
AIT	Augmented Inspection Team
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
CFR	Code of Federal Regulations
EAB	Events Assessment Branch
GL	Generic Letter
HPE	Human Performance Evaluation
IAEA	International Atomic Energy Agency
IEEE	Institute of Electrical and Electronic Engineers
IIT	Incident Investigation Team
IN	Information Notice
INPO	Institute of Nuclear Power Operations
IRB	Incident Response Branch
IRS	Incident Reporting System
LER	Licensee Event Report
NAS	Non-reactor Assessment Staff
NMSS	USNRCs Office of Nuclear Material Safety and Safeguards
NRR	USNRCs Office of Nuclear Reactor Regulation
NUDOCS	Nuclear Documents System
OECD/NEA	Organisation for Economic Co-operation and Development/Nuclear Energy Agency
OPA	USNRCs Office of Public Affairs
PN	Preliminary Notification
PRAB	Probabilistic Risk Assessment Branch
ROAB	Reactor Operations Analysis Branch
SCSS	Sequence Coding and Search System
USNRC	United States Nuclear Regulatory Commission
VIB	Vendor Inspection Branch

1. INTRODUCTION

The United States Nuclear Regulatory Commission (USNRC) has the responsibility of regulating the civilian use of nuclear energy in the United States. As such, the USNRC regulates all facets of the nuclear fuel cycle, the production of commercial nuclear power, and the operation of research reactors; biological, medical and non-reactor applications, and the storage and disposal of all non-military nuclear materials, byproducts, and wastes.

One of the most important lessons that was learned from the Three Mile Island accident was the absolute necessity for conducting systematic analyses and evaluations of operating experience.

In 1979, the USNRC formulated an action plan to assure that in the future, operational experience would not be overlooked, and would be carefully analysed and used to improve nuclear safety and prevent future accidents.

The USNRC's action plan resulted in the creation of several organizations to meet the aforementioned objectives. This paper provides an overview of those different organizations and it explains how they work independently, and yet interact to assure that operational experience from U.S. and foreign plants are systematically reviewed and meticulously analysed to assure that important operating data is not overlooked, and to assure that lessons are learned from operational events (and/or conditions), and the lessons learned which are important to safety are disseminated to the worldwide nuclear community.

For the purpose of this paper which focusses upon nuclear power plants, it will suffice to note that the USNRC does have programmes for the review of operational data in the areas of biological and medical applications of radioactive materials, and in the area of fuel production, storage, and the disposal of nuclear material byproducts and wastes. The Office of Nuclear Material Safety & Safeguards (NMSS) is in charge of reviewing and analysing events in these areas. In addition, the USNRC's Office for Analysis and Evaluation of Operational Data (AEOD) Non-reactor Assessment Staff (NAS) reviews and analyses events involving nuclear medicine and other nuclear materials.

Figure 1.1 is an organizational chart indicating which USNRC offices are actively involved with the review, assessment, and feedback of operational data (for commercial reactors). Figure 1.2 illustrates ways in which operational events or conditions may be addressed (the actual path for each event is dependent upon its perceived significance and the applicability or similarity of conditions at other plants).

Figure 1.3 is a flow diagram illustrating USNRC's rapid response to operational events.

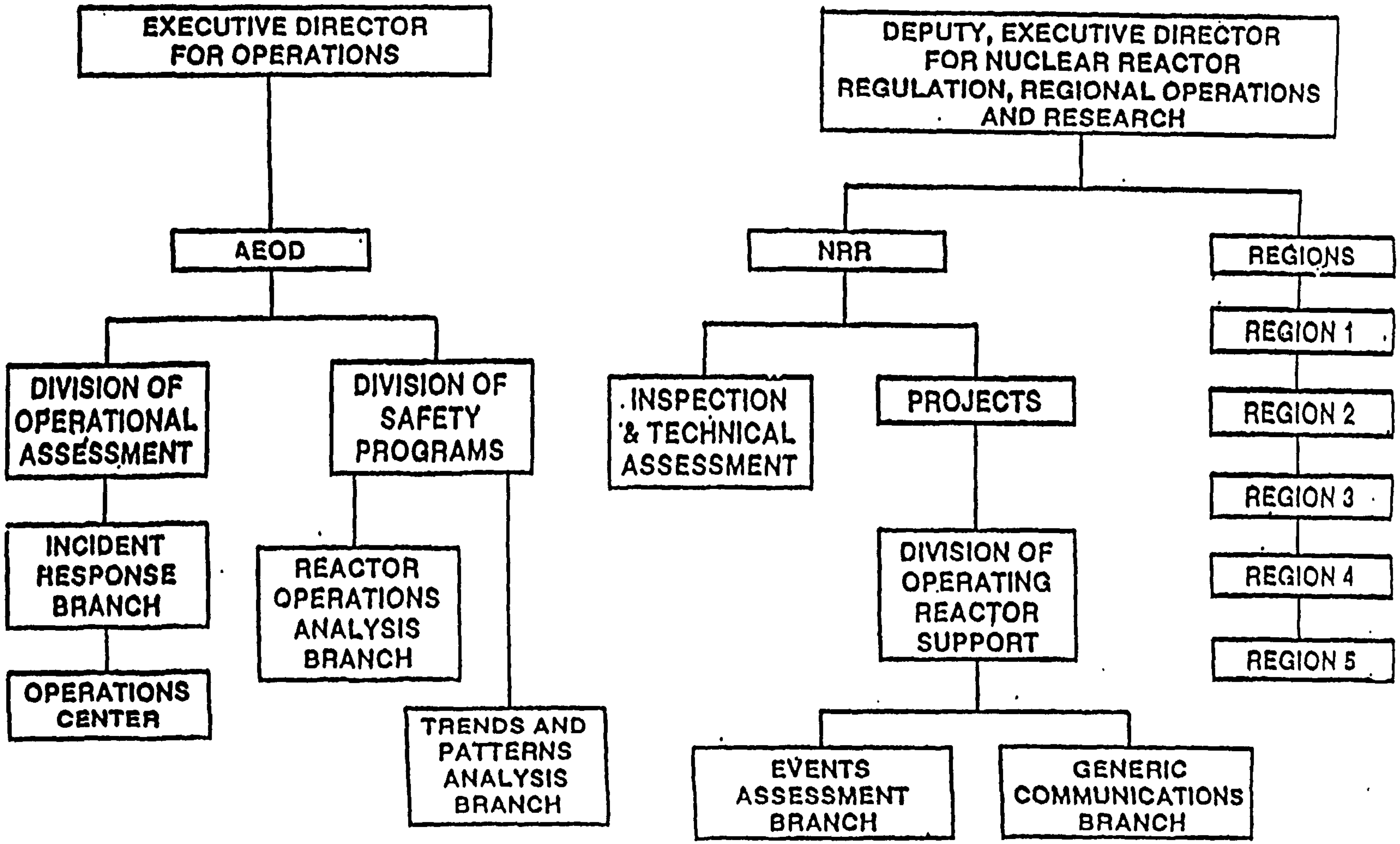


FIG. 1.1. USNRC organizations associated with review assessment and feedback of operational data.

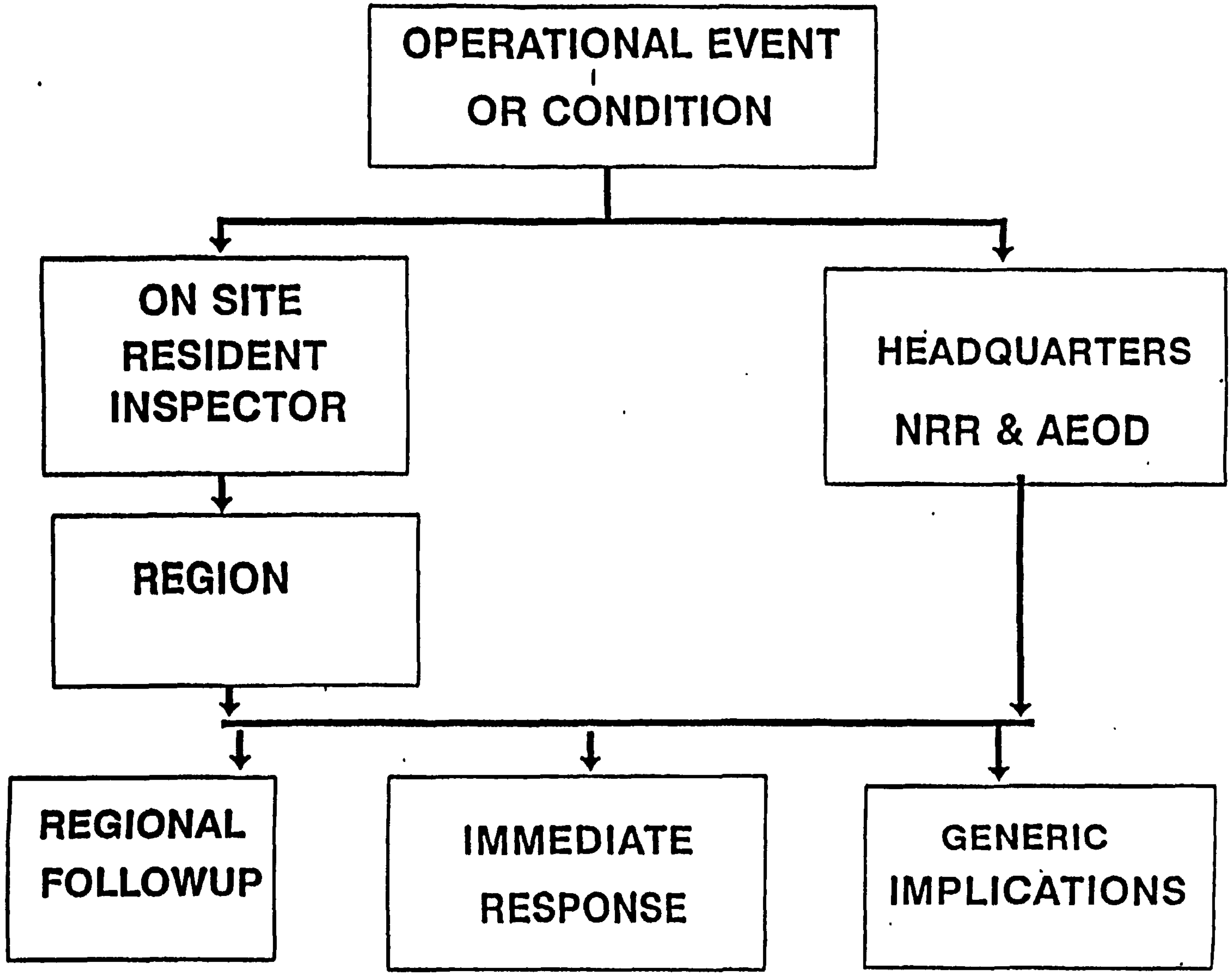


FIG. 1.2. Flow chart - Ways of addressing operational events.

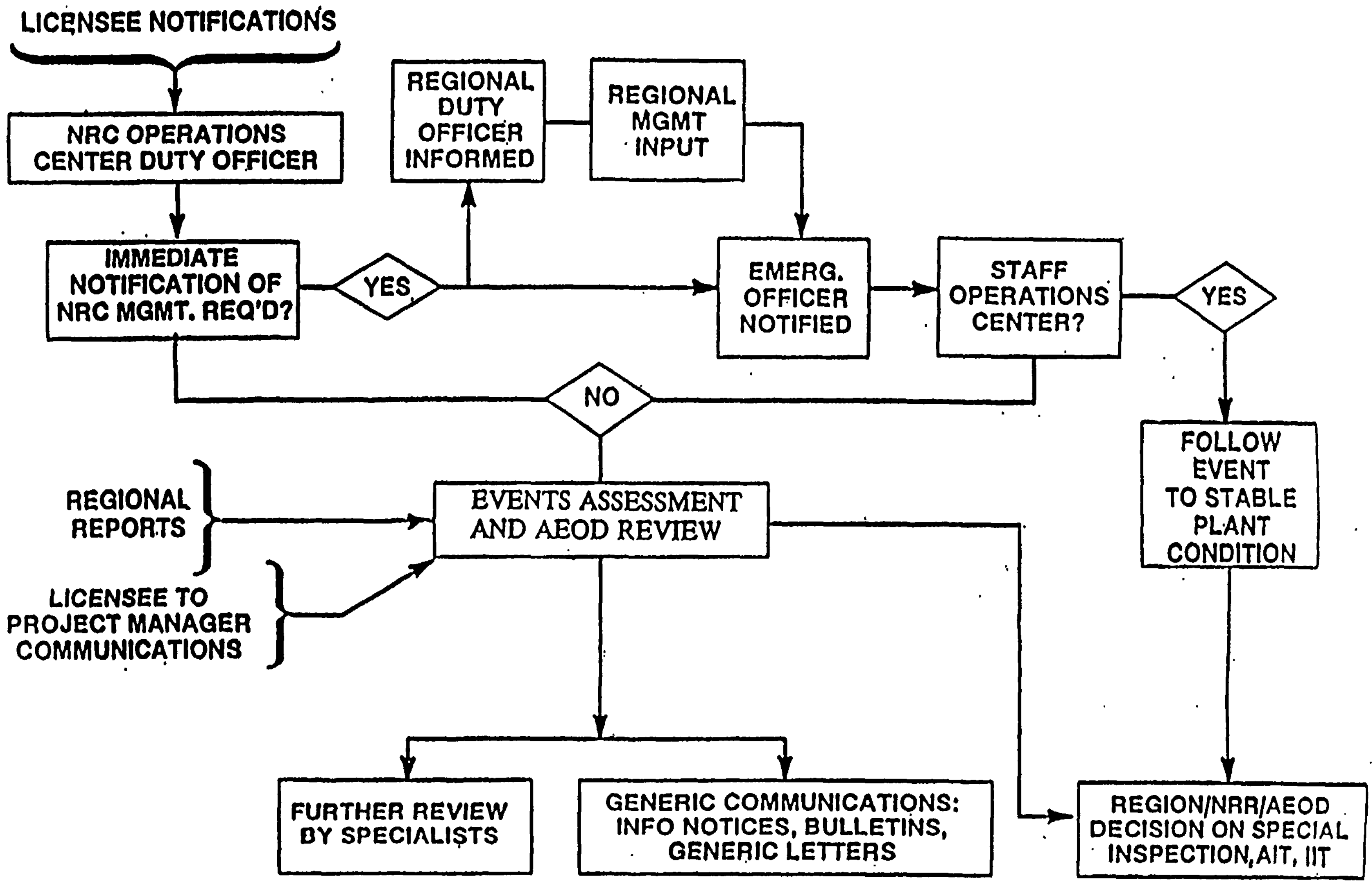


FIG. 1.3. USNRC rapid response to events.

2.1. NUCLEAR PLANT OWNER-OPERATORS (MANDATORY REPORTING)[1]

2.1.1. Prompt notifications: 50.72 Reports (Oral)

"Red" phone notification reporting incidents] to the USNRC. Types of events or conditions that are reportable are designated in Table 2.1.

2.1.2. Thirty-day notifications: 50.73 Reports (Licensee Event Reports) (Written)

Written report to the USNRC within 30 days as outlined in NUREG-1022. The types of events or conditions that are reportable are designated in Table 2.2.

2.2. MANUFACTURERS, SUPPLIERS, DESIGNERS OF NUCLEAR PLANTS AND THEIR EQUIPMENT (PART 21 REPORTS)

Timely notification of deficiencies associated with the products or services they provided (Code of Federal Regulations [10 CFR 21} reporting) which affect safety-related equipment or affect the plant safety margins or the health or safety of the public. 10 CFR 21 reports are made to the USNRC either by facsimile or telephone to the Headquarters Operations Center, or in writing to the USNRC Document Control Desk. The process for dissemination of 10 CFR 21 reports received by the Headquarters Operations Center is similar to the one used for prompt notification reports (50.72 reports) as described Section 3.1.1.1. The process for dissemination of written reports sent to the Document Control Desk is noted in Section 3.2.

2.3. US NUCLEAR REGULATORY COMMISSION INSPECTORS

2.3.1. Daily highlight reports

Events at licensed facilities deemed to be important by the resident inspectors are communicated (by phone) by the resident inspector to the appropriate USNRC regional office staff. If the regional office staff determines the event to be of significance or interest, the event (or condition) is documented in a Regional Office Daily Report. Often times, these events/reports are expanded upon and are distributed as Preliminary Notification (PN) of Event or Unusual Occurrences, and are then widely distributed within the USNRC. The procedure for review, analysis, and disposition of these types of reports are similar to that of 50.72 reports, as described in Section 3.1.1.2 of this report.

2.3.2. Periodic inspection reports

Routine inspections of licensed facilities are most often conducted by the resident inspector. Sometimes, the resident inspector is assisted by appropriate USNRC regional office based inspectors, or USNRC staff members knowledgeable in the areas of the inspection (e.g. procedures, human factors, specific equipment hardware, health physics, etc.).

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[1]These reports are systematically reviewed, analysed, and acted upon by several USNRC offices. Section 3.1.1 provides descriptions of USNRC's review process for 50.72 reports, and Section 3.1.2 provides descriptions of USNRC's review process for 50.73 reports.

TABLE 2.1. EVENTS OR CONDITIONS REQUIRING 50.72 REPORTS[a]

"RED" PHONE NOTIFICATION

Event classification	Type of event
Emergency	Declaration of an emergency
	Major loss of emergency assessment or communication capability
External events	Natural phenomenon that threatens safety
	Actual threat to safety of plant - should include toxic chemical spills
	Security events/fitness for duty
Transient	Engineering safety features actuation (including reactor scram)
	Emergency core cooling system discharge to reactor vessel
Design/major barrier problem	Principal safety barriers seriously degraded
	Unanalysed condition
	Outside design basis
	Condition not covered by procedures
	Events found while shutdown which significantly compromise safety
Equipment out of service	Initiation of shutdown required by Technical Specifications
	Deviation from plants Technical Specifications in an emergency
Common cause/common mode	Event or condition that alone could prevent shutdown, remove decay heat, control release of radioactivity, mitigate the consequences of an accident
Other	Radiation release above thresholds (gas, liquid)
	Transport of contaminated person off-site
	News media release or other government agency notification
	Fuel storage cask problem

[a]Extensive criteria and reporting requirements are provided in 10 CFR 50.72.

TABLE 2.2. EVENTS OR CONDITIONS REQUIRING 50.73 REPORTS[a]

30 DAY WRITTEN NOTIFICATION

Event classification	Type of event
External events	Natural phenomenon that threatens safety
	Actual threat to safety of plant
	Security events/fitness for duty
Transient	Engineering safety feature actuation (including reactor scram)
Design/major barrier problem	Principal safety barriers seriously degraded
	Unanalysed condition
	Outside design basis
	Condition not covered by procedures
Equipment out of service	Completion of shutdown required by Technical Specifications
	Operation prohibited by Technical Specifications
	Deviation from plants Technical Specifications in an emergency
Common cause/common mode	Event or condition that alone could prevent shutdown, remove decay heat, control release of radioactivity, mitigate the consequences of an accident
	Event where single cause caused at least one independent train or channel in multiple systems or two independent trains in single system to become inoperable
Other	Radiation release above thresholds (gas, liquid)

[a]Extensive criteria and reporting requirements are provided in 10 CFR 50.73.

2.4. NON-UNITED STATES PLANTS

2.4.1. International organizations

All notifications received from international organizations such as International Atomic Energy Agency (IAEA), Organization for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA), etc., are provided to AEOD for review analysis and dissemination. The confidentiality of the individual government, plant, and utility are protected. AEOD, USNRC's Office of Nuclear Reactor Regulation (NRR) events assessment branch (EAB), and other appropriate NRR branches also review the reports for significance and applicability to U.S. facilities. Information about events or conditions which are considered to be significant and applicable are dispositioned in a manner similar to that of significant prompt notification 50.72 reports (as described in Section 3.1.1.2).

2.4.2. Foreign governments, foreign utilities, other foreign sources

Individual reports received directly from other governments, non-US utilities, or other sources other than the international reporting agencies cited in 2.4.1 above are generally received by the USNRC's International Programs office which distributes them to AEOD and EAB. AEOD, EAB, and other appropriate NRR branches review the reports for significance and applicability to US facilities. Information about events or conditions which are considered to be significant and applicable to US plants are dispositioned in a manner similar to that of significant prompt notification 50.72 reports (as described in Section 3.1.1.2). The confidentiality of the individual government, plant, and utility are protected.

3. DISSEMINATION, REVIEW, AND ANALYSIS OF OPERATIONAL DATA

3.1. NUCLEAR PLANT OWNER-OPERATORS

3.1.1. Prompt notification 50.72 reports and regional daily reports

3.1.1.1. Dissemination of prompt notification 50.72 reports

50.72 reports ("Red" phone notification) are received at the USNRC's Headquarters Operations Center which is manned 24 hours a day, every day of the year by AEOD's Incident Response Branch (IRB). Following each call, the IRB duty officer determines in accordance with established criteria the need for immediate follow up. He notifies the headquarters emergency officer who determines if an item requires immediate USNRC management attention. Upper USNRC management (Office Directors, Regional Administrators, and the USNRC Commissioners) are notified of the event. If immediate USNRC attention is not required, the IRB duty officer notifies the appropriate regional duty officer, and the 50.72 reports are routinely transmitted electronically to appropriate USNRC offices (NRR/EAB, Vendor Inspection Branch [VIB], Probabilistic Risk Assessment Branch [PRAB], and AEOD's Reactor Operations Analysis Branch [ROAB], IRB, NMSS, Office of Public Affairs [OPA], etc.).

3.1.1.2. Review analysis and disposition of 50.72 reports and regional daily reports

Each normal workday morning (non-holiday, Monday through Friday), AEOD (ROAB and IRB), and NRR (EAB, VIB, PRAB) convene a conference call to discuss operational events that have occurred since the previous morning meeting.

The conferees discuss the events which were reported in 50.72 and regional daily reports. They determine which events warrant additional USNRC follow up and they decide which organization will conduct the detailed review and analysis.

Frequently, the initial information is scant and telephone discussions with the licensees, and sometimes site visits (plant operations or licensing staff) are necessary to obtain additional information about the event (or condition). Depending upon the perceived significance of the event or condition, the disposition of the event or condition may range from having the report entered into the computerized 50.72 database which is kept by AEOD's IRB, to shutting down the plant and having a special fact-finding team visit the facility (i.e. Special Inspection/Human Performance Evaluation [HPE] team visit/Augmented Inspection Team [AIT]/Incident Investigation Team [ITT].

Events described in 50.72 reports, regional daily reports, and other sources (such as foreign reports, Part 21 Reports, etc.) which appear to be significant and are chosen by AEOD or NRR/EAB for follow up action are reviewed by a special events assessment panel. the panel meets weekly to evaluate the significance of each such event. The panel is comprised of six members, five from NRR, and one from AEOD/ROAB. The chief of EAB serves as the panel chairman.

The panel's final disposition of the significance of the events is provided to USNRC management for consideration or further action (e.g. generic communication, enforcement or other regulatory action). Events or conditions that are considered to be significant by the panel are reported and issued in a USNRC headquarters daily highlight report. The headquarters daily highly report is widely disseminated in the same manner as the Regional Office Daily Report (as noted in Section 3.1.1.1).

Often times, the AEOD or EAB may determine that a generic communication is warranted to alert other nuclear plant operators (U.S. and foreign) of the event. Depending upon the perceived significance of the event, the USNRC can issue one of several types of generic communications ranging from an Information Notice (IN) which alerts licensees to the event but does not require it take specific action, up to issuing an order which may require licensees to make plant modifications and perform analyses which require significant expenditure of manpower and money. The generic communications are prepared and issued by NRR's Generic Communications Branch. AOED has the responsibility of issuing reports of important events at U.S. commercial reactors to non-U.S. organizations, via OECD/NEA incident reporting system (IRS) reports.

To complete the systematic review of operational data, IRB retains a searchable computerized database of all 50.72 reports, and EAB is required to write brief closeout reports to document USNRC actions that were taken in response to each 50.72 report.

3.1.2. Licensee event reports: 50.73 reports

3.1.2.1. Dissemination of licensee event reports: 50.73 reports

The licensees submit licensee event reports (LERs) (mandatory per 10 CFR 50.73) to the USNRC's Document Control Desk, the NRR project manager, the appropriate USNRC Regional Administrator, the USNRC resident inspector and the Institute of Nuclear Power Operations (INPO). Subsequently, the USNRC document control staff distributes copies to AEOD, EAB, and other USNRC staff. Copies are also sent to the plant's local Public Document Room and to USNRC contractors who operate and maintain the Sequence Coding and Search System[2] (SCSS) and Nuclear Documents System (NUDOCS) databases. SCSS and NUDOCS are readily accessible, computerized databases. SCSS contains only LERs; however, each LER is subdivided into individual sequences. The sequences are encoded by the contractor and as a result, the individual sequences are also searchable. Features of SCSS are shown in Table 3.1.

Each LER is also entered into the NUDOCS database (a text searchable database). In addition to all recent LERs, the NUDOCS database contains almost all correspondence between the USNRC and licensees, manufacturers, the general public, and internal USNRC documents. Other operational experience documents noted in this paper which are stored (and readily accessible from NUDOCS) are inspection reports, PNs, GLs, INs, Bulletins, AOED studies, etc.

[2]SCSS is a complex system which requires more explanation than is allowable in this paper.

TABLE 3.1. SEQUENCE CODING AND SEARCH SYSTEM

MAJOR FEATURES

•	All LERs - only LERs
•	Maintained for AEOD by contractor (Oak Ridge National Laboratory)
•	Coded sequence of events for each LER
•	Cause-effect relationships shown
•	Initial conditions given
•	Events given
•	Boolean logic to combine data elements

3.1.2.2. Review, analysis and disposition of licensee event reports: 50.73 reports

Although many staff at USNRC review LERs, AEOD is tasked with the responsibility of reviewing all LERs. Figure 3.1 is a flow chart illustrating AEOD's process for review and evaluation of LERs. Table 3.2 is a list of the criteria used by AOED for reviewing ("screening") LERs (and other reports as well) for significance. As part of AEOD's screening process, each LER is assigned one of four categories depending upon its safety importance as outlined in Table 3.3.

In determining the course of long-term follow up to be taken for events classified as significant by AEOD's screening process for which no USNRC action has been taken (NRR or Regional Offices), AEOD considers the following questions:

1. What happened?
2. Why did it happen?
3. Should it have happened?
4. Has it happened before?
5. What could have happened?
6. What are the lessons learned?
7. What corrective actions are needed?

3.2. DISSEMINATION, REVIEW, ANALYSIS AND DISPOSITION OF PART 21 REPORTS

Part 21 reports which are received by the USNRC's Headquarters Operations Center are disseminated, reviewed and analysed in the same manner as 50.72 reports (see Sections 3.1.1.1 and 3.1.1.2 of this report).

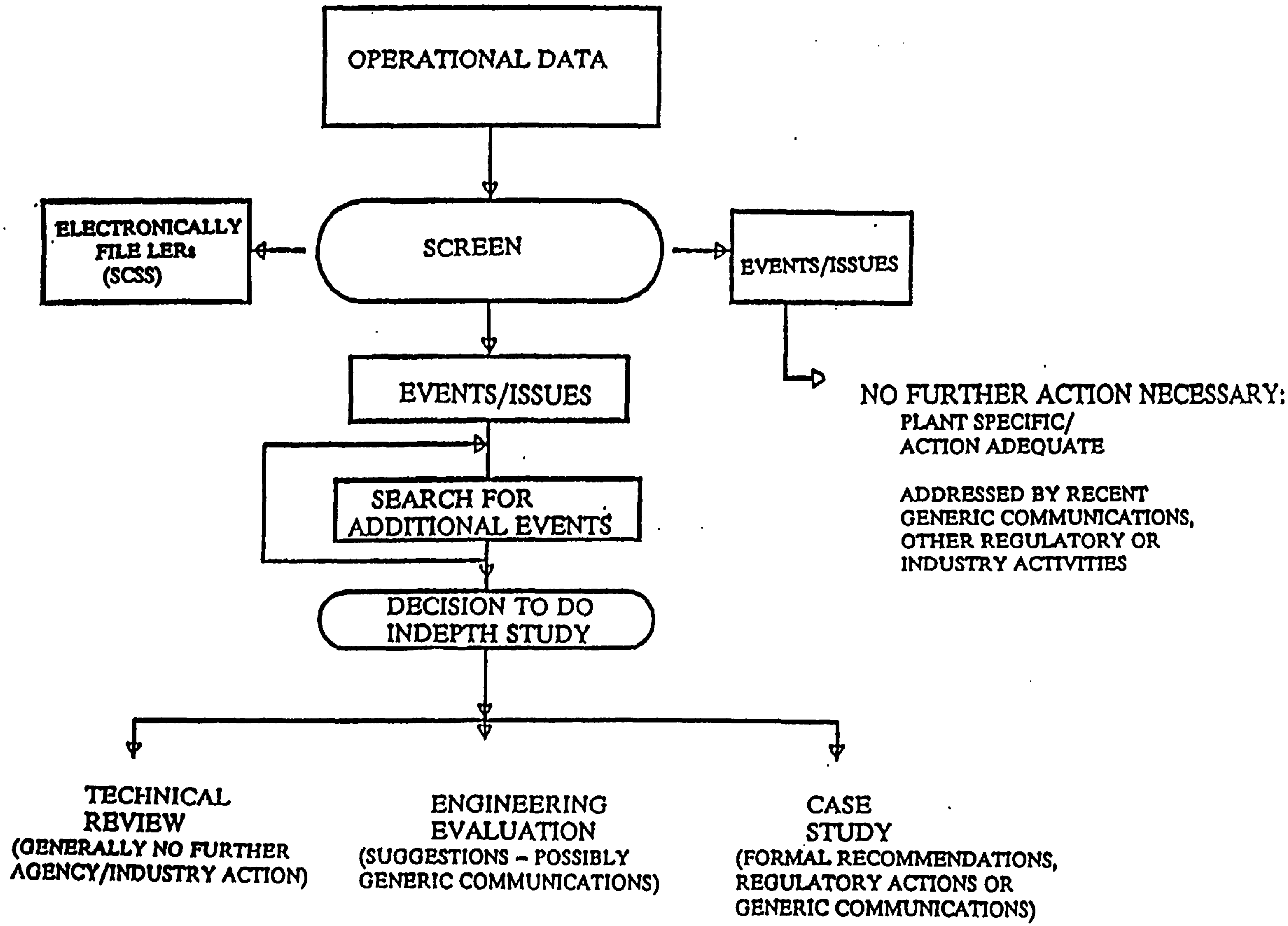

FIG. 3.1. AOED's review and evaluation of LERs.

TABLE 3.2. TYPICAL AEOD CRITERIA FOR IDENTIFYING SAFETY SIGNIFICANT EVENTS

1.	Event sequence not previously analysed which could be far more serious with credible alternative conditions.
2.	System interaction resulting from a previously unrecognized interdependence of systems and components.
3.	Improper operation, maintenance, or design that has or could cause common cause/common mode failure of a safety system.
4.	Unexpected system or component performance with serious safety implications or radiation release.
5.	Multiple failures (including personnel errors) occurred in the event.
6.	Equipment failures (including non-safety equipment) that caused serious transients and challenges to safety systems.

TABLE 3.3. CRITERIA FOR DETERMINING LER CATEGORIES

Category	Criteria
1	Those events of such obvious importance that actions should be initiated immediately by AEOD or other office or organization to ensure plant safety. (If action is initiated by another office, AEOD may not need to take action.) Because LERs are received more than 30 days after an event, it is rare when LERs fall within this classification.
2	Those events (or combinations of events) which appear to have safety importance but do not require immediate action to ensure plant safety.
3	Those events (or combinations of events) which require additional consideration to permit assignment to Categories 1, 2 or 4.
4	Those events with little apparent importance to safety. Those events together with all other LERs are included in the SCSS database for possible automated analysis of trends and patterns or search and review as part of a subsequently defined specific safety concern.

Written Part 21 reports are submitted to the USNRC Document Control Desk. The written reports are entered into the NUDOCS database, and they are forwarded to NRR's VIB for review and disposition. VIB staff evaluates the significance of the information provided in the written reports. The evaluation frequently necessitates additional communication with the vendors and the affected utilities. Based upon Part 21 reports and other operating experience (50.72, 50.73 reports, inspection reports, etc.), VIB frequently conducts inspections of vendors' and licensees' facilities to verify that they comply with the quality programmes that they have committed to in accordance with USNRC regulations (e.g. 10 CFR, Part 50, Appendix B).

VIB is tasked with the responsibility of managing Part 21 report database[3], and VIB is also responsible for keeping track of USNRC, licensee, and vendor actions which result from each Part 21 report. With regard to Part 21 reports, the USNRC's most important function is to assure that significant events (or conditions) are disseminated to all possible affected facilities. Most often, dissemination is provided by the manufacturer; however, on many occasions, VIB has authored generic communications (e.g. GLs, INs, Bulletins) to inform all licensees.

3.3. DISSEMINATING REVIEW, ANALYSIS AND DISPOSITION OF NRC INSPECTOR REPORTS

3.3.1. Daily highlight reports

The daily highlight reports are disseminated, reviewed, analysed and dispositioned in a manner similar to that of 50.72 reports (described in Sections 3.1.1.1 and 3.1.1.2). One of the major differences is that the daily highlight reports are not entered in the 50.72 database, or NUDOCS. However, daily highlight reports that are expanded upon and distributed as a PN of event or unusual occurrence are entered into the USNRC's computerized NUDOCS database.

3.3.2. Periodic inspection reports

The inspection reports are widely distributed within the NRC (including the licensing project manager in NRR, EAB, AEOD). Externally, the inspection reports are distributed to the licensee, INPO, and the local public document room.

All inspection reports are entered into the USNRC's computerized NUDOCS database. EAB and AEOD routinely review inspection reports. When operating experience noted in an inspection report is germane to issues or events that AEOD and EAB are studying, they are considered in AEOD and EABs studies, and they are frequently included or referenced in AEOD and EAB studies, and in USNRC generic communications.

An important mechanism for feedback of operational data noted in inspections and inspection reports is that of direct communication between the USNRC and the utilities management. In the event that the inspectors find a condition which is not in conformance with plant's license conditions (Technical Specifications or plant safety analysis), the USNRC inspectors discuss their findings with plant management during their inspections. In some rare cases, when the inspector finds a condition where significant reductions in safety margins exist, the licensee may be required to reduce power levels or shut down immediately. In addition to oral discussions with the plant management at the completion of the inspection and follow up written reports, the USNRC's regional office may notify utility upper management of any area in which the plant appears to be operating in violation of its license. The utility is required to respond directly to the USNRC managements' "Notice of Violation" within 30 days, and the utility must outline actions to be taken to bring the plant into conformance with its license.

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[3]In addition to being entered into the Part 21 database, the VIB inspection reports are entered into the USNRC's computerized NUDOCS database.

4. CLOSING THE FEEDBACK LOOP

4.1. UNITED STATES PLANTS

U.S. plants are informed by the USNRC of the result of its evaluations of operational experience by either generic communications (INs, Gls, Bulletins), NUREG reports, letters from the regional offices (as noted in Section 3.3.2). In addition, EAB and AEOD communicate periodically with industry groups such as INPO, EPRI, and NUMARC informing them of major findings in the areas of operational data analysis.

These industry groups receive copies of all major non-proprietary AEOD and NUREG reports on operational data, and they inform their member utilities of such activities.

Frequently, USNRC holds workshops with licensees informing them of major findings in the area of operational data and provide guidance on regulatory issues and generic communications.

Many USNRC members (AEOD, NRR, USNRC's Office of Nuclear Regulatory Research) serve on standards committees for the American Nuclear Society (ANS), Institute of Electrical and Electronics Engineers (IEEE), and the American Society of Mechanical Engineers (ASME). As such, they interface with U.S. industry to feedback operating experience into appropriate industry codes and standards. In addition, staff frequently interact with industry and present results of their studies at appropriate professional society meetings (ANS, IEEE, ASME, etc.).

4.2. UNITED STATES CONGRESS

Currently, four times a year, the USNRC issues an "Abnormal Occurrence Report" to the U.S. Congress informing them of significant events which occurred at U.S. facilities (commercial reactors, research reactors, biological, medical, and non-reactor applications, storage and disposal of non-military nuclear materials, byproducts and wastes). These reports are issued as NUREG reports, and as such, are widely circulated within the nuclear community.

4.3. NON-UNITED STATES PLANTS

AEOD submits reports to OECD/NEA (IRS) to inform the international community of important operational events which have occurred at U.S. plants.

IRS reports are provided to inform member countries of the major safety significant events (such as events which were reported to the U.S. Congress in Abnormal Occurrence reports, the results of IITs, AITs, generic communications, AOED case studies, etc.).

Annex X
OPERATIONAL SAFETY FEEDBACK SYSTEM IN ARGENTINA

OPERATIONAL SAFETY EXPERIENCE FEEDBACK SYSTEM
Description of procedures and criteria used in Argentina

Fundamentals

One of the primary objectives in Nuclear Safety is preventing the occurrence of severe accidents. A methodological analysis of the most critical accidents occurred in Nuclear Power Plants allows for the following conclusions:

1. The failures that led to severe accidents had already occurred before at the same installation or in other similar ones; however, due to diverse fortuitous circumstances they had not had significant consequences as far as safety is concerned.
2. Important accidents are the result of the appearance of several combined small failures along with the occurrence of some human errors.

Considering the above reasons, in order to increase safety in nuclear installations, a system must be available allowing for an adequate and profitable use of Operational Experience, in which relevant events are used as an efficient indicator for pointing out eventual failures and weaknesses existing in the systems, as well as the safety-relevant operating conditions.

Background information:

The Operating License of the Atucha NPP in 1974 and that of the Embalse NPP in 1984 contain requirements for the communication of incidents and abnormal situations during operation.

In 1981, the Regulatory Authority issued its Standard AR-3.9.2 regulating, in further detail, the communications and reports that must be prepared by the installations when relevant events occur or when conditions affecting safety arise.

As of 1983, the Regulatory Authority has had an active participation in the meetings that led to the creation of the IAEA's IRS System, later on participating in the same. Further on, it integrates the INES system for fast incident reporting following the international severity scale.

Simultaneously, the Nuclear Power Plants in operation integrate the operators' incident reporting systems, WANO, and COG in the case of the Embalse NPP.

In 1988, the Safety Analysis Division enforces the use of the IRS Data Base.

As of 1994, a team for cooperation among CANDU Senior Regulators is integrated.

Presently, the System is defined by the Installation's Procedures (PI-1063) and by the Regulatory Authority's Procedures (PA-12 and PL-34 Rev 3), which are expected to be compatible and coherent with each other.

Objective of the System's Procedures:

In the process of making good use of operational experience, participation by the facility and by the regulatory authority must be fostered; in some cases, the designers or suppliers of components must also participate, as well as international agencies distributing information. This situation requires that all the participating agencies generate their own procedures and that the latter are coherent and compatible with each other.

The set of procedures should allow for fulfilling the following:

a) Defining a mechanism for the reception and preparation of reports on significant events (format of the reports).
b) Defining a method for the transmission, management and filing of the information in an adequate data base, so as to ensure its traceability.
c) Defining a method for the analysis and selection of events that may lead to eventual corrective actions.
e) Appointing the agencies and persons responsible for performing the analyses and for making decisions.
f) Defining consultation mechanisms and interfaces among designers, manufacturers, operators and regulators, so as to design the practical measures that will be required or recommended, taking into account costs, risks and the feasibility of the corrective actions.
g) Defining how to perform a follow up of the adopted corrective actions.
h) Defining a mechanism for data transmission, so as to diffuse operational experience.

Basic Principles of the System

1. The most important factor is that highly qualified and experienced professionals are used to analyze the events, assess the root causes and design the corresponding corrective actions. The quality of the feedback is directly proportionate to the quality of the participating professionals.
2. The Regulatory Authority's participation and leadership in the System is highly important for two reasons:
   - Incidents are very valuable safety indicators in controlled installations and may lead to changes in the requirements or standards issued by the Authority.
   - A fair judgment on the conditions and circumstances under which incidents and failures occur is a basic element in assessing the causes and this generally *implies self-criticism by the involved operating personnel itself*, which is not always easy to attain. This is why an independent review of the analysis and evaluation tasks must be ensured.

THE 8 STAGES OF THE DATA MANAGEMENT PROCESS

1st. STAGE: Detection, characterization and communication.
2nd. STAGE: Classification and inclusion of reports in data bases.
3rd. STAGE: Selection of events for analysis.
4th. STAGE: Analysis of Direct and Root Causes.
5th. STAGE: Assessment of Corrective Actions.
6th. STAGE: Review, Search for consensus and Approval.
7th. STAGE: Implementation and follow-up of Corrective Actions.
8th. STAGE: Diffusion of the experience gained.

Reporting System

The events included in any of the following categories must be reported:

1. A significant decrease of the safety level in the systems related to: reactivity control; pressure, flow or temperature control in the primary circuit; the parameters in the moderator system; the parameters in the secondary system.
2. Detection of deficiencies in the installation's design, construction, operation or Quality Assurance Program.
3. A risky situation that was not established, analyzed or foreseen in the Safety Report.
4. Verified non-availability of the protection systems, of safety-related instruments or of essential supplies.
5. A significant degradation of one of the main safety barriers (fuel element clads, primary pressure circuit and confinement system).
6. Installation's Boundaries or Operating Conditions surpassed.
7. Occupational exposure or discharge of radioactive effluents into the atmosphere exceeding the corresponding authorized limits.
8. Natural or man-induced internal or external events that may conceivably affect, either directly or indirectly, the installation's safety.
9. Non-scheduled outages.

Reporting terms and format of the reports

1. Immediate reporting (by FAX or eMAIL) ⇒ (form # 1)
2. Reporting within 24 hours (> level 1 in the INES scale) ⇒ (form # 2)
3. Detailed report (including cause assessment) ⇒ (form # 3)

Committee for the Review of Operating Experience:

Its task is performing a follow-up of operating experience and recommending corrective actions on the basis of the following data:

- Reports on Relevant Events.
- Minutes of the Responsible Organization's Technical Review Committee
- Minutes of the Internal Safety Committees of NPPs in Operation.
- Performance Indicators.
- Data Base of the IAEA's IRS System.
- Data Base of the INES System.
- Reports from the "Program for cooperation among CANDU reactor Regulators".
- Results of the Operation Reliability Program.
- Results of the In-Service Inspection Program.
- Results of the Periodic Tests Program.
- Results of the Preventive Maintenance Program.
- Results of the Material Aging Monitoring Program.
- Inspection Reports.

Committee for the Review of Operating Experience is membered by specialists in the fields of Nuclear Safety, Probabilistic Safety Analysis, Quality Assurance, Reactor Operation and Maintenance, Human Factors and Radiological Protection.

Completion of the cycle for profiting from operational experience

A follow-up of the corrective actions, so as to assess the efficacy of their application and the fitness of the results obtained, is of prime importance in the completion of the Cycle for Profiting of Operational Experience.

The utilities shall apply the necessary measures for corrective actions to be performed. The Quality Assurance programs in the installations shall involve procedures for evaluating the performance of corrective actions and the efficacy of their results.

Formulario N° 1

COMUNICACIÓN INMEDIATA DE EVENTOS RELEVANTES

N°	Instalación	Ocurrencia del evento	DÍA | MES | AÑO | HORA
Potencia antes del evento (Mwt)		Informado Inspector Residente SI ☐ NO ☐	

TIPO DE EVENTOS	
SALIDA DE SERVICIO NO PROGRAMADA	☐
EVENTO EN EVOLUCIÓN CUYO CONTROL NO FUE AUN LOGRADO	☐
EVENTO QUE AFECTA A LAS BARRERAS DE SEGURIDAD	☐
DEGRADACIÓN DE UNA FUNCIÓN DE SEGURIDAD	☐
FENÓMENO NATURAL O EXTERIOR QUE AMENAZA LA SEGURIDAD	☐
LIBERACIÓN DE MATERIAL RADIACTIVO	☐
IRRADIACIÓN O CONTAMINACIÓN DE PERSONAL	☐
AMENAZA DE SABOTAJE	☐
ROBO DE MATERIAL RADIACTIVO	☐
TODO OTRO EVENTO CONSIDERADO RELEVANTE	☐

DESCRIPCIÓN SINTÉTICA DE LA SITUACIÓN Y MEDIDAS ADOPTADAS

Formulario N° 2
COMUNICACIÓN EN 24 horas. DE EVENTOS RELEVANTES

N°	Instalación	Rev.	Ocurrencia del evento DÍA | MES | AÑO | HORA

Potencia antes del evento (Mwt)	Potencia en el momento de la comunicación (Mwt)

TIPO DE EVENTOS	
DISPARO DE UN SISTEMA DE SEGURIDAD	☐
EVENTO QUE AFECTA LAS BARRERAS DE SEGURIDAD	☐
DEGRADACIÓN DE UNA FUNCIÓN DE SEGURIDAD	☐
FENÓMENO NATURAL O EXTERIOR QUE AMENACE LA SEGURIDAD	☐
AMENAZA DE SABOTAJE	☐
DESAPARICIÓN DE MATERIAL RADIACTIVO	☐
PERDIDA SIGNIFICATIVA DE COMUNICACIÓN CON EL EXTERIOR	☐
PARADA O REDUCCIÓN DE POTENCIA	☐
LIBERACIÓN DE MATERIAL RADIACTIVO FUERA DE LA CENTRAL	☐
LIBERACIÓN DE MATERIAL RADIACTIVO DENTRO DE LA CENTRAL	☐
IRRADIACIÓN O CONTAMINACIÓN DE PERSONAL	☐
DAÑO FÍSICO EN EL PERSONAL	☐
TODO OTRO EVENTO CONSIDERADO RELEVANTE	☐
SISTEMAS AFECTADOS:	
COMPONENTES FALLADOS:	

DESCRIPCIÓN DEL EVENTO/SITUACIÓN

CAUSAS Y FACTORES DESENCADENANTES
LIBERACIONES DE MATERIAL RADIACTIVO OCURRIDAS:

MEDIDAS ADOPTADAS

ORGANIZACIONES O PERSONAS INFORMADAS	
ESTATALES:	LOCALES:
OTRAS:	

Formulario N° 3

FORMATO PARA LA COMUNICACIÓN DETALLADA DE EVENTOS RELEVANTES AL ENREN

1. Categorías de los sucesos que requieren informe:

2. Situación de la central antes del suceso:

3. Sistemas fallados/afectados:

4. Componentes fallados/ afectados:

5.1 Causas observadas:

5.2 Causas fundamentales:

6. Consecuencias para la explotación:

7. Características del suceso:

8. Tipo de falla:

2. RELATO DESCRIPTIVO.
3. EVALUACIÓN DE LA SEGURIDAD.
4. CAUSAS FUNDAMENTALES Y MEDIDAS CORRECTIVAS.
4.1. CAUSAS FUNDAMENTALES.
4.2. MEDIDAS CORRECTIVAS.
5. ENSEÑANZAS OBTENIDAS.

Annex XI
THE STAGBAS2 DATABASE AND THE PRODUCTION OF THE INCIDENT CATALOGUE AND TREND CATALOGUE IN SWEDEN

1. INDICATORS AND INDICATOR SYSTEMS

Safe operation is crucial for the nuclear power plants. Unsafe operation can influence both the employees and the surrounding environment in a negative way. It is therefore important to monitor and control the plant safety status and thereby avoid a change towards a potentially unsafe state.

It has been recognized that accidents with possibly serious health and economic risks, hardly may occur without any premature, often unknown incident, which will be a warning to the operator that the plant safety status is degrading.

These indicators can provide timely indications of changes in the plant's safety margins and can give more direct information on the factors contributing to the risk level of the plant. The indicators or indicator combinations can thereby provide a means for warning of impending problems before actual accidents occur. In addition to their preventive function the indicators are aimed to facilitate the follow up of the effectiveness of the corrective actions.

Indicators of various kinds are currently being used by the nuclear industry and regulatory authorities to monitor plant performance. These indicators reflect a set of operational data that have some degree of correlation with individual plant safety performance. The standardization of the so called "Overall Performance Indicators" (unit capability factor, unplanned automatic scrams/1000h critical operation, collective radiation exposure, etc.) has been considered to be an important matter of worldwide interest. An example of such indicators is presented in the annual reports published by the Swedish Radiation Protection Institute of releases of activity to the environment and radioactive exposure to different worker groups in an NPP.

In 1990, both WANO (World Association of Nuclear Operation) and IAEA (International Atomic Energy Agency) implemented ten standardized Performance Indicators (PIs) for international use. In 1991/1992, however, the IAEA took actions to define and create performance indicators to be used by the regulatory bodies. In March 1992, regulators met in Vienna at IAEA to discuss and write a first draft to be reviewed at a later time, as a guideline for regulators. In late 1992 the draft report will be reviewed at a technical committee meeting organized by the IAEA. However, some concern has been raised about the relation to safety of some of these overall performance indicators. Therefore, an additional need for development of more plant specific and explicitly safety related indicators has been recognized. A preventive function might be fulfilled better by using more detailed and in-depth performed qualitative and quantitative analyses of safety significant incidents. That is one of the reasons why SKI puts lot of efforts on creating its own indicator system.

These types of Operational Safety Indicators (OSIs) are based on all kinds of operational experience. SKI has developed an indicator system based on LERs (Licence event reports) and reactor trip reports. At present, a research project, SIK-1, is going on with the goals to define suitable indicators and to create complete indicator systems. The project is part of the Nordic nuclear research programme (NKS).

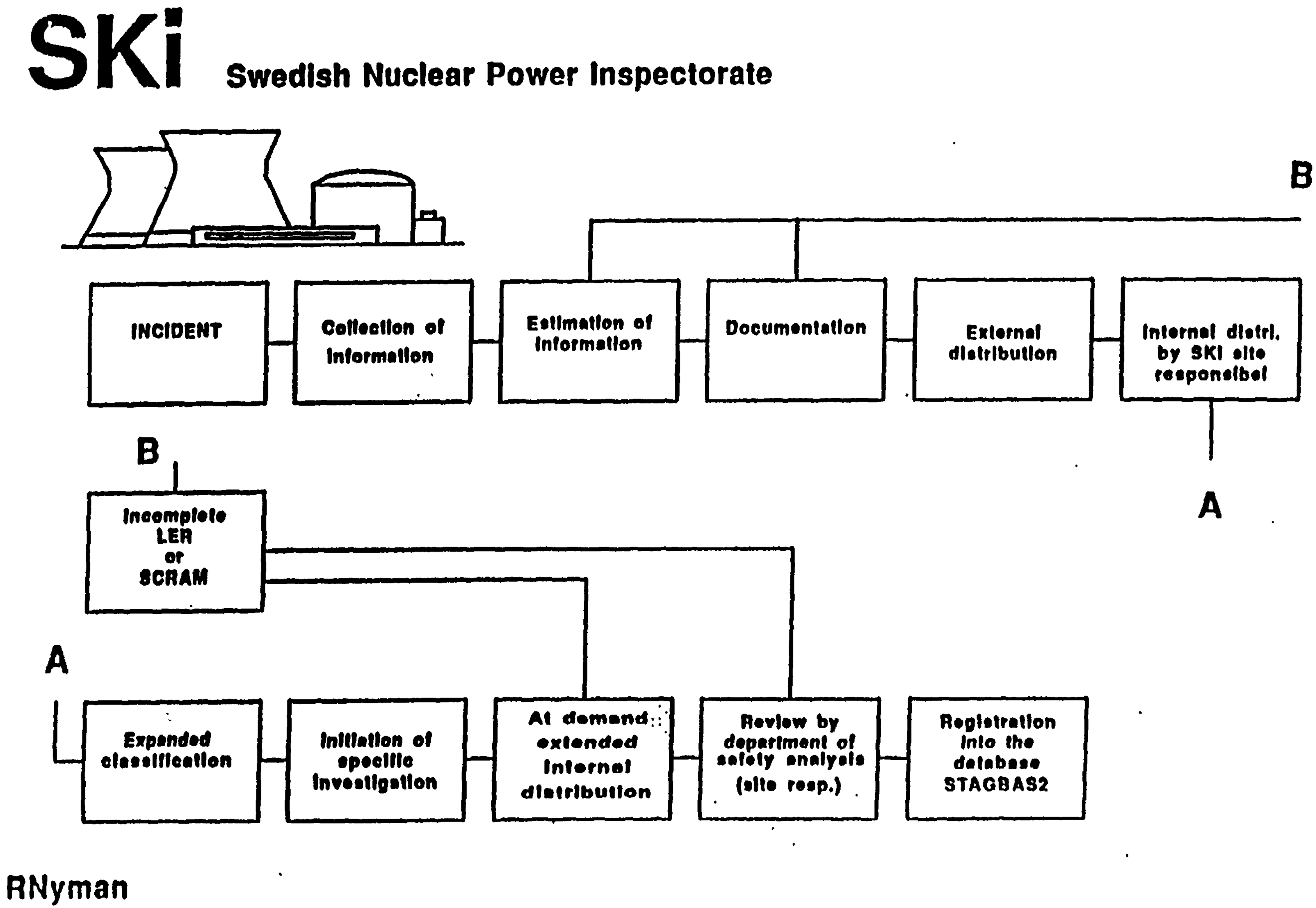

FIG. 1. From incident to database (StagBas2) - Loop of information flow.

2. AN INDICATOR SYSTEM BY USING STAGBAS2, THE INCIDENT CATALOGUE AND TREND CATALOGUE

Failures in safety related systems result in an LER report. As an example, failure during a test of a safety related system will be reported. Potential "LER events" are discovered by or reported to the control room operators. Staff from the control room and from different technical departments have daily meetings where they evaluate different events and make a judgement on whether or not the event shall be treated as an LER. The judgement is made on the basis of the LER framework put together in the plant specific Technical Specifications. The LER is then reported to SKI and stored by SKI staff in the database STAGBAS2. The loop of information flow is shown in Figure 1.

The amount of stored information in the database is large. To be able to see and find tendencies of stored data, the content of information is presented in a systematic way in an incident catalogue. With these reports, one for each NPP in Sweden as basic documentation, SKI will afterwards extract and transfer a set of interesting indicators, into a new summarizing report - the "trend catalogue". The chosen indicator or observation area will be a goal for further investigations and analysis. The different disturbance analyses and suggestions for safety improvements are also stored in the trend catalogue. The document will therefore list a set of observation areas to be used at the SKI and by SKI inspectors in the nearest future.

Besides the above mentioned routines, whose aims are to identify failure frequencies and trends, there are well prepared programmes for the judgement of each unique LER at SKI by its specialists.

2.1. THE STAGBAS2 DATABASE

STAGBAS2 provides information about a wide spectrum of events ranging from LER reports with less significant effects on safety systems or components to more serious incidents, e.g. of the type - human errors. There are, however, no quantitative failure data failure rates etc. stored in that database. That kind of information is available from other types of databases, common for the nuclear power plant community and the regulatory body.

The overall objective with STAGBAS2 is to be able to search, condense and analyse incidents and incident information and thereby reveal significant conditions that otherwise would be hidden in the large amount of information available. The LER reports shall have the level of detail and quality, necessary to allow valid analyses of safety indicators and trend analyses. Until now, all LERs and reactor trip reports from 1983 and onwards have been updated in accordance with new procedures at SKI.

STAGBAS2 is today suited to the requirements on classified input data made by SKI and to take advantage of the detailed information about events available in descriptive text.

The LER event is documented on a special LER form. This documentation is made by operating support personnel guided by the special procedures for this task. The report should then be made available to the authority without any delay, within 24 hours. Supplementary information is added by SKI on a special accompanying sheet. The reporting sheet and the accompanying sheet is then registered in STAGBAS2. The event descriptions in STAGBAS2 are detailed, in principle as detailed as is practically possible with regard to the large number of events. The detailed descriptions give many possibilities for definition and choice of different safety indicators.

LER classification is guided by a classification scheme. The scheme aims to describe how events reports shall be documented in STAGBAS2. The scheme is used to achieve a stable classification of the reports, so as not to depend on when and by whom the classification is performed. The scheme identifies all descriptive fields and allowed keywords. Classification examples are also given. A quick reference guide with the most commonly used keywords is available.

30-40 different descriptive areas (data fields) are used to define an event. Each event is one record in the database. Examples of descriptive areas are: affected plant, date, power level, failure cause, failure mode, component type, component identification and corrective actions taken. Descriptive texts or keywords are entered into each field. Each record in the database is automatically given an individual access number by the database handler.

STAGBAS2 is currently running in a UNIX environment on a SUN Sparc Workstation. It is accessible from different workstations connected to the computer network SKINET at SKI. It is also accessible through modem connections for external users.

Figure 2 visualizes two parameter values which are very easy to pick up from the STAGBAS2 database. The left hand part of the figure shows the TTF, time to failures history and the right hand figure shows the TBF, time between failures history. The beginning of the time scale, 83.01.01, is at the top of the figure and the bottom line of the figure is the end of the analysis time, 91.12.31. The high peak on the TBF part is roughly in 1985-1986 and gives quite a good insight of the history of operating experience and the maintenance learning process. The report frequency is constantly on a high level since 1985-1986.

2.2. THE INCIDENT CATALOGUE

The objective of the STAGBAS2 database, also mentioned earlier, is that the user shall be able to easily trace information about different safety related events using the keywords as search entries, and to analyse different variations (trends). The results of the different search patterns are compiled in a special incident catalogue. The incident catalogue presents this information in diagrams, there are, e.g. diagrams showing the yearly rate and the time trends for certain types of events.

The idea of the catalogue is, in addition to give an understandable view of the STAGBAS2 content, to be a generator for new ideas concerning actions to increase the safety. Different trends/results for different plants should be objects for questions and/or investigations of the causes to this. The catalogue shows the numbers and frequencies and the distribution of different incident types.

The presentation in the catalogue is hierarchial. First, general criteria as the number of LERs per plant or per safety function (a group of systems) and last, failure causes for specific component types are presented. The catalogue is subdivided into one section covering plant specific issues and another section covering generic issues. The generic part summarizes the results from a given search pattern applied on the set of events collected from all Swedish NPPs. The trend analyses which are based on the combined experiences from all plants are often based on a large and well defined set of data. Plant specific analyses are always based on the incidents reported for each specific plant and the possibilities to analyse are therefore limited because of the small number of events available. The division into one plant specific and one generic part makes it possible to compare trend and quantitative results for a specific plant with results for the whole set of plants (generic results).

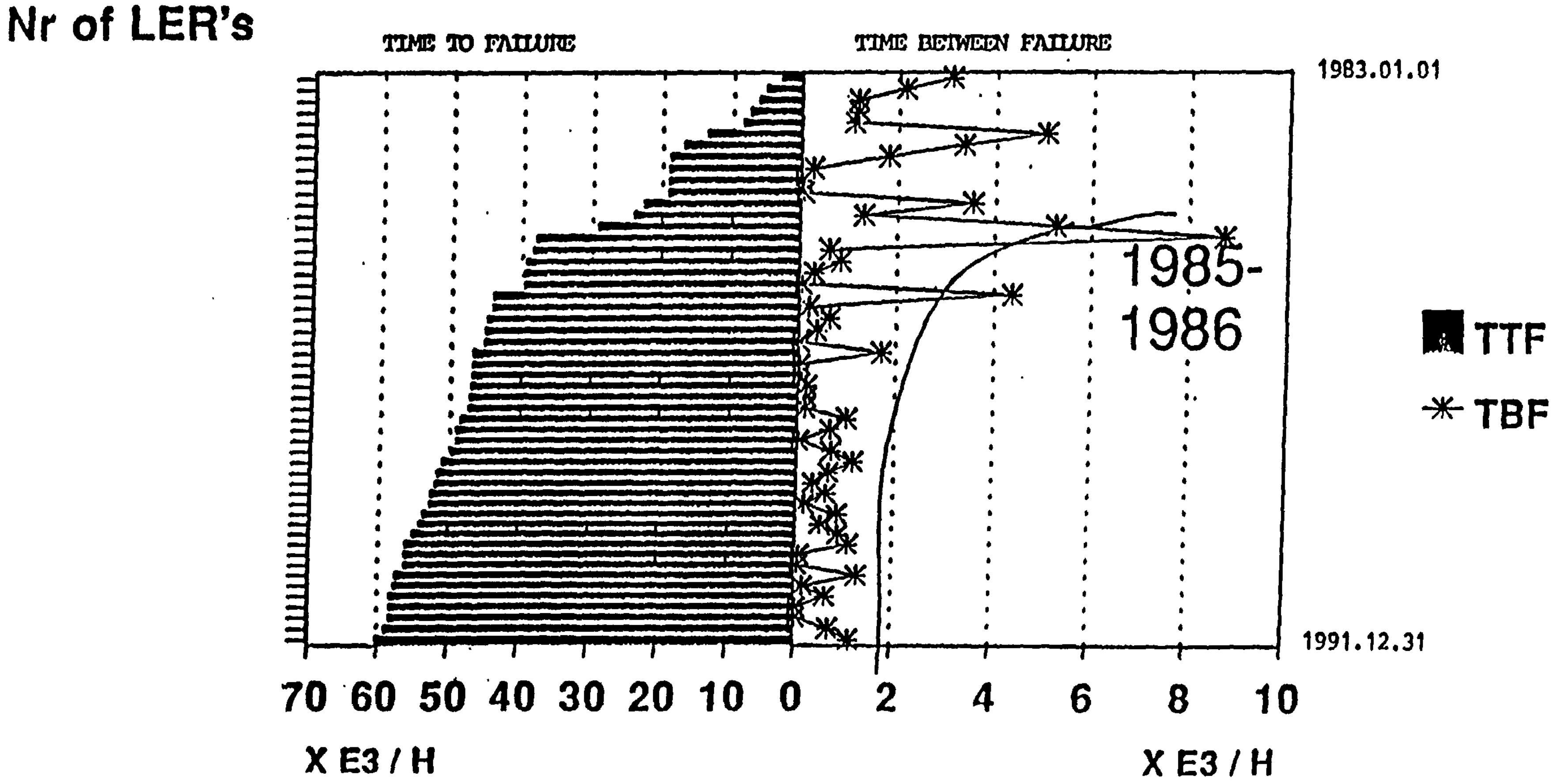

FIG. 2. Unit X, TTF vs. TBF - Tech. Spec. area 3.03; reactivity control.

Frequencies of occurrences within a defined follow up area will change in a time period and it is therefore necessary with a continuation in event reporting and analysis. This in turn, creates new search patterns (key words) and time intervals for the necessary performance of trend analysis. Therefore, the incident catalogue has to be a living instrument (tool), which frequently also has to be updated.

2.3. THE SCREENING PROCESS

The trend analysis using STAGBAS2 as database and tool, looks into the number of occurrences, e.g. per year, of events corresponding to a certain key word combination. The aim is to look into the data and make transparent any deviations from the normal or from a reference set of data. The screening can be based on the following criteria:

- Unacceptable number of incidents per unit of time.
- Heavy increasing number of incidents.
- Heavy decreasing number of incidents.
- Remarkable plant to plant variations.
- Remarkable component to component variations.

As a guideline and support in the screening process of the unique specific indicators or observation areas there will be a short presentation of the results from the unit specific probabilistic safety assessment (PSA) in the incident catalogue (see Fig. 3). The presentation of PSA results is given as risk importance for safety systems, components, unit functions, etc. Functions, systems and components which are considered weak from a safety point of view and at the same time indicate a high number of occurrences or an increasing trend have to be given priority.

In the incident catalogue, SKI gives the reader some interesting service in form of calculated probabilities for the most interesting trends which indicate an increasing or decreasing trend sloop, see Table 1. The probability numbers calculated give the reader possibilities to rank the trends/indicators also by theoretical tools and just not only a visual comparison of all results. Theoretical tools for trend analysis are described in chapter 3.

2.4. THE TREND CATALOGUE

Search patterns screened out from the incident catalogue, (by different standard types of criteria) will be stored in the "trend catalogue" for more detailed analysis. Where possible, trend analysis results based on data from other databases can be used to strengthen the validity of our own trend analysis results. The retrospective analysis of the database also implies quality assurance of the database content and verification of the coverage.

Search patterns that have gone through the above follow up analysis and thereby have shown an evident failure trend or patterns in conflict with safe operation, are being scrutinized for a longer period of time.

The analyses that are carried out in the trend catalogue are done with the aim to identify similar trends between a number of events within a defined group. Examples of such cases can be certain types of ageing phenomena or failure affected component types or components identities.

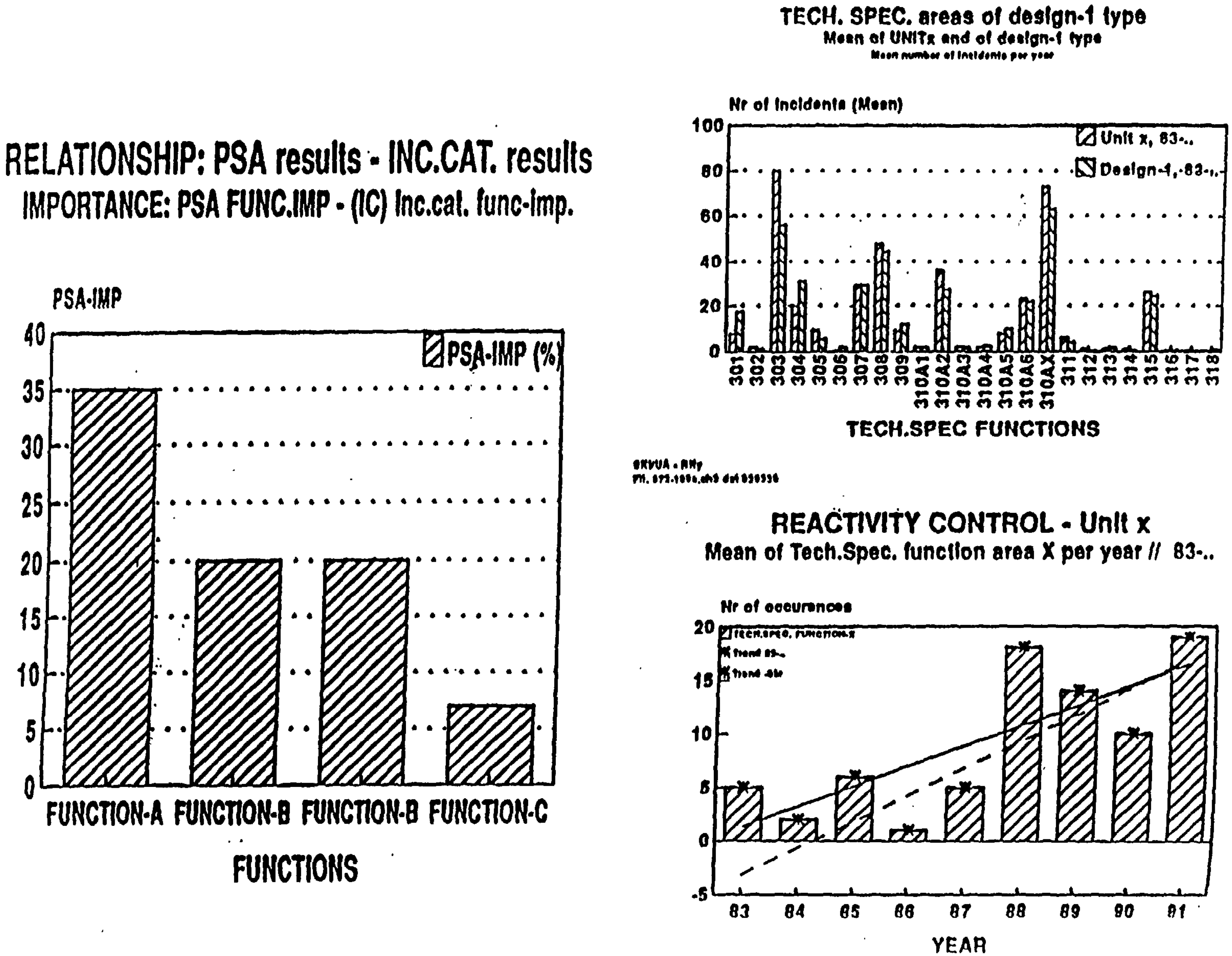

FIG. 3. Relationship: PSA results - INC.CAT. results.

A useful application of the incident catalogues and trend catalogues is that the analysis results will be used in the daily safety work and in the experience feedback work at SKI and at the NPPs. The PSA work will also from now on have the ability to use tools created in the incident catalogue work, to transfer PSA results in an easier way to the experience feedback programmes. Another practical application of the catalogues is that the SKI inspectors can use the information in the inspection work.

3. THEORETICAL TREND ANALYSIS MODELS

3.1. THE ENHPP CODE/MODEL

One of the methods to be used for trend analysis in STAGBAS2 is the ENHPP code. The ENHPP code (Estimation of Non-Homogeneous Poisson Process) is developed by VTT, the Technical Research Center in Helsinki, for statistical purposes of non-homogeneous Poisson processes.

An exact reliability analysis for complex reparable systems/components is often quite complicated due to the complicated failure process. The ENHPP code approximates the stochastic model to a simplified model not yet not exact, but still exact enough to provide practical results. Such a simplified mathematical theory is, e.g. to suppose that the time to failures for a complex system follows a non-homogeneous poisson process. The ENHPP code assumes that a repair results in a condition of the system/component that is identical to the status just before the failure occurred. This can be said to be "as bad as an old" systems/components. The ENHPP code has been used mainly for ranking purposes of the increasing trend identified. The ENHPP code also calculates the probability for increasing or decreasing trends. Examples are shown in Table 1.

3.2. TREND CODE - TREND WARE

Another model for trend analysis is now being developed by Studsvik Ltd. The trend model, TREND WARE, has been implemented in the computer program TSEBA, developed by Studsvik. TSEBA was initially developed to calculate failure intensities for components in the Swedish nuclear power industry.

3.3. MULTIPLE CORRESPONDENCE ANALYSIS

Correspondence analysis is a technique to study multi-variate data. A graphical presentation of analysis results visualizes the data properties and helps the iteration. In order to examine the associations among categories of several variables, the categories are presented as points plotted in a two dimensional space. This two dimensional space is selected so that it represents best the n-dimensional cloud. The positions of the category points in the plot reveal the feature of the data. Here we present a small example to illustrate the correspondence analysis. Let us assume that we have obtained failure data from three plants during four years, as indicated in Table 2.

TABLE 1. ENHPP ANALYSIS IN INCIDENT CATALOGUE

		UNIT X			ENHPP BWR	
Indicator	Search pattern	Betafactor Unit x	P(B)>1	Rang X	Betafactor BWR	P(B)>1
Transmitters Transformators Pumps Busbars	RO+Unit X+K*+ÅR RO+Unit X+L*+ÅR RO+Unit X+P*+ÅR RO+Unit X+S*+ÅR	2.51 0.87 1.75 1.19	0.98 0.31 0.95 0.71	2 14 6 11	- 0.93 1.65 1.33	- 0.32 0.99 0.98
Valves	RO+Unit X+V*+ÅR	0.86	0.19	16	-	-
Pressure relief system Aux. feedwater system	RO+Unit X+314+ÅR RO+Unit X+327+ÅR	0.59 0.76	0.01 0.29	22 17	- -	- -
Aux. power system	RO+Unit X+651+ÅR	1.20	0.65	10	-	-
Computer system Ventilation system Fire protection system	RO+Unit X+655+ÅR RO+Unit X+747+ÅR RO+Unit X+762+ÅR	0.63 0.70 1.18	0.02 0.15 0.69	20 18 12	- - -	- - -

TABLE 2. EXAMPLE FOR ENHPP ANALYSIS

	Year 1	Year 2	Year 3	Year 4
Plant 1	0	2	3	4
Plant 2	0	3	0	0
Plant 3	2	2	1	3

In the two dimensional plot obtained by the correspondence analysis, which represents the above data, we have two sets of points, one representing the plants and the other one the years. Each point in the plot is a display of the complete profile of the corresponding category. The main rules in the interpretation of the plot are the following: the distance between the category points within a variable is vital (the closer the points are to each other, the more similar profiles). In our example, the point representing year 2 and plant 2 lay in the same direction indicating association of these two groups. Simply said, the correspondence analysis compares the different distributions in the data set and measures how strong the coupling between them is.

4. CONCLUSIONS

An effective way of retrieval of information in a database with the properties described above, as well as meaningful reliable operating experience feedback and trend analyses require:

- A reliable and long living (stable) definition of what shall be considered as a "reportable event". Changes in the definition must be carefully considered because of its direct effect on the reporting frequency, and the indirect effect on the trend analysis results.

- The quality is in most cases heavily dependent on the initial documentation of the incident. Succeeding checks of the degree of completeness and quality can be added to the data content, but can never completely compensate for a badly documented event.

- Well defined routines for key word assignments. This means that procedures for the classification work must be available. The procedures describe all description areas. For each description area, allowed key words are given. Also the key words are defined. The chosen description areas should cover current and future demands. The choice of description areas is also governed by what is considered practically possible to classify. Well defined application areas for the database and the accompanying requirement specification will simplify all work with the database. The classification must be simple and not too time consuming. The requirement of a simple and fast classification can be accomplished by using classification flow diagrams and other types of quick reference guides. The classification work should be performed by a limited number of persons.

- A well organized and motivated organization for experience feedback where all parties can take part or have access to the result from the information exchange.

- Administrating the database structure and maintaining it requires a lot of work.

The continuous experience feedback on an everyday basis is of great importance for the plant personnel, for the authority, and for other interested parties in the nuclear power community. The safety work can thereby be performed with ambitious goals on preventing accidents and other disturbances.

The indicator system at SKI is based on licence event reports. However, it is only one of many channels available for experience feedback. And as complete as possible understanding can only be achieved by using as many inputs as possible for creating a safety indicator system.

Annex XII
EXPERIENCE FROM SCREENING AND ANALYSIS OF SAFETY RELATED EVENTS AT SWEDISH NUCLEAR POWER PLANTS

1. INTRODUCTION

The feedback of experience is one of the important issues which has always been of interest to people concerned with nuclear safety. The importance of this issue has been made more and more evident with the progress of nuclear power technology.

The Swedish Nuclear Power Inspectorate (SKI) stated, in 1974, a national reporting system for safety relevant events and reactor trips. This system gave a basis for statistics which could be used in the review of safety, but no systematic analyses were performed. Certain spectacular events were, of course, analysed, but the systematic search for initiating events and subsequent event sequences, until then overlooked, was conspicuous by its absence as very few people were thinking along this line at the time. The Rasmussen Report was issued in 1974 and gave emphasis to transients, encouraging a new approach which departed from the previous focus on pipe breaks. However, the results of the Rasmussen Report were still not in effect when the Three Mile Island (TMI) accident occurred, and accident which, in some ways, confirmed the Rasmussen Report's findings. The fact that the Davis Besse precursor to TMI had been missed by the feedback system contributed to increased interest in operating experience.

The emphasis in safety analysis was not put on transients. Human factors were stressed. Experience feedback was emphasized even more than before. All over the world different bodies - authorities, utilities and vendors - adapted themselves to the new situation, as did SKI. After the TMI accident, SKI was reorganized. Probabilistic and human factors experts were appointed on the permanent staff and the organization of SKI was changed to meet these new needs. The idea from the beginning was that analyses and experience feedback should be accomplished within the probabilistic safety analysis group. This idea did not appear to be effective. Thus, it was decided that these tasks should be handled on a project basis. The ASK (Analys av Stöningar i Kärnkraftanläggningar, i.e. Analysis of Disturbances in Nuclear Power Plants) was developed as a permanent activity supporting the head of the Office of Regulation.

The processes within SKI and the ASK activity are shown in a flow diagram (Figure 1). The diagram shows how the reports on events arrive at different departments within SKI, how the information is screened and handled by different groups, how a decision is reached and how the final decision is made known to the utilities, with a request for a written answer. The feedback process is complete when the utility involved has responded to SKI and other concerned utilities are notified.

This process has three stages which especially difficult: screening, analysis and implementation of analysis results. They will be dealt with in turn below.

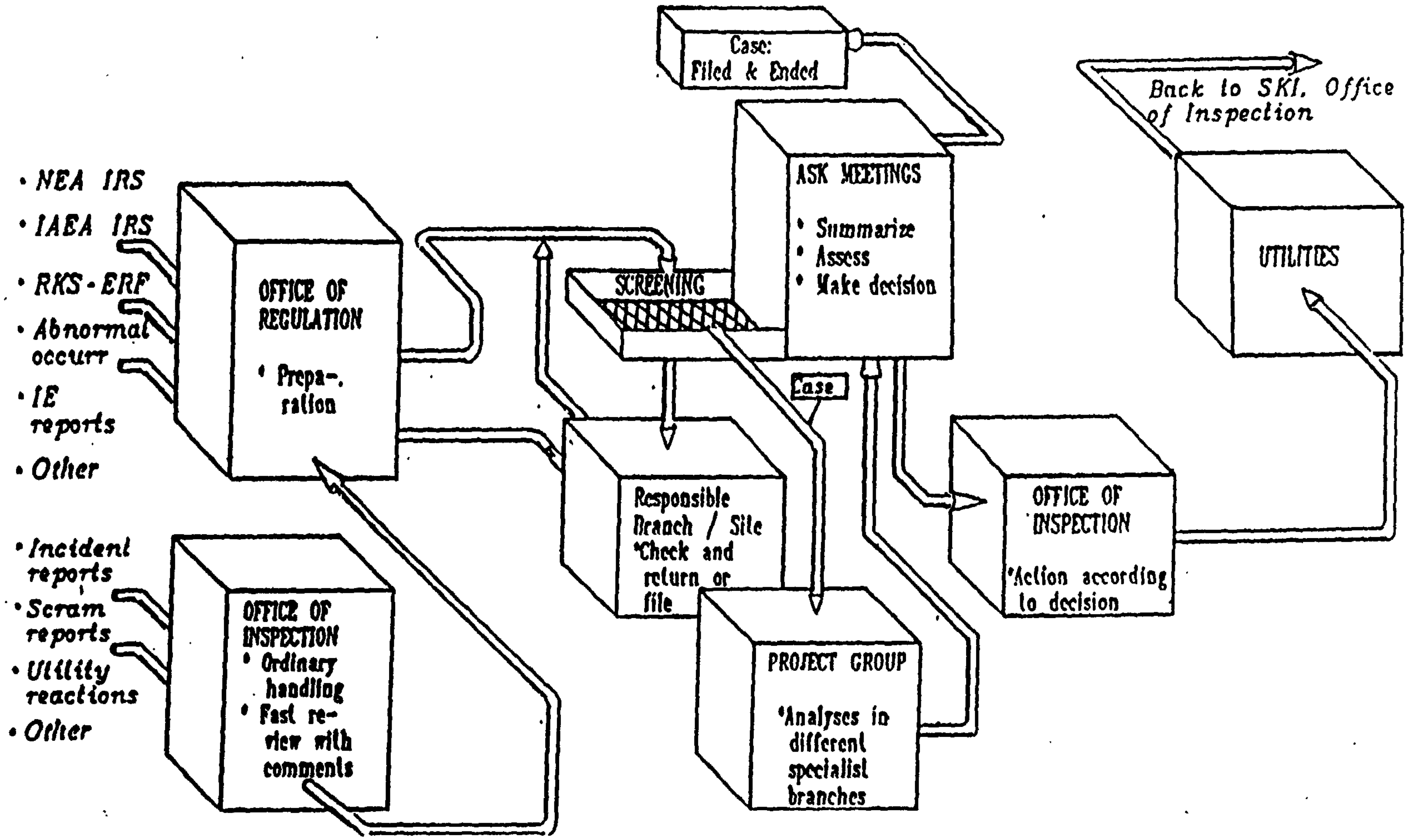

FIG. 1. The feedback process within SKI, including the analysis of disturbances in nuclear power plants - ASK (RKS: Nuclear Safety Board of the Swedish utilities; ERF: feedback of operating experience; IE: internal experience).

2. SCREENING

2.1. WHICH EVENTS SHOULD BE ANALYSED?

The safety significance of domestic reported events could vary from the more or less negligible to the very serious, as the reporting criteria of the technical specifications are wide ranging, covering everything from a failing component at recurrent testing to a serious transient. The safety significance of reported events of foreign origin, e.g. Nuclear Energy Agency of the OECD Incident Reporting System (NEA-IRS) reports, is somewhat different. Those reports have already been screened by the reporting country and are, from their point of view, of safety significance.

Thus, it is essential to have a screening process which can detect those events that are important enough to be deal with at SKI. This is a delicate task, especially with reports of domestic origin, because while in some cases the importance of events is quite obvious, in others it is 'hidden' in the wording of the report. For reports of foreign origin, the problem is different, as the screening must be able to determine whether or not the problems in the reported events are applicable to Swedish reactors and, if they are, whether they are relevant to a specific reactor or are generic in nature.

There are different ways to make such a screening procedure effective and reliable. When studying how other organizations act it is revealed that procedures are by no means standardized. The specific organizational structure is often crucial.

At SKI, which has a project organization to handle the procedures, the resources are not large enough to be able to designate a certain person to carry out the screening. It is also difficult to find one person who can cover all of the disciplines necessary to understand the different aspects involved. Thus, a screening group has been formed consisting of five experienced people representing five different disciplines: human factors, probabilistic safety analysis, systems, analysis, core physics and inspection. Every Monday morning the group has a short meeting (not more than one hour) where reported events from the preceding week are handled. On an average, about ten Swedish and three foreign reports are discussed every week.

The handling procedure for domestic reports is a little different from that used for foreign reports. Domestic events are reported according to a special format which was developed within SKI, in co-operation with the utilities, in 1974 and is still in use. An example of such a report is given in Figure 2 and contains information on the event discussed in Section 3.1. It should be mentioned here that this reporting format is old fashioned and is not able to adequately cope with the new reporting demands which have arisen during the 1980s, especially in the man-machine area. Efforts are under way for a change to a more up to date version.

The contents of the domestic report are read aloud to the group by one appointed member of the group. Preparation is not necessarily demanded before hand. Most reports are of no interest for further analysis, but may have special interest for certain branches within SKI. They are therefore transmitted to these interested branches. If such a group feels that the screening group has made a mistake, it has the opportunity to return the report for another screening. However, the probability that the screening group should make a mistake is small, owing to the expertise of the group and the pluralism represented by five people. The experience gained up to now is also encouraging.

O S K A R S H A M N S V E R K E T   B L O C K 2

Distribution

[ ] Preliminary

[x] Final report no: 2-86/87

[ ] Abnormal occurrence

[x] page 2-enclosed

[x] Reportable event according to Tech. Spec. 5.5.8.2

[ ] Final report according to Tech. Spec.. section 5.5 encl.

System: 666   Component: MG 221

Title: MG221 tripped on overfrequency

Date of event: 24 JAN 1987   Time: 02.36 am

Reported by: Lars Bennarsten   Date: 26 JAN 1987

MODE OF OPERATION	WAY OF DISCOVERY	SYMPTOM
[ ] Cold shutdown	[x] Cntrl room superv.	[x] Alarm in cntrl room
[ ] Refuelling	[ ] Operator round	[ ] Malfunc., elec sys.
[ ] Hot shutdown	[ ] Function test	[ ] Malfunc., mech sys.
[ ] Nuclear heating	[ ] Repair	[ ] No function
[ ] Hot standby	[ ] Planned mainten.	[ ] Spurious function
[x] Power operation	[ ] Annual outage, rev	[ ] Damaged el equipment
[ ] House load turb op	[ ] Special inspection	[ ] Damaged mech equipm.
	[ ] By chance	[ ] Vibrations
	[ ] Other	[ ] Leak
Thermal power % 106		[ ] Smoke, fire
		[ ] Noise, smell
MW el. 620		[ ] Other symptom

EFFECT ON OPERATION	EFFECT ON EQUIPMENT	COMPONENT
[x] Auto reactor trip	[ ] System out of oper.	[ ] Pressure vessel
[ ] Auto " shut down	[x] System func reduced	[ ] Heat exchanger
[ ] Manual reac. trip	[ ] Component out of op	[ ] Pipe
[ ] Isolation	[ ] Component func red.	[ ] Flanged coupling
[ ] Turbine trip	[ ] Damage, other part	[ ] Valve, incl. pos.
[ ] Dump blocking	[ ] No consequence	[ ] Pump, fan
[ ] Load reduction	Other Consequence	[ ] Motor, generator
[ ] To house turb load		[ ] Regulating equipm.
[ ] No influence on op		[ ] Switchgear equipm.
		[ ] Cable
		[x] Other

ACTION TAKEN/PLANNED	CAUSE	POSSIBLE CAUSE BEHIND
[x] Exchange of part(s within reported ob	[ ] Corrosion, erosion	[ ] Design
[ ] Exchange of reported object	[ ] Abnormal wear	[ ] Material
[ ] Repair of parts	[ ] Unbalance, fatigue	[ ] Manufacture
[ ] Adjustment, calib.	[ ] Water hammer	[ ] Install./Construction
[ ] Cleaning, lubric.	[ ] Deform., displacem.	[ ] Maintenance
[ ] No action on comp.	[ ] Crack	[ ] Oper. procedures
[ ] Other action	[ ] Break	[ ] Oper. error
	[ ] Fire, explosion	[ ] Incorr. water chem.
	[ ] Earth fault (elec)	[x] Other possible cause
	[ ] Short circuit	
	[ ] Loss of voltage	
	[x] Other cause	

FIG. 2. Event report form (translated from Swedish).

The foreign reports are handled somewhat differently because, first, they are already screened from the point of view of safety and, second, they represent events which have already been analysed by the country of origin, resulting in a more or less comprehensive report. For this reason, they are not read aloud to the members of the screening group, but are instead prepared in advance. The procedure is that two people from the screening group read the reports independently before hand and one of them then briefly presents the contents to the group with the purpose of clarifying their possible relevance for Swedish reactors. The handling procedure thereafter is the same as for domestic events.

What criteria are used to select the important events and what are the thresholds? The fact is that no explicit criteria exist as yet. Instead, 'implicit' criteria in the minds of the five experts are utilized. When an interesting event occurs, the reported facts are discussed in detail and complementary information is collected from final safety analysis reports, flow diagrams, etc. A consensus is then reached on whether or not the event should be subjected to deeper analysis. It should be noted that the decision making process of the screening group as a whole is different from that for the same persons individually. The threshold is determined by the combined knowledge of the group members, reflecting current SKI opinion on reactor safety.

3. ANALYSIS

Analyses are, of course, carried out regularly by the different SKI departments according to their fields of specialization. However, also dealt with here is a process which tries to integrate aspects from different specialists in order to reach a final conclusion with respect to a certain event. The general procedure for carrying out the analysis is given below.

When an event report has been selected by the screening group for further analysis, the different aspects crucial for the decision are listed. This forms the basis for choosing the members of the project group responsible for the analysis of the event. This ad hoc group consists generally of five people, to a maximum of ten persons - a chairman, a technical secretary and the specialists from the responsible departments. The department head is of course responsible for the work being done, but mostly he delegates the work. The analyses are carried out independently, reports are written and finally presented at a project meeting by spokesmen according to their different specialist qualifications. The findings are extracted, listed, synthesized and discussed by the members of the project group.

The objectives, with a heterogeneous group such as this, are diverse. One is that persons completely independent of the kind of analysis being presented are better able to detect 'residual' items. Another is that a presentation of an event from the perspective of someone else's area of competence can reveal new insights which can very well generate ideas or even solutions to problems within other specialists areas. A third is that consultation by people from different departments within the regulatory body creates an understanding of the importance of their respective areas of responsibility. The difficulties with a group of members taken from a hierarchial organization are obvious, as the 'normal' tasks generally are assigned priorities. It is essential to clarify the objectives of the ASK activities and make clear the obligations of the members of the group in an instruction sanctioned by the Director General.

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EVENT SEQUENCE AND CONSEQUENCES:

The frequency on 666 MG 221 had slowly risen from 50.0 Hz to 50.2 Hz over a period of about 12 hours, after which it suddenly rose to 53 Hz. The rotary converter thereby tripped, causing a loss of voltage on the AC busbars 664 G04 and 665 C02 lasting 1.5 seconds until automatic changeover to feed from busbar 663 E04 had taken place. In addition to the reactor trip, the power supply to three of the process computers pcts was cut off. All event during 1.6 minutes of the introductory phase of the disturbance sequence were lost. As a result of the brief loss of voltage on G04 and C02, two HC pumps stopped and the main governor was deenergized, leading to reactor trip on SS5, high level in reactor vessel.

CAUSE:

Fault tracing of 666 MG 221 has not provided any concrete evidence as to the cause of the trip. The sequence points to a fault in the tachometer-generator or the rotary converters speed governor. One possible explanation is that oxide has formed on connection pins to the governors circuit board. A similar fault has previously been observed for other governors, but cannot be verified in this case.

ACTIONS TAKEN OR PLANNED:

The following measures have been adopted:

A. the tachometer generator and a circuit board in the converters speed governor have been replaced.
B. Cleaning of carbon brushes and holders from carbon dust.
C. Check of components included in starter.
D. The alarm limit for high frequency has been lowered from 51.0 Hz to 50.4 Hz.
E. Trial operation of rotary converter under no-load and load.

FIG. 3. Reproduction of a Swedish licensee event report (LER) form; description attached to LER (translated from Swedish).

Often there has to be more than one project meeting in order to come to a mutually agreed upon position. It is felt to be of importance to work until a consensus has been reached. Unfortunately, this means that the analysis of one single event might take considerable time before the point is reached where the case can be closed. This implies that several events are often handled during the same time period.

After a consensus has been reached among the members of the project group, a final paper is written which summarizes the findings and conclusions. This also forms the basis for a formal letter which is sent, according to normal SKI procedures, to the utility in question with a request for a written answer (see Section 4).

3.1. EVENT AT OSKARSHAMN UNIT 2

The following example is based on the event at Oskarshamn Unit 2 in February 1987 (Fig. 3).

3.1.1. Description of the event as reported by the utility

Oskarshamn Unit 2 had been running without any disruption at full power for five months. At 02:36 a.m., on 24 February 1987, the following happened. The frequency on a motor generator of the rotary converter in the battery backed AC system had slowly increased from 50.0 to 50.2 Hz over a period of about 12 hours, after which it suddenly rose to 53 HZ. The rotary converter thereby tripped, causing a loss of voltage on two 400 V AC busbars lasting 1.5 seconds, until automatic changeover to feed from one of the diesel backed busbars had taken place. In addition to the reactor trip, the power supply to three of the six process communication terminals to the process were cut off. All information during the 1.6 minutes of the introductory phase of the disturbance sequence was lost. As a result of the brief loss of voltage on the busbars, two main circulation pumps stopped and the main governor was de-energized, leading to a reactor trip due to high water level in the reactor vessels.

Fault tracing of the converter did not provide any concrete evidence as to the cause of the trip. The sequence pointed to a fault in the tachometer generator or the rotary converter's speed governor.

The following measures were adopted: (1) replacement of the tachometer generator and a circuit board in the speed governor; (2) cleaning of the carbon brushes and holders; (3) check of components included in the starter; (4) lowering the high frequency alarm limit from 51.0 to 50.4 Hz; (5) trial operation of the rotary converter under no load and load conditions.

3.1.2. Actions using ASK

As has been mentioned above, the screening group decided to make a special case of this event. The reason for this was that the event was rather complicated and affected safety relevant systems. Additionally, the cause and the effects were unclear. Following procedures, a project group was appointed representing the different departments within SKI which might have an influence on the evaluation of the event. The chairman formulated the task of the project group, reflecting the thoughts of the screening group, and distributed a message on a standardized format to the selected departments. In this particular case, the following were asked for:

(1) an analysis of the sequence;
(2) whether other disturbances with the same initiating event had occurred;
(3) whether there were any man-machine interactions;
(4) a trend analysis.

The following departments were selected: Reactor Technology, Probabilistic Analysis, Human Factors and Inspection.

About one mont after the formation of the project group, the first project meeting was held. Each participant presented a report based partly on the initial information and partly on any new information which the different departments were able to collect. After discussion, it was concluded that the utility should provide a report on the safety implications of the loss of the busbars in question. The matter was postponed until the response had been received.

In November 1987, the next project meeting took place. The project members presented their reports with the following conclusions:

(1) The unit is designed not to trip because of this type of disturbance. However, this was obviously not true for the current approved maximum power level, 106% of nominal (Inspection).

(2) The event was by no means unpredicted or unique and, as such, was not considered to have any major safety significance. All Swedish utilities were well aware of the problems with the rotary converters and had already shown an acceptable degree of interest in solving them (Reactor Technology).

(3) The process computer had no immediate safety role, but its logging function is sometimes of vital interest. Therefore, an improvement of the computer's power supply was considered to be worth further discussion. Conventional instrumentation should not be removed when implementing computer logging of the same parameters (Reactor Technology).

(4) Common cause initiating events made this event of special interest calling for future analysis of the reliability of the auxiliary power supply (Probabilistic Analysis).

(5) For the purpose of understanding human behaviour in these types of circumstances, it was thought that it would be interesting to know more about reactions in the control room. What type of information was needed? Was there any problem in maintaining the role assignments? (Human Factors).

The conclusions, generally agreed upon, were as follows:

(a) the short (1.5 seconds) power loss did not result in any damage or cause any unstable or unpredictable condition. Therefore, the event did not have any significant impact on reactor safety;
(b) generally, disturbances involving man-machine interactions should be studied immediately as people tend to forget or distort information;
(c) A-and-B trains should not be mixed in the same component or location.

4. IMPLEMENTATION OF ANALYSIS RESULTS

The feedback process is not complete until the reactor plant management has received the result from the analysis process described above and has responded to possible questions or demands from SKI. The project ends when the plant management has responded and the proposed actions, whatever they are, have been approved by SKI. How these actions are accomplished is covered by the usual regulatory procedures. This is natural, considering that some actions, such as further investigations, hardware replacement or improvement, might very well require quite a long period of time. It is possible that the corrective actions or improvements suggested as a result of the experience gained from an analysed event (domestic or foreign) are applicable to several units. This is also taken care of in the analysis process.

In order to be able to retrieve useful information from reported events with similar characteristics, SKI has developed two computer databases, STAGBAS and ASKBAS. In the first, reports from incidents at Swedish units are stored and in the second are stored reports from foreign reactors. A recently installed local area network (LAN) will give SKI personnel a convenient method of retrieving data from the databases.

The analysis process, now being practiced for one and a half years, has resulted in spin-off effects which will have an impact on other in-house activities. Some also lead to R&D projects. For example, a project has been suggested for developing a database which would assist in giving systems safety parameters and other essential plant data when assessing the importance of an event at a particular nuclear power plant (NPP) worldwide. Other spin-offs are development of tailored analysis methods, improved understanding of human errors, or a programme for enhancing inspection and review procedures and enhancing knowledge regarding administrative functions at the plants. The increased need for updated and selected information during screening and analysing has also speeded up the installation of the LAN mentioned above.

5. SUMMARY

Improving safety by collecting and disseminating information from safety related events, both from Swedish and foreign NPPs, has been practiced at SKI since at least 1974. Analyses have been carried out and utilized for different isolated purposes. However, not until recently has a more interdisciplinary oriented and systematic effort been possible. Both the screening group and the different analysis project groups consist of persons from different departments and with different skills.

Some of the experience derived from this work:

(1) it is important for there to be a continuous screening effort (once a week);
(2) what can be achieved by a group of experts from different fields is greater than what can be achieved by the same persons working as individuals;
(3) this kind of work demands considerable time and effort;
(4) as a spin-off, R&D projects are identified and unresolved safety issues are assigned a priority;
(5) in order to obtain useful results, the allocation of resources for an analysis group must be guaranteed;

(6) in the light of the experience from the analysis work, a need for a human factor oriented way of reporting, as a complement to the technical view, has been identified;
(7) to make it possible for less experienced people to carry out the screening effort (if there is a 'problem of resources'), the existing implicit criteria must be stated explicitly;
(8) training in analysis methodology is needed.

Annex XIII
INCIDENT INVESTIGATION METHODS AND PRACTICES IN FINLAND

1. REPORTING OF EVENTS

The supervision and assessment of events at nuclear power plants is an essential part of the regulatory control of the operation of nuclear power plants. The requirements applicable to the contents and submission of operational reports are given in regulatory guide YVL 1.5 which prescribes that a report be submitted in the event of, i.e.

- significant for safety (special reports of which examples are given in the YVL Guide);
- reactor trips; and
- other operational transients.

Daily and monthly reports on operational events and component failures shall be submitted. All significant events (and significant operational transients) shall be reported also by telephone during office hours and they shall be included in the next daily report. Events with a bearing on safety shall be reported even outside office hours. The procedures for alerting the regulatory body (STUK) in accidents are described in the emergency preparedness plans of the Finnish utilities.

2. INVESTIGATION OF OPERATIONAL EVENTS

The utility has always the primary responsibility to investigate incidents that have taken place at their plants. In some cases also the regulatory body makes an independent investigation of the incident.

When information of an operational event has reached STUK, the management of the Department of Nuclear Safety decides upon the assignment of an investigation team. The team may be assigned immediately after the event or later at a departmental meeting, on the basis of proposals made by the Offices. The head of the team of investigators is usually assigned from the Incident Evaluation and Reporting Section. The rest of the team members are assigned on the basis of the incident.

An investigation team will be assigned in particular when a utility's own organization is assessed not to have functioned as planned in connection with an event, and also when the event is assessed to lead to significant modifications in the plant's technical structure or to amendments to plant instructions. A team of investigators is set up every time an incident is assessed at level 1 or higher on the INES scale. A team may be set up later also in cases when the investigation made by the utility is found unsatisfactory.

The investigation team visits the plant site to establish details of the incident and to review the operational report and other reports on the incident drawn up by the utility. In connection with the investigation, the team assesses the root causes of the incident and fills in a data collection form which is fed into a computer register. During incident investigation, the following points in particular will be paid attention to:

- correspondence to earlier events and actions thereupon;
- the root cause of the incident;
- the safety significance of the incident;
- immediate and long-term actions of the utility design deficiencies;

- deficiencies in the operation and maintenance of the plant (human error, organizational weaknesses);
- effect on regulatory control practices.

The incident investigation team prepares a report which includes the team's recommendations. A filled-in incident data form is always attached to the report. In this form special emphasis is laid on human factors issues and on the performance of the utility's organization.

The operational event registration form is developed within STUK but it contains features of several root cause analysis methods well known worldwide; i.e. the Swedish MTO and the Root Cause Analysis System of Savannah River Plant. The MTO method is a Swedish application of the HPES method (Human Performance Enhancement System) developed by INPO.

Cause types which have led to an incident are specified in the form for entry in a computer file. There are five different causes to choose from: a design, manufacturing, operational and maintenance error plus inefficient organizational activity. For each incident, the causes are classified more specifically.

Root causes are reported in text format. The maximum number of root causes in the form is three. It is characteristic of a root cause that:

- an incident is directly attributable to it;
- the incident would not have occurred without the root cause in question;
- the root cause can be eliminated by practicable means.

The root cause can often be identified by repeating the question "why" until an obvious cause-and-effect relationship disappears or until the elimination by practicable means of the apparent root cause is not possible/reasonable. Therefore, e.g. laws of nature, their consequences or natural phenomena are not root causes.

In preparing the form, practices in different countries and in different organizations were studied. The form is unique, through; it contains several questions concerning human error and organizational issues. In the part concerning human error, especially cognitive errors made by operators are identified. Also procedural errors, errors related to ergonomic design of the control room and various factors influencing the working environment are identified.

Co-operational deficiencies between units, erroneous evaluation of safety significance, conflicting objectives, inefficient allocation of resources, incorrect assignment of responsibilities, etc. are identified in the part concerning organizational issues.

A so called barrier analysis shall be presented on the form. The filler-in shall report the systems and methods which serve to prevent the incident and he shall assess their adequacy and efficiency during the incident in question. A barrier means all methods and physical arrangements which should have prevented the incident or which should have mitigated its consequences.

The head of the team of investigators at a departmental meeting presents the investigation reports containing the team's proposals for the development of STUK's regulatory inspection function.

3. THE OPERATIONAL EVENTS REGISTER

The following types of incident will be fed into the register of operational events:

(1) incidents at level 1 or higher on the INES scale;
(2) incidents for which a special report has been drawn up;
(3) operational transients with related deficiencies in the functioning of the utility's organization;
(4) operational transients which have brought about significant changes in plant structure or in procedures;
(5) incidents with related multiple component failures in one or several systems.

The whole of the Department of Nuclear Safety's personnel is entitled to read the operational event register. Individual reports and various summaries can be searched, scanned and printed from the register. The Incident Evaluation and Reporting Section is entitled to input data into the register.

4. INCIDENT REPORTING SYSTEM (IRS)

To assess operational experience gained at nuclear power plants abroad, an operational experience assessment team (KKR) is in action within STUK.

KKR assesses NEA and IRS reports and, based on the assessments, disseminates these reports for detailed review and clarification.

When a report has arrived in STUK, the distributor first assesses the scope necessary for the review and sends the reports to appropriate KKR members for review, on the basis of their field of expertise.

An inspection report is drawn up of every report which has been submitted for detailed review; the inspection report contains, i.e. proposals on the actions possibly to be taken. The line organization of STUK reviews the proposals and is responsible for further action. By means of a checklist, KKR follows up the actions to be made on the basis of the reports.

KKR meets at determined intervals (a few times annually) to discuss new events, the review of reports and the actions based on the events to be followed up.

5. FOLLOW-UP OF OPERATIONAL EVENTS BY UTILITIES

Both Finnish nuclear utilities, Imatran Voima Oy (IVO) and Teollisuuden Voima Oy (TVO) have established procedures for the follow up of domestic and foreign operational events.

It is important that the utilities profoundly analyse operational events at their plants and carry out the necessary corrective actions. It is also vital that the lessons learned from operational events abroad will be utilized as well as possible.

The line organization of IVO investigates all operational events at the utility's plants. A group has been set up to analyse events abroad. IVO carries out root cause analyses of significant events at own plant units by a method developed by IVO.

There is a team at TVO which looks into operational events at own plants and abroad. Root cause analyses of significant events are made by the MTO method.

Both utilities submit their root cause analyses to STUK.

Annex XIV
OLDBURY-ON-SEVERN POWER STATION (UK)
MANAGEMENT CONTROL PROCEDURE
OPERATIONAL EXPERIENCE FEEDBACK

1. PURPOSE

The purpose of this document is to:

(a) outline the process of Operational Experience Feedback (OEF) at Oldbury-on-Severn power station;
(b) identify the station procedures for OEF;
(c) identify the responsibilities on the station for OEF.

The purpose of the OEF system is detailed in a policy document, Board Procedure on Operational Experience Feedback (BP/NCG/001).

2. SCOPE

2.1. This Management Control Procedure (MCP) applies to all CEGB staff working at Oldbury-on-Severn power station.

2.2. OEF covers three areas of activity:
  (a) events occurring on-site;
  (b) events occurring off-site;
  (c) on-site safety systems unavailability monitoring.

2.3. This MCP addresses the processes, interdepartmental interfaces and responsibilities involved in these three basic areas.

2.4. The process of OEF on-site includes:
  (a) event detection and reporting;
  (b) screening for significance to Oldbury;
  (c) analysis of event reports;
  (d) reporting of analyses and recommended remedial actions;
  (e) acceptance and tracking of actions resulting from recommendations.

2.5. This MCP does not cover Plant Status Reports (see OLD/MCP/12 - Establishment of the Operational Status of Plant and Apparatus).

3. RESPONSIBILITIES

3.1. ENGINEERING MANAGER

The engineering manager shall ensure the effective operation of the station's OEF system. This responsibility includes:

  (a) ensuring that on-site events are adequately investigated to identify root causes and corrective actions;
  (b) ensuring that a system is in place to collect and analyse plant data for use in performing on-site event investigations;

(c) co-ordinating the review and approval of recommended corrective actions with on-site review meetings, (Safety Management Meeting, Engineering Management Meeting and Production Meeting);
(d) ensuring that approved corrective actions are implemented by nominated station staff;
(e) regularly reviewing the OEF system;
(f) authorizing event reports for inclusion into the Site Incident Register and/or NUPER database, and/or other external reporting systems.

3.2. OPERATIONAL FEEDBACK ENGINEER (OFE)

The OFE is responsible for overall co-ordination and administration of the OEF system on-site.

Specific responsibilities include:

(a) screening of on-site deviations from normal operating conditions and/or practices to identify those events requiring further investigation;
(b) co-ordinating on-site event investigations with staff;
(c) ensuring affected department managers have input into the development of proposed remedial actions;
(d) reporting certain plant events (as defined in Board Procedure BP/NCG/001) to Nuclear Co-ordination Group (NCG) via the NUPER database;
(e) maintaining status tracking systems for all OEF reports processed on-site;
(f) entering all on-site events onto the station OEF database (OLDER);
(g) determining the applicability and priority of OEF reports on the NUPER system through:

- mandatory assessment of significant event reports/plant event reports identified by NCG using the appropriate station engineers as necessary;
- deferred assessment of plant event reports as identified on NUPER bulletins using appropriate station engineers as necessary;

Note: Formal points of contact for the OFE are generally at Section Head level and are detailed in Appendix A.

(h) reporting recommendations for remedial actions to the Safety Management meeting;
(i) identifying and recommending OEF items to the training engineer for inclusion in the training programmes for all appropriate station staff;
(j) monitoring assessment and remedial actions arising from OEF reports;
(k) providing periodic reports to Station Executive and Safety Management meeting on the status of OEF events on-site;
(l) gathering unavailability data for key safety related plant for inclusion in the corporate safety systems unavailability database (NUSPRA);
(m) providing appropriate staff with relevant information for dissemination within their Department/Branch/Section;
(n) periodic review of this MCP.

3.3. BRANCH HEADS, SECTION HEADS, SHIFT CHARGE ENGINEERS

Operational experience feedback responsibilities for Branch Heads, Section Heads and Shift Charge Engineers include the following:

(a) registering significant abnormal occurrences by entering them into the appropriate station log and by notifying the OFE and ensuring that they are tabled at the Morning Production Meeting as appropriate;
(b) preparing evaluations on events for mandatory or deferred assessment in conjunction with the OFE and the appropriate plant area engineer(s);
(c) supporting investigations of on-site events as required;
(d) distributing OEF information to appropriate personnel;
(e) implementing approved remedial actions.

3.4. PLANT AREA ENGINEERS

OEF responsibilities for plant area engineers include the following:

(a) preparing evaluations on events for mandatory or deferred assessments in conjunction with the OFE and appropriate shift charge engineer/section head;
(b) supporting investigations of on-site events as required;
(c) implementing remedial actions approved by the Safety Management meeting;
(d) using the OEF database to maintain a high level of awareness of OEF items that are directly or indirectly applicable to relevant plant responsibilities (operation, maintenance or modification);
(e) plant performance monitoring and trend analysis (see OLD/MCP/12) and notifying the OFE of adverse trends in plant performance likely to be of value to other sites.

3.5. EMERGENCY PREPAREDNESS ENGINEER

OEF responsibilities for the emergency preparedness engineer include the following:

(a) assessing event reports provided by the OFE;
(b) implementing remedial actions approved by the Safety Management meeting.

3.6. TRAINING ENGINEER

OEF responsibilities for the training engineer include collaborating with the OFE investigations of both on-site and off-site events to identify training needs and implementing agreed remedial measures in accordance with OLD/MCP/10.

3.7. ALL STAFF

All staff have a responsibility to contribute to the OEF process and have the following responsibilities:

(a) reporting any known or suspected deviations from normal plant operation to their branch heaD, section head or shift charge engineer;
(b) reporting of near misses to their section head, shift charge engineer, or station safety officer.

4. PROCEDURE

The OEF system discriminates between the two main inputs, namely:

(a) on-site event reports;
(b) off-site event reports.

4.1. RESPONSE TO ON-SITE EVENTS

Deviations from normal operating conditions and/or practices may occur daily. In most cases these will be trivial with no nuclear safety significance in their own right. However, it is important to identify personnel and plant deviations judged to have operational significance of a safety or commercial nature requiring investigation and remedial measures.

Consequently it is necessary to have a screening process based on a knowledge of all or a large proportion of, on-site deviations.

A deviation judged to be significant and therefore requiring investigation is defined as an event.

This screening process and the subsequent response is described below.

4.1.1. Detection of and response to on-site deviations

4.1.1.1. The OFE shall:

(a) monitor on-site meetings (see OLD/MCP/07), review station logs and reports, and carry out routine plant walks to identify deviations occurring on-site;

(b) screen the deviations identified for their significance and, if significant, classify them for entry into one or more of the following information systems:
- the local Oldbury event reporting database (OLDER);
- the NUPER database;
- the site incident register.

The criteria for this classification are detailed in OLD/EODI/18/001 - Reporting and recording of on-site events;

(c) obtain the engineering manager's agreement to the proposed reporting classification;

(d) provide an oral report of the event to the Safety Management meeting and/or Engineering Management meeting (OLD/MCP/07) as appropriate.

4.1.1.2. When events have an obvious immediate operational significance, they shall be reported to the Morning Production meeting. Subsequently, the Production Manager (or nominee) shall ensure timely notification in writing to all relevant staff by the Directed Reading system (OLD/MCP/03 - Document Control).

4.1.2. Recording and analysis of on-site events

The OFE shall:

(a) record the event onto the appropriate database (see 4.1.1 (b) above) as agreed with the engineering manager. The process for this is detailed in OLD/EODI/18/001;

(b) with the help of nominated engineers on-site (see Appendix A) and reports produced by other off-site departments, investigate the vent to establish direct and root causes and recommend remedial measures.

4.1.3. Presentation of on-site events to management and placement of actions

4.1.3.1. The OFE shall report the event to the Safety Management meeting for consideration and acceptance of the recommendations for remedial measures and the subsequent placement of actions. The procedure adopted by the OFE for responding to on-site events is detailed in OLD/EODI/18/002.

4.1.3.2. The following shall be recorded in the minutes of the Safety Management meeting:
(a) acceptance of recommendations and actions;
(b) nomination of a responsible engineer to undertake the actions and monitoring them through to completion;
(c) specification of a target date for completion/review of actions.

4.1.3.3. The nominated responsible engineer shall be notified by his department manager (or nominee).

4.1.4. Tracking and completion of actions

4.1.4.1. The OFE shall monitor progress of actions arising from on-site events and periodically report back to the Safety Management meeting.

4.1.4.2. Final completion of all actions shall be agreed by the department managers (or their nominees) and recorded in the minutes of the Safety Management meeting.

4.1.5. Circulation of reports arising from on-site events

4.1.5.1. The OFE shall screen on-site event reports and circulate them to Production, Engineering Resource, and Administration Departments as appropriate.

4.2. RESPONSE TO OFF-SITE EVENTS

Notification of off-site events to the power station occurs through one or more of the following channels:

(a) The NUPER database which is routinely accessible to the station.
(b) Panel of inquiry reports.
(c) Nuclear Safety Committee minutes.
(d) Other ad hoc reports, both written and verbal.

The procedure adopted by the OFE for responding to off-site events is detailed in OLD/EODI/18/003.

The station response to events reported through the above channels is detailed below.

4.2.1. Off-site reports via NUPER

NUPER is the principal source of off-site event information.

The station response is as follows:

4.2.1.1. The OFE shall:

(a) receive significant event reports and plant event reports specifically identified by Nuclear Co-ordination Group for Mandatory Assessment by the station (as specified by BP/NCG/001);
(b) place them with a nominated on-site engineer(s) (see Appendix A) for evaluation of their significance to Oldbury;
(c) with the help of the nominated engineers, recommend remedial actions to the Safety Management meeting for consideration and approval of the proposals by the department managers (or nominees) and the placement of actions.

4.2.1.2. The procedure for the placement and tracking of actions is identical to that for on-site events as detailed above (4.1.3 and 4.1.4).

4.2.1.3. Every week, the OFE shall:

(a) perform a deferred assessment of new entries onto NUPER for their specific significance to Oldbury with help from nominated engineer(s) (see Appendix A) as required;
(b) recommend remedial measures where necessary to the Safety Management meeting for consideration and acceptance of the proposals by department managers (or nominees) and the placement of actions.

4.2.1.4. The procedure for the acceptance, placement and tracking of actions is identical to that for on-site events as detailed above (4.1.3 and 4.1.4).

4.2.1.5. As an input to on-site staff training, the OFE shall extract a monthly list of new NUPER summaries and circulate it to the Production, Engineering and Resources Department managers.

4.2.1.6. Department managers shall identify those events having potential use in training their staff and notify the OFE by returning the marked up list.

4.2.1.7. The OFE shall then obtain full NUPER reports for each event identified and send them to the training engineer for circulation to the relevant staff under Directed Reading System (OLD/MCP/03 - Document Control).

4.2.1.8. The OFE shall circulate a quarterly bulletin of NUPER summaries, provided by NCG (BP/NCG/001) to all branch heads for information.

4.2.1.9. The OFE shall make full NUPER reports available to branch heads on request.

4.2.2. Panel of enquiry reports

4.2.2.1. The OFE shall:

(a) assess panel of enquiry reports relating to nuclear safety for their significance to Oldbury using nominated engineers as necessary;
(b) with the help of the nominated engineers, recommend appropriate remedial actions to the Safety Management meeting for consideration and acceptance of the proposals by the department managers (or nominees) and the placement of actions.

4.2.2.2. The procedure for the acceptance, placement and tracking of actions is identical to that for on-site events as detailed above (4.1.3 and 4.1.4).

4.2.3. Nuclear Safety Committee and ad-hod reports

4.2.3.1. In response to:

(a) the station manager (or nominee) identifying potentially relevant events from Nuclear Safety Committee proceedings, or
(b) written/verbal reports of off-site events, or
(c) information from any other source.

The OFE shall:

(a) assess the event/information for its significance to Oldbury, with help from nominated engineers as required;
(b) with the help of nominated engineers, recommend appropriate remedial actions to the Safety Management meeting for consideration and acceptance of the proposals by the department managers (or nominees) and the placement of actions.

4.2.3.2. The procedure for the acceptance, placement and tracking of actions is identical to that for on-site events as detailed above (4.1.3 and 4.1.4).

4.3. SAFETY SYSTEMS UNAVAILABILITY MONITORING

4.3.1. The requirements for safety systems unavailability monitoring for key safety systems is specified in Board Procedure BP/NCG/001.

4.3.2. Key safety systems are identified in the Safety Systems Inventory (Appendix 1 of NCG/NOS(88) 6 - NUSPRA: A Safety Systems Unavailability Data Recording Scheme).

4.3.3. The duty control room supervisor shall use the Production Department's Plant Status Monitoring System (OLD/PODI/12/011 - State of Plant Monitoring and Availability) to log when those systems specified in the Safety Systems Inventory become unavailable and are returned to service.

4.3.4. The OFE shall routinely extract unavailability data from the Plant Status Monitoring System and provide it to the performance and feedback engineer at NCG for entry into the NUSPRA database.

4.3.5. The performance and feedback engineer at NCG will, in response, provide the OFE with performance indicators to enable adverse trends in unavailability to be detected.

4.3.6. Detailed responsibilities for safety systems unavailability monitoring are included in OLD/EODI/18/004 - Safety Systems Unavailability Monitoring.

4.4. ROUTINE TRAINING

Recommendations for routine training arising from event reports shall be reviewed by department managers and incorporated into the appropriate station training courses as prescribed in OLD/MCP/10 - Selection, Qualification and Training of Staff.

5. DEFINITIONS

5.1. DEFERRED ASSESSMENT

The non-urgent assessment of PER summaries by appropriate nominated station engineers to identify local implications and appropriate remedial actions.

5.2. DEVIATION

A departure from normal operating conditions, procedures or practices.

5.3. DIRECT CAUSE

The apparent or superficial cause for an event.

For example: the event resulted from a failure of a relay (direct cause) which was a consequence of inadequate maintenance (root cause).

5.4. EVENT

A deviation judged to be worthy of a root cause analysis.

5.5. MANDATORY ASSESSMENT

The urgent assessment by appropriate station engineers of significant event reports or specific plant event reports designated by NCG to identify local implications and appropriate remedial actions.

5.6. NOMINATED ENGINEER

The engineer identified by the station executive or branch heads as a formal contact for the OFE. He assists the OFE on site investigations following events.

5.7. PLANT EVENT REPORT

An event report that satisfies the NUPER reporting criteria.

5.8. REMEDIAL ACTIONS

Actions taken to rectify any known or suspected deficiencies in station plant, procedures or staff as identified by the OEF process.

5.9. RESPONSIBLE ENGINEER

The engineer with responsibility for carrying out actions identified by the Safety Management meeting following events on-site or off-site.

5.10. ROOT CAUSE

The cause that, if removed, would prevent re-occurrence of the event.

5.11. SIGNIFICANT EVENT REPORT

A plant event report that Nuclear Co-ordination Group judge to have significant generic safety implications for all nuclear stations.

6. REFERENCES

OLD/EODI/18/001	Reporting and Recording of On-site Events
OLD/EODI/18/002	Response to On-site Events
OLD/EODI/18/003	Response to Off-site Events
OLD/EODI/18/004	Safety Systems Unavailability Monitoring
OLD/PODI/12/011	State of Plant Monitoring and Availability
OLD/MCP/11/001	Abnormal Occurrences
BP/NCG/001	Operational Experience Feedback (for Nuclear Licensed Sites)
NCG/NOS/(88)6 NUSPRA	A Safety Systems Unavailability Data Recording Scheme

7. RECORDS

7.1. Records resulting from the procedures described in this MCP are maintained by the OFE.

7.2. Documented records (forms, event reports, etc.) are held in Operational Feedback Section.

7.3. Records shall be managed in accordance with OLD/MCP/05 - Management of Station Records.

7.4. The following records shall be kept:

- Requests for mandatory assessments.
- Responses to mandatory assessments.
- Responses to deferred assessments.
- NUPER on-site reports.
- OLDER on-site reports.
- Site incident register.
- Panel of inquiry reports.
- Ad hoc reports.

APPENDIX A: POSTS NOMINATED AS FORMAL POINTS OF CONTACT FOR THE OFE

The holders of theses posts are to provide advice and assistance in the analysis/evaluation of on-and-off-site events and the development of remedial measures.

(1) PRODUCTION DEPARTMENT

Shift charge engineers (SE)
Mechanical/Civil engineer (SE)
Electrical/Fuel route engineer (SE)
Production services engineer (SE)

(2) ENGINEERING DEPARTMENT

Electrical/Instrumentation engineer (SE)
Mechanical engineer (SE)
Reactor engineering/Engineering services engineer (SE)
Station chemist (SE)
Station physicist (SE)
LTSR engineer (SE)

(3) RESOURCES DEPARTMENT

Health physics services engineer (SE)
Health physics operations engineer (1E)
Emergency preparedness and safety engineer (1E)
Planning engineer (1E)
Quality assurance engineer (SE)
Training engineer (SE)

(4) ADMINISTRATION

Assistant station administration officer (personnel and services)

- Criteria defined to identify significant events.
- Methods of incident collection to log and screen all significant events.
- Methods of analysis, i.e.
  - identification of root causes and lessons learnt;
  - application of precursor analysis;
  - re-examination of the PSA.
- Methods to feedback information to other operators.
- Follow up, and to investigate what action had been taken.
- A system that is documented transparent and can be audited.

Annex XV
LIST OF ACTIONS TAKEN IN RESPONSE TO EVENTS REPORTED TO THE IRS IN AREAS OF EMERGENCY CORE COOLING SYSTEM AND HUMAN PERFORMANCE IN MAINTENANCE AND TESTING (SPAIN)

EMERGENCY CORE COOLING SYSTEM

1. DAMAGE OR CRACKS IN ECCS PIPING AND FITTINGS DUE TO WELDING, WATERHAMMER, OR PIPING TEMPERATURE STRATIFICATION

(A) Replace cracked component and investigate similar piping.
(B) Review weld inspection procedures to ensure their effectiveness in detecting cracks.
(C) Perform evaluation of the potential for temperature stratification existence and the possible effects under the as-found conditions.
(D) Monitor piping temperatures periodically to detect leaks that produce steam voids into low pressure/low temperature systems from high pressure/high temperature systems.
(E) Following waterhammer events, perform system walkdowns to identify any piping or support damage.

2. DEGRADATION OR LOSS OF POST LOCA RECIRCULATION DUE TO DEBRIS, BLOCKAGES OR FOREIGN OBJECTS IN ECCS CIRCUIT

(A) Systems must be properly cleared and rinsed on completion of maintenance work.
(B) Draw up a functional requalification tests that will enable system availability to be adequately checked.
(C) Measure flow rates during periodical testing of safety injection systems and investigate any abnormal variations.
(D) Inspect borated water tanks and other water paths visually, and with remote TV, prior to circulation of coolant in the system.
(E) In the case of insulation covering that could block recirculation flow, it was found that the effect of insulation on sump performance is plant specific and depends on the type and quantities of insulation used in the primary system layout within the containment and post-LOCA recirculation flow rates.

3. ECC AND RHR PUMP MALFUNCTIONS DUE TO GAS LACKING, PARALLEL INTERACTION OR RELIEF VALVE DAMAGE ON MINI-FLOW DISCHARGE LINES

(A) Implement procedures to minimize the potential for gas or air intrusion into the system.
(B) Implement periodic venting of the accumulated hydrogen as a preventive measure and incorporate int in the procedures.
(C) Inspect high points in the system periodically to ensure piping remains full.
(D) For vent valves that allow flow in the reverse direction, check valves were added to correct the problem.
(E) In the case of parallel pump interaction, the immediate corrective actions consisted of an interim change in emergency operating procedures. Permanent corrective actions consisted of installing check valves in each train downstream of the recirculation lines.
(F) Improve the design of pumps and enhance the reliability of their components.
(G) Revise plant procedures to require the piping upstream of the relief valves to be refilled prior to installation of the relief valves and vented through relief valves by hydraulic pressure to eliminate air.

4. BACKFLOW INTO HIGH PRESSURE INJECTION SYSTEM (HPIS) LEADING TO HPIS PIPING BEING SUBJECT TO EXCESSIVE TEMPERATURES AND STRESSES

(A) Replace overstressed piping.
(B) Upgrade pipe supports.
(C) Install a second check valve in each HPIS line inside containment.
(D) Install temperature monitors in the HPIS lines outside containment between the containment penetration and the first outboard check valve.
(E) Perform leak rate testing for all check valves in the HPIS lines inside containment.
(F) Test leak tightness of valves periodically prior to plant startups and following any challenge to the system.

5. INADVERTENT ACTUATION OF ECC DUE TO LOGIC TESTING OR WRONG ACTION OF PERSONNEL

(A) In one case, due to the spurious behaviour of the trip unit and to prevent the inadvertent start of the system, it was decided to disable the system during future logic tests.
(B) Design modifications were proposed to protect sensors against mechanical actions (e.g. maintainer shaking a transducer).
(C) Introduce organizational and technical measures.

6. DEFICIENCIES IN ECCS MOTOR OPERATED VALVES (MOVs) DUE TO "PRESSURE LOCKING", LACK OF ADJUSTMENT TO LIMIT SWITCHES

(A) Replace the failed valve motor and implement design modifications to other MOVs.
(B) In the case of pressure locked valves, provide a vent path to release high pressure.
(C) See other actions related to MOVs in the appropriate section.

1. INADEQUATE TESTING PROCEDURES

1.1. TESTING OF EMERGENCY BUS UNDERVOLTAGE LOGIC CIRCUITRY

(A) Test all the affected undervoltage logic circuits.
(B) Simulate a loss of electrical power by opening the undervoltage or relay potential transformer test switches in order to test:
  (B1) the capability of the UV logic circuits to disconnect a load from emergency electrical buses;
  (B2) the capability of the diesel engine to start automatically;
  (B3) the capability to connect the required emergency electrical loads to the emergency electrical power buses.
(C) Revise the test procedure.

1.2. CIRCUIT BREAKER TESTING PROCEDURE

(A) Follow the procedures established by the manufacturer of the circuit breaker.
(B) Check each component of the circuit breaker.

1.3. DETERMINING THE PROPER SETPOINT OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM ACTUATION

(A) Immediate corrective actions:
  (A1) perform operability testing of back up core cooling systems;
  (A2) adjust the HPCI turbine startup speed control ramp speed;
  (A3) increase the different pressure setting;
  (A4) revise the procedures.

(B) Long-term corrective actions:
  (B1) perform a detailed engineering review of the complete HPCI system and its operating listing to determine whether further corrective actions are desirable;
  (B2) revise the existing surveillance procedures;
  (B3) establish new procedures to monitor HPCI startup transient performance stability.

1.4. TESTING OF THE PRIMARY REACTOR CONTAINMENT PENETRATIONS

(A) Investigate alternative test methods that provide accurate local leak rate tests.
(B) Replace bellows that are unacceptable with testable bellows.

1.5. PARALLEL PUMPS INTERACTION SURVEILLANCE

(A) Change the emergency operating procedures to prevent possible damage to a pump during safety injection actuation.
(B) Check valves in each train downstream of the recirculation line.

1.6. SAFETY RELATED MOTOR OPERATED VALVE (MOV) TESTING AND SURVEILLANCE

(A) Temporary depressurize the valve body in the test procedure.
(B) Replace the actuators with more powerful actuators with modified instrumentation in control.
(C) Review motor operated valves (MOVs) design bases.
(D) Verify MOV switch settings initially and periodically.
(E) Test MOVs under design basis conditions where practicable.
(F) Increase the drive tripping torque setting of all valves of the same type.
(G) Perform recalculation with respect to torques set at commissioning.

2. TEST NEVER PERFORMED

2.1. ISOLATION VALVES IN THE EMERGENCY CORE COOLING SYSTEM (ECCS) RECIRCULATION LINES

(A) Perform more realistic leak rate evaluation.
(B) Implement procedures to lessen the consequences of valve leakage.
(C) Examine the modification valves.
(D) Include the leak rate testing of valves as part of the in-service testing programme.

2.2. INSTRUMENT AIR SYSTEM CHECK VALVES

(A) Replace check valves which are not designed for air service with soft seat spring loaded valves.

2.3. CHECK VALVE IN THE PRIMARY PUMPS FIRE PROTECTION SYSTEM

(A) Requalify by a test of the proper valve installation.

3. INADEQUATE REVIEW AND TESTING OF SAFETY SYSTEM MODIFICATIONS

(A) Review any safety related design change including the effects on all related systems.
(B) Complete tests of the systems under conditions that are as near possible to the conditions expected when the systems are needed.

4. IMPROVEMENT OF TEST CONDITIONS

(A) Operate test on the isolation valve of a safety injection (SI) accumulator tank during hydro test of the accumulator with 2000 ppm borated water instead of pure water to avoid dilution problem.

5. POOR CO-ORDINATION BETWEEN OPERATIONS AND MAINTENANCE PERSONNEL

(A) Develop plant operations and maintenance personnel procedures to ensure complete communication of procedures to be performed and of their status during valve testing and manipulation.
(B) Conduct additional briefing of reactor service personnel on the procedure of boric acid injection into the primary circuit.

(C) Develop a programme of electric driven acceptance hand over testing after maintenance.
(D) Update the relevant procedures and issuing service notifications by maintenance to operating department.
(E) Clarify the responsibilities when preparing routine tests and institute a double control over the decisions related to performing periodic tests.

6. INADEQUATE MAINTENANCE

This section addresses the following aspects of maintenance deficiencies:

- inadequate procedures;
- failure of equipment;
- insufficient training of maintenance personnel;
- poor organization of work by plant management personnel

6.1. INADEQUATE PROCEDURES

(A) Issue service notifications.
(B) Bring instructions for operating the components into line with the requirement of the technical operating instructions.
(C) Correct maintenance documentation with respect to inspection of components.
(D) Correct testing and maintenance schedules in line with the manufacturer's recommendations.
(E) Review components operating procedures to include technical modifications of design.
(F) Pending permanent modification, increase the inspection frequency of concerned components.
(G) Change documentation of test results.
(H) Amend maintenance manual and testing programmes.
(I) Supplement the testing programme by preliminary checks.
(J) Improve the procedure quality level by including operating instructions on preventive replacement of failed material based on the cause of the failure.
(K) Modify the periodic surveillance procedure in order to pay particular attention to the behaviour of a component.

6.2. FAILURE OF EQUIPMENT

(A) Replace the faulty material with higher reliability material.
(B) Extend the checking and control on similar faulty components.
(C) Modify the material.
(D) Develop a design change to correct the problem.
(E) Examine carefully lubrification practices.
(F) Change the testing regime and the test frequency.

6.3. PERSONNEL TRAINING OF MAINTENANCE PERSONNEL

(A) Improve personnel proficiency by conducting additional briefing of service personnel on operations.
(B) Improve compliance with procedures and personnel qualification.
(C) Brief the plant personnel on the results of investigations.
(D) Dismiss the deficiencies personnel until their knowledge check and a briefing on the event causes are conducted.

6.4. POOR ORGANIZATION OF WORK BY PLANT MANAGEMENT PERSONNEL

(A) Upgrade training programmes to improve staff awareness and make a commitment to enhance management control.

(B) Increase the frequency of personnel management exercises.

(C) Plan maintenance work during plant shutdown on safety related equipment.

(D) Develop procedures to ensure complete communication of procedures to be performed and of their status while performing testing and maintenance activities.

(E) Revise permit to work with special devices during maintenance activities and update the relevant procedures.

(F) Specify the responsibilities involved in periodic testing and maintenance work preparation and introduce a double checking system.

(G) Review unresolved previous problems.

Annex XVI
DISSEMINATION AND EXCHANGE OF INFORMATION IN SPAIN

Every licensee in Spain sends its UERs and Annual Reports on OSEF to Unidad Eléctrica SA (UNESA), who is an association of all Spanish electrical utilities. UNESA distributes UERs and Annual Reports to all Spanish licensees and a national training centre called TECNATOM.

As members of INPO, WANO and NSSS owners groups, Spanish licensees report according to applicable requirements. So do they with suppliers of safety equipment.

CSN elaborates event reports to the IRS according to the reporting criteria of the system. CSN elaborates circulars to all Spanish plants describing specific Spanish events, their roots causes and lessons learnt requiring their analysis when it reaches conclusions different to licensee's ones.

Licensees receive INPO, WANO, safety equipment suppliers reports on OEs containing recommendations when applicable, as well as all IRS' reports. UNESA plays the role of centralization and distribution of information coming from INPO, WANO, IAEA and NEA.

1. USE OF IRS IN SPAIN

Analysis of IRS reports is not mandatory in Spain to the licensees. CSN sends the IRS reports, as well as the NEA databank, to UNESA for distribution to the licensees.

CSN carried out in 1990 and 1991 a study so as to check out whether or not there would be worth to require a systematic analysis of IRS reports to the licensees. The conclusions of that study were:

- Most IRS reports referred to equipment, systems or technologies different to those of Spanish plants.
- Some IRS reports were found applicable to Spanish plants, but the subject had already been analysed by the licensee because the issue had been coped with by an INPO, USNRC, German GRS (Spanish plants have either American or German design), vendor document that the licensee is required to analyse.
- There remains a tiny fraction of IRS reports, may be 1 or 2 a year, that are applicable and not coped with in the frame of the regular OEFs process.

Consequently, a decision was made as to not to require the systematic analysis of IRS reports to the licensees. Instead, it was agreed that CSN would endorse for analysis specific IRS reports deemed applicable and not included in the regular OSEF.

In the other hand, IRS system has a major application on the regulatory side. The IRS co-ordinator screens all reports coming and distributes to the experts those he consider of their interest. As an example, there were distributed copies of 9 IRS to 13 people over the period January-June 1994.

The IRS NEA databank is installed on the CSN local network since 1990, allowing any staff member to make his own searches. A manual on how to enter and make use of the system was issued and it is updated as the NEA databank software does.

Main use of the system consists of searches carried out by different staff members when making their own evaluations or studies. A search on the IRS NEA databank looking for similar previous events is a regular step when CSN carries out an in depth event analysis.

The IRS system is more useful to CSN than to licensees, because it is the only source the regulatory body has available to search OE at international scale on informatical support. Licensees count on INPO systems and services to get so, as well as vendors technical support.

Annex XVII
EXTERNAL OPERATING EXPERIENCE DISSEMINATION AND EXCHANGE IN SLOVENIA

External operating experience reports are originated from many different sources. These reports are generated by INPO, NRC, other utilities or nuclear steam supply system suppliers and other international agencies such as WANO and IAEA. These reports are the following:

- INPO Significance Event Reports (SERs);
- INPO Significant Operating Experience Reports (SOERs);
- Licence Event Reports (LERs) from plants of similar design;
- NRC Bulletins, letters, Information Notices;
- Westinghouse Technical Bulletins (WTBs);
- IAEA Reports.

External operating experience reports are going through screening process to select for evaluation of those reports having potential significant impact on the safety and reliability of the plant and to eliminate from the further review those reports determined not to be applicable to the plant.

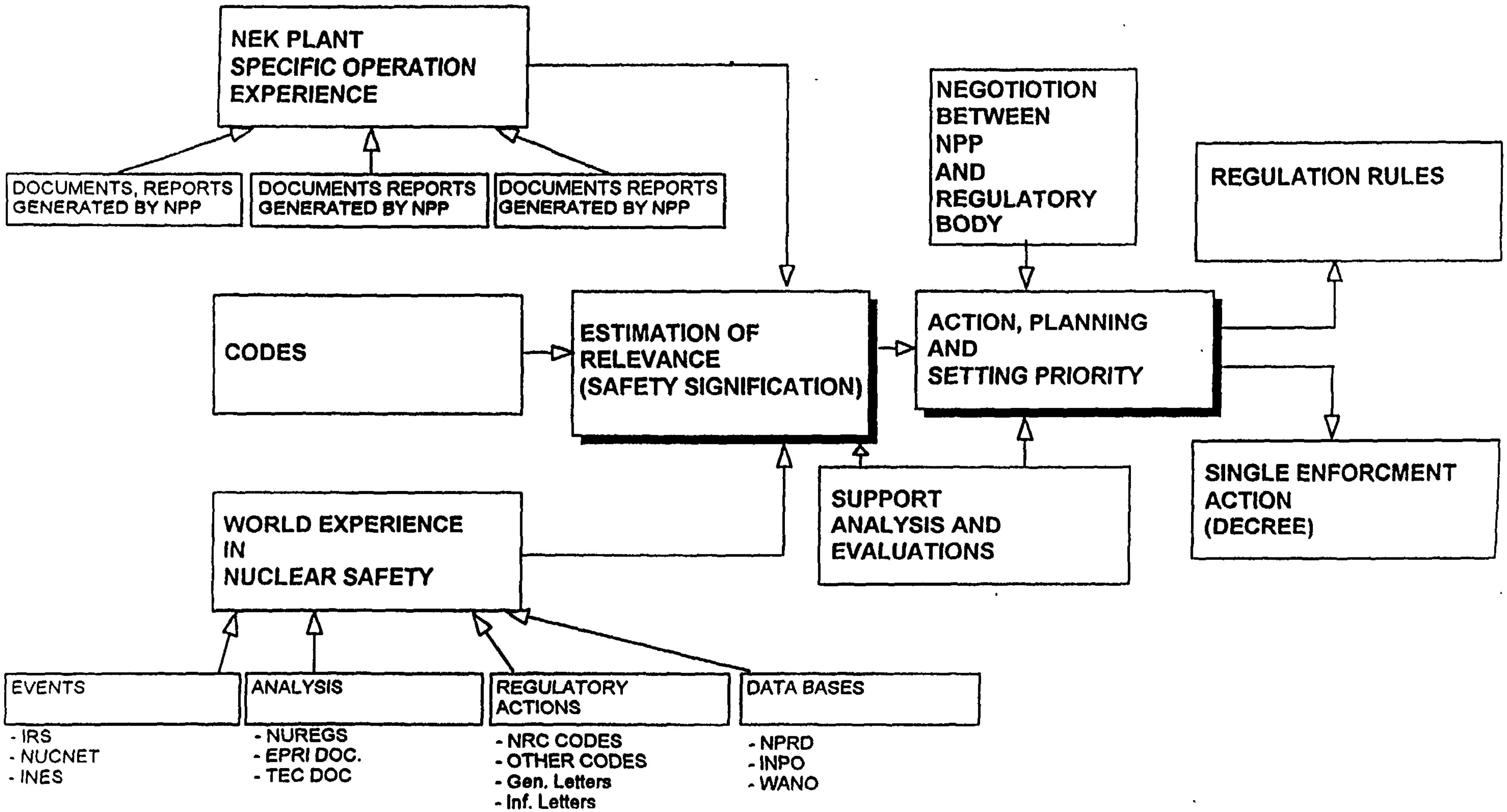
A lot of reports are sent directly to different departments only for their information.

Reports identified as applicable and significant to NPP Krško are evaluated by PPG to determine the implications an operating experience item might have for the plant and to propose corrective actions.

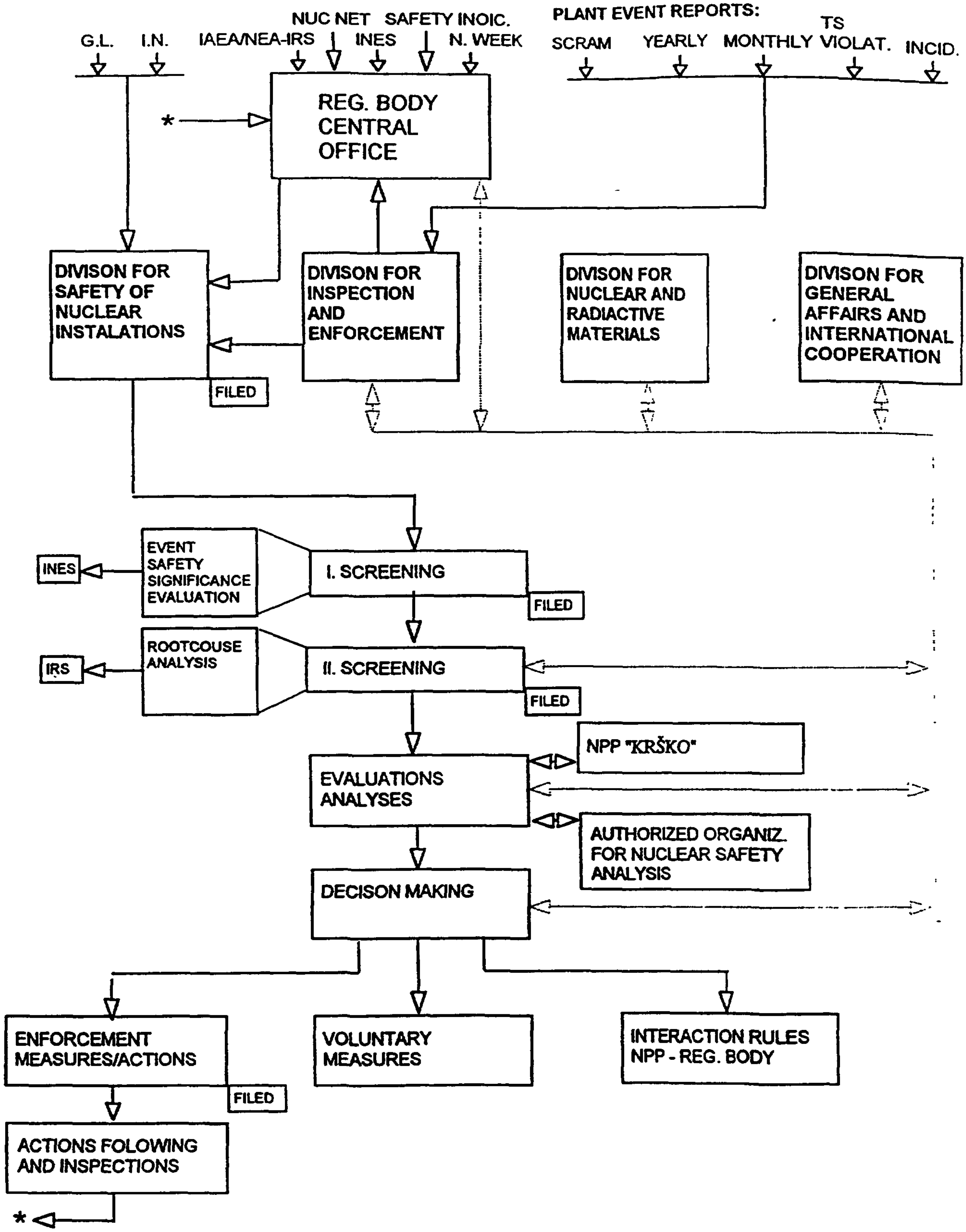
The evaluation process assess the adequacy of plant procedures, design and operating practices for minimizing the probability and consequences of the reported events occurrences.

Event investigation report is sent to KOC for review and approval. After that, PPG distributes operating experience information to all the affected managers/ superintendent/supervisors who are responsible for implementing the approved corrective actions.

PPG supervisor is responsible for using the tracking system to keep the plant management informed on the status of the operating experience assessment programme. These status reports are used by management to identify problem areas and to ensure timely processing of operating experience items.



Principal scheme - VSE of operating experience in regulation



Operating experience flowchart on regulatory level

Annex XVIII
DISSEMINATION AND EXCHANGE OF INFORMATION IN SOUTH AFRICA

Nationally, since South Africa has only one nuclear power plant, and since the plant management, Eskom Head Office and the CNS are kept informed regarding all safety significant events as they occur and are investigated, no special dissemination step to these parties is necessary. The experience feedback to, for example operating or maintenance staff is performed during scheduled or special training sessions. On occasion, case study reports are written on the occurrence, and used for feedback sessions to the staff.

Internationally dissemination is performed as follows:

- The CNS decides if it is necessary to send a report to the IAEA/IRS, and if so, prepares and submits the report. By virtue of having being kept informed on a continuous basis, the CNS has access to all necessary information. A copy of the report is sent to Eskom.
- Eskom, at the plant level, decides whether it is necessary to submit a report to, for example, WANO, INPO, EdF. If yes, the report is prepared and submitted from the plant. A copy of the report is usually sent to the CNS for information.

FOLLOW-UP

The NSL group at Koeberg tracks all the significant events. They monitor that investigations are being performed in the required time scales and that reports are issued. They also ensure that report-back presentations to KORC and the NSRC are made. The NSL group also performs trending and statistical analysis of the significant event (occurrences) programme. For example, how many events were caused by different aspects of equipment failure, how many were due to different aspects of personnel error.

The CNS independently performs statistical analysis on the significant occurrences to identify trends and the distribution of the causes of the occurrences.

A QA audit of the "functional control of deficiencies and corrective actions" is performed on an annual basis. The intent is to determine compliance with the CNS Licensing Documents and with the Generation Nuclear Group Management Manual, as well as the effectiveness of the overall programme. The CNS participates as an observer during this (5-day) audit.

Annex XIX
SUPERVISION OF OSEF BY THE CSN

The CSN audits the OSEF programme of each licensee every two years. The scope of the audit is:

- **Organization of the OSEF programme**, i.e. staff, training in root cause methodologies, procedures for conducting the OSEF programme, quality, completeness and easy handling of OSEF records.
- **Performance**. It is assessed by checking out several OSEF items, either in-house and external, as for quality of screening, analysis, appropriateness of corrective measures and timing for implementation.
- **Senior management awareness of OSEF matters**. Usually a good way to ascertain this issue is to check out whether or not internal audits to the OSEF programmes on behalf of the licensee high management, the scope and results of such audits.

CSN presents its audit findings to the licensees and corrective measures are discussed and agreed.

ANNUAL REPORT ON THE OSEF PROGRAMME

The annual report is submitted to the CSN by every licensee. It is also distributed to all other Spanish licensees in order to make them abreast of the OSEF programmes results at each plant.

The report is usually issued within the first quarter of the next year. The content of the report is:

- In-house events: a table of all LERs happened over the year, indicating LER reference, status of all corrective measures. If corrective measure is still not complete, an estimated dateline shall be included.
- Events of the other Spanish plants: a table of all such LERs that the licensee has considered applicable to its plant (usually just a fraction of all, ranging from 5 to 10%), containing the same information as above.

- Non-spanish experiences: a table of required external experience. For an USA[1] designed plant it means all:

    - significant operating experience reports (SOER) issued by the Institute of Nuclear Power Operations (INPO). Usually 2-4 a year;
    - significant event reports (SER) issued by INPO. Around 20 a year;
    - safety equipment non-compliance communications (10CFR21) issued to the USNRC by vendors supplying safety equipment installed in the plant. The estimation of such items may range between 5-20 a year, heavily depending on the number of suppliers considered and the plant design.

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[1]CSN endorses modifications to the Code of Federal Regulations, part 10, USNRC Bulletins and Generic Letters to Spanish plants designed by US vendors. Any such an item shall be analysed by Spanish licensees as for applicability to their facilities. A semi-annual report is submitted to CSN describing the results of those analyses.

For Trillo, a German[2] designed plant, it means all:

- Weiterleitungsnachricht (WLN) issued by the German GRS. Usually ranging from 10 to 20 per year.
- Safety equipment communications received from suppliers. It has been agreed that this requirement is covered by analysing all applicable Service Reports and Experience Reports issued by the NSSS vendor: Kraftwerk Union (KWU).

- Experiences required by CSN. Maybe an IRS report or experiences on Spanish plants that do not meet the reporting criteria but have been deemed of safety interest by CSN, e.g. defective components discovered during preventive maintenance.

[2]Analysis of applicability of German Nuclear Safety Standards (KTA), recommendations issued by the Reactor Safety Commission (RSK) is required to Trillo NPP, that submits its conclusions to the CSN in a semi-annual report.

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