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EVALUATION OF RELIABILITY DATA SOURCES

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FOREWORD

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In plant design reliability data are used to evaluate the safety implications of various redundancy strategies. In plant operation reliability data are needed for the evaluation of maintenance schedules, allowable outage times and to optimize test intervals.

Other uses of realiability data include: root cause and failure trend analysis, common cause failure analysis, plant performance indicators and optimization of spare parts inventory. Of particular importance is the use of reliability data in probabilistic safety assessments.

It is clear that the establishment and the maintenance of a reliability data bank based on specific NPP operational experience requires considerable commitment from the organization managing the work. Experience shows, however, that benefits clearly compensate for the costs involved, particularly if one considers the multiple uses of the collected data in NPP safety and availability analysis.

Clearly, if data are to be suitable for use in PSA studies, their collection should be organized in a way which meets the needs of the PSA analyst. This implies that a close liaison between the people who are responsible for the collection of data and the PSA analysts should be established. The combined experience of the Agency's member countries in the various aspects of data collection, analysis and retrieval represents a much richer source of data than that which could be provided by any single country.

In order to ensure that this wealth of experience can be made use of, a Technical Committee Meeting was convened by the IAEA in Vienna, 1–5 February 1988. The general objective of the meeting was to compile and to disseminate ongoing work and experience with reliability data sources including aspects of data collection, analysis and retrieval.

This Technical Document, prepared by the participants in the meeting, highlights the issues discussed during the meeting. The document reviews, based on the information available in the group of participants, the experience in Member States and identifies areas where further work is needed.

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EXECUTIVE SUMMARY

To compile and to disseminate on-going work and experience with reliability data sources, including aspects of data collection analysis and retrieval, a Technical Committee Meeting was convened by the IAEA in Vienna, 1-5 February 1988.

This technical document, prepared by the group of participants in the meeting, reviews, based on the information available to the group, the experience in Member States and identifies areas where further work is needed.

Chapter 2 reviews the role of reliability data in plant safety and availability. It also identifies major issues in data collection, evaluation and utilization. Among these issues the incompleteness and inconsistency of data, the factors affecting failure data, data validation and the use of generic and plant specific data are addressed.

In Chapter 3 the experience of some countries in establishing and operating data bases is compiled in the form of tables. Both event data bases and reliability parameters data bases are addressed.

The following aspects were compiled:

- status and general characteristics of data bases
- main information items provided
- modes of data collection
- manpower and computer resources in data collection
- means of data manipulation
- potential application of the data banks

A full section of this chapter is devoted to the Component Event Data Bank (CEDB) of the JRC Ispra managed European Reliability Data System.

In addition, the chapter contains a detailed description of information to be collected for an adequate failure characterization and use in safety analysis. Quality assurance aspects in data collection, validation and screening are also addressed.

New data collectors can benefit from the experience gained during data collection and data base formation in other countries. Therefore, comments on the experience available and potential pitfalls are given.

Chapter 4 addresses data management and analysis Once the data base is established special attention should be given to the computerized data base management system in order to choose the most appropriate one for each particular application. Information found in data bases must be processed using appropriate statistical techniques selected in accordance with the characteristics of the data and requirements of the analysis.

A number of data bases containing component reliability parameters are available in the open literature. They usually differ with regard to size, level of detail and the ultimate source of data. The comparison of main characteristics of representative data sources is also found in Chapter 4. Special attention is devoted to the Reliability Data Book of CEDB and to the IAEA compilation of generic component reliability data.

The data for rare events (some initiating events are usually of that kind) usually cannot be collected from plant operational experience. Special studies used to assess that kind of data are presented.

Information about events can be used to generate system reliability indicators. Reliability indicators which condense a large amount of information about the system operation into a few quantitative parameters is discussed.

Finally, the participants in the TCM identified aspects which deserve special attention by those involved in collecting and using of reliability data. Among others are some remarks about historical quality of data, operating times needed to derive parameters, root causes of failures, importance and cost of data collection efforts, manpower requirements and needs for international cooperation and standardization.

1. INTRODUCTION

Reliability data plays an important role in Nuclear Power Plant safety and availability.

In plant design reliability data are used to evaluate the safety implications of various redundancy strategies. In plant operation reliability data are needed for the evaluation of maintenance schedules, allowable outage times and to optimize test intervals.

Other uses of realiability data include, root cause and failure trend analysis, common cause failure analysis, plant performance indicators and optimization of spare parts inventory. Of particular importance is the use of reliability data in probabilistic safety assessments.

It is clear that the establishment and the maintenance of a reliability data bank based on specific NPP operational experience requires considerable commitment from the organization managing the work. Experience shows, however, that benefits clearly compensate for the costs involved, particularly if one considers the multiple uses of the collected data in NPP safety and availability analysis.

Data can be derived from generic or plant specific information. For use in Probabilistic Safety Assessment (PSA), data is needed for the frequencies of initiating events, the rates of the different component failure modes, common cause failure rates, repair times, unavailabilities due to tests and for preventive maintenance and human error probabilities.

Specific field data collection campaigns, generic published information, laboratory testing, expert opinion, and abnormal event data from operational experience are the most common sources of data.

Available data is, however, often of limited use in PSA studies because of the way in which the data have been collected. Frequently events are recorded without providing any information concerning the time over which the data have been collected or the number of demands which have been made during this time. Other shortcomings in available data sources are the lack of sufficient detail concerning the event or operating conditions and the lack of consistent and well specified definitions, including component boundaries, operating environment, design specification and available information on failure mode or root cause.

Clearly, if data are to be suitable for use in PSA studies their collection should be organized in a way which meets the needs of the PSA analyst. This implies that a close liaison between the people who are responsible for the collection of data and the PSA analysts should be established. The combined experience of the Agency's member countries in the various aspects of data collection, analysis and retrieval represents a much richer source of data than that which could be provided by any single country.

In order to ensure that this wealth of experience can be made use of, a Technical Committee Meeting was convened by the IAEA in Vienna, 1-5 February 1988. The general objective of the meeting was to compile and to disseminate ongoing work and experience with reliability data sources including aspects of data collection, analysis and retrieval.

This Technical Document, prepared by the participants in the meeting, highlights the issues discussed during the meeting. The document reviews, based on the information available in the group of participants, the experience in Member States and identifies areas where further work is needed.

2. ROLE OF RELIABILITY DATA

Reliability data plays an important role in Nuclear Power Plant (NPP), availability and safety. The establishment and the maintenance of a reliability data bank based on specific NPP operational experience requires considerable commitment from the organization managing the work. Experience shows, however, that benefits clearly compensate for the costs involved, particularly if one considers the multiple uses of the data associated with NPP design and operation.

2.1 Probabilistic Safety Assessment

Safety assessment in general and probabilistic safety assessment (PSA) in particular provide important insights to be considered in connection with the design and operation of nuclear plants.

Three different levels of PSA are defined in the literature. Thus, PSA may be focused on plant analysis and in calculating the frequency of core damage. It may also address the physical phenomena leading to uncontrolled radioactive release and its associated probability or the risk of harm or injury to the general public.

This document is restricted to the first of these objectives (level 1 PSA).

In normal operation of the plant none of the above conditions would be reached. They can only be experienced if the plant operates beyond its design limits. The plant can be placed beyond design if it is subject to accidents or operational transients, that are not controlled by the various control and safety systems.

In order to calculate or assess the probability of the above conditions the analyst must determine the frequency of occurrence of accident initiating events, together with the unavailability of the appropriate safety systems. This can only be done if the appropriate data on initiating events and component failures are known.

The application for which PSA has been used most widely in the past is identification of design or operational weaknesses having an impact on the core damage frequency. Evaluation of the dominant accident sequences, and system failures and human errors can identify relatively weak points that, if improved, would most effectively reduce the expected core damage frequency.

In order to make most use out of the probabilistic safety assessment, it has to be continuously updated. Therefore, operational experience must be monitored to ensure that the various assumptions used in the analysis are not violated.

The use of specific operational data is of utmost importance, because the examination of such data can highlight those areas where safety needs further consideration. As a result of the safety review, the resources for safety improvements can then be allocated more effectively.

Probabilistic Safety Assessment methodology provides a very important systematic means of predicting new accident scenarios. These scenarios can be extracted either from a PSA, if available, or by constructing dedicated event and fault trees for specific accident sequences. This information can be used in operator training to alert him to the various accident scenarios. Additionally, it may be used to identify weak points in the knowledge of the plant personnel

For this subject, the concept of a living PSA is applicable and useful. This form of training necessitates an updating process for in-plant data collection and analysis to periodically revise the plant specific accident scenarios used for training based on in-plant event experience and to provide information required to modify both emergency and normal procedures of operation.

2.2 Uses of reliability data in NPP Design

Several options might be available to achieve a specified function at the design stage of the plant. The designer must consider the structure of the systems, the choice of materials, the reliability of the available options, quality assurance programme, etc. Reliability data are used to evaluate the reliability for each of the various design options. This information enables the designer to judge which of the option should be adopted, taking into account a cost-benefit analysis and therefore enabling him to realize a more rational and economic design.

Evaluation of design option imply in the specification of different levels of redundancy and diversity.

A general procedure to specify redundancy and diversity levels includes:

i) specify the possible different system structures;
ii) perform reliability analysis for each defined system structure using the same data set input;
iii) evaluate the dependencies and common mode aspects for the different redundancy and diversity levels (including considerations of diversity and separation);
iv) compare and evaluate the results for the various system structures.

The identification of specific aspects of the level of redundancy considers that:

a) redundancy can be established on different levels to achieve a given safety goal;
b) the logic of redundancy is not unique (e.g. parallel, voting logic); therefore various options have to be studied,
c) the effectiveness of the redundancy is dependent on the operational (i.e. whether passive or active trains are used) mode as well as on the reliability of the individual trains.

The identification of the diversity level involves analysis of the effects of:

a) functionally diverse systems/components, e.g. for reactor protection and shut-down, decay heat removal;
b) technically diverse systems/components, e.g. for different power (steam or electrical) supply units;
c) manufacturing diversity, i.e the components manufactured by different companies.

In conclusion for the identification of the appropriate redundancy and diversity levels reliability data are needed beyond the simple component failure rates, i.e. common mode failure data and human interactions are also required. It should be noted that the great variety of possible design options can increase substantially the amount of effort needed to complete this task.

2.3 Use of reliability data in NPP Operation

The use of reliability data in NPP operation can be characterized in three general areas namely: Operational Management, Safety Management and Operator Training.

2.3.1 Operational Management

In the framework of reliability analysis for operational management generally two types of information are needed:

(i) a description of operational procedures, and a description of the different operational modes that need to be considered in the analysis, and

(ii) reliability data as input for the analysis.

2.3.1.1 Maintenance Schedules

Specifically, to evaluate plant maintenance schedules on a probabilistic basis the following analytical steps should be performed:

a) define the calendar period to be considered. Basically it can be categorized as

- refuelling period, i e. several weeks per year,
- operational period, i.e. about one year,
- one campaign, i.e. one continuous operational period plus subsequent refuelling period (normally one year for PWRs),
- several campaigns, i.e. several years or refueling period,

b) determine the different maintenance schedules which can be considered as possible options. Typically long term schedules (for several campaigns) are desirable. A maintenance schedule should include all plant systems, subsystems and complex maintenance activities.

c) Obtain the input reliability data required. For maintenance scheduling three types of data are needed:

- failure data;
- repair and recovery data; and
- test induced failure data (i.e. errors occuring before or during test which render the system unavailable after the test).

d) calcule the equipment/system reliability or unavailability for different maintenance policies and its impact on plant safety.

e) compare the results, and derive the economic worth of the different policies considered in the analysis.

It should be noted that in the course of the analysis, special attention must be paid to maintenance activities, failures caused by the maintenance actions, as well as to the specific procedures used for both technical and economic evaluation

2.3.1.2 Optimisation of Test Intervals

For optimisation of test intervals the following procedure can be used (together with the information listed):

a) definition of time period considered. Basically it can be:
- one test cycle (i.e. one stand-by period plus test period)
- one operational campaign of the plant (i.e. several test cycles).

b) collection of input information for reliability analysis. The information involves:
- test procedure description (i.e. frequency of testing of each train of a redundant system, shift period within train tests);
- reliability data for stand-by period.

c) calculation of unavailability of components/systems considering different test frequencies;

d) consideration of their impact on the probability of occurence of dominant cut sets in PSA,

(e) evaluation and interpretation of results.

The unavailability of a component/system is subject to two opposite effects if the test frequency is changed.

1. In case of more frequent testing the contribution from stand-by latent failures decreases,
2 the contribution to system unavailabilities due to testing and from test caused failures increases.

A time dependent reliability analysis based on the minimum cut set of the system is normally performed to determine the minimum time averaged system unavailability. This is important when the system has an important impact on plant safety and availability.

It must be emphasized that the optimisation described above involves only technical aspects. In practice for the selection of the actual test frequency other aspects, such as available manpower and economic efficiency also need to be considered.

2.3.1.3 Outage Times

When a failure or an abnormality of a safety system is detected by the surveillance activities, the system goes into a repair. Technical specifications (Operating Rules in U.K.) restrict outage times of these systems to prevent unacceptable increase in the risk. If longer outage times are required, the plant must be shutdown. PSA produces the dominant minimal cutsets which lead to core damage. These cutsets indicate the importance of the safety systems and of the various components. If contribution of outage of a system is small or negligible, the allowable outage time can be increased if required. It is argued that if the outages due to repairs do not affect significantly the availability of the various safety systems, repairs can be carried out during plant operation. Repair times are more controllable than the failure data. Therefore, operating experience can be used only as one of the ingredients in estimating or planning repair times for a certain equipment. PSA can be used for the determination of the permissible outage time of the system or components. Reliability data used normally in PSA are needed for this application. Appropriate safety criteria must be used to specify acceptable risk levels, against which outage times can be judged

2.3.1.4 Spare Parts

The repair times are a function of the spare part inventory at site. Storage space and budget for the provision of spare parts are, however, conditions to be considered. Actual component reliability data, allowable outage times, data on accessibility of components, costs and space available, repair times and estimated time needed to obtain replacement parts, can be used to optimize the inventory of spare parts.

2.3.1.5 Availability of Plant

Availability of the plant has an impact on both its economics and its safety.

Reliability data of various plant systems can be used as performance indicators of plant safety. In this framework, PSA results particularly the importance of safety systems in the total core damage frequency can be of most use.

2.3.2 Safety Management

2.3.2.1 Backfitting and/or Design Improvements

The U.S.NRC has conducted a program named " Integrated Plant Safety Assessment, Systematic Evaluation Program". The aim of this program was to review some older plants with respect to the more recent regulations imposed on the new plants. As a result, backfitting issues and design improvements were identified. However, before imposing corrective actions to be taken by the utilities, a PSA was performed to evaluate the relative risk impact of the proposed changes or improvements. The program included the development of the fault-trees for the relevant system pending improvement or backfitting. Actual failure data was used for the quantification of fault trees.

2.3.2.2 Accident Management

Reliability data are also used in the area of accident management. Reliability analysis and PSAs can be used to provide guidelines to the operators. The accident sequences must be reviewed to specify appropriate operator actions, as well as to identify safety features which can be used in accident management. The aim is to stop accident propagation and to mitigate consequences of accidents.

Emergency preparedness should take into consideration the human ability to cope with procedures under situations of stress. Therefore, human reliability analysis, as well as ergonomic aspects should be considered.

The reliability data needed for this task are those normally used in PSA, and the human reliability data in conditions of stress.

2.4 Reliability Data Requirements

2.4.1 Initiating Events

For the selection and frequency evaluation of initiating events (IE's), two types of data sources are generally required:

a) Event data - actual occurrences in NPPs such as those reported in LERs

b) Failure data - generic or plant-specific failure data for evaluation of low frequency initiating events.

The main source needed is the event data from the operating experience of NPPs. Ideally, any occurrences in a nuclear plant should be analyzed with respect to a periodically updated master list of initiating events, in order to determine whether it is an event that can be classified as an "old" type already identified in the master list, or whether a "new" type of event has occurred, which should then open a new category in this master list.

An example of a list of IE's that developed over several years and is frequently used in the last years is the EPRI-NP 2230 list of 41 PWR IEs and 37 BWR IEs. All LERs in the NRC file up to the end of 1983 have been put into the above categories. This list covers anticipated transients only. In a number of recent PSAs this list was augmented to include other sources of initiating events based on additional LER evaluations and other considerations. Thus, it can be seen that at this time the PSA analyst does not have a well agreed master list of IEs where all LERs are included on a periodical basis. Such a list is desirable for use as the prior for the selection and determination of the relatively frequent IEs.

In order to be useful as a prior, the IE master list should include an estimate of the frequency of each IE. This estimate could be based on the data accumulated in national or international event data banks, and on published PSAs.

The low frequency (rare) events do not occur frequently enough to be established from the experience with LERs. In these cases engineering judgement is used for their selection, and a special analysis (fault-tree or other methods) is performed to determine their frequency. An example is the interfacing LOCA. In this case the location of the break and its size are important for the PSA analysis. However, the few IERs available are just an indication of the possibility of such an intiating event, and the frequency of this event for different break sizes and location needs to be separately assessed.

A common master list for selection of low frequency IEs for PSA does not exist. Rather, different PSAs generate their own lists based on previous studies and plant specific IERs. There is a need for a master list also for the low frequency (rare) initiating events for future PSAs. Elimination from this master list of those events which may be not relevant for a particular plant should then be done on a case by case basis, and thoroughly documented

It is recommended that master list for IEs and their expected (generic) frequency should be prepared for use in the categorization of operating experience, and to be the starting point in choosing IE for PSA study.

2.4.2 Failure Data

In order to undertake the various reliability calculations the analyst requires the following information: (i) mechanism of failures, (ii) failure modes and (iii) failure rates.

First the analyst must decide on the mechanism of failures to be considered, such as whether they are demand-dependent or time-dependent, whether they are random or dependent on each other. Second, he needs the failure data separated into a sufficiently fine selection of failure modes. Additionally, numerical failure data must be characterised either by their mean or median values and uncertainty, or by their distributions. The failure data at the component level is required for most applications. system or subsystem data may be useful in some of the applications discussed in section 2.2.

2.4.3 Testing

Numerical calculation of reliability parameters depend on the operational mode of the components/systems; i.e. different data are needed if continuously operating components or stand-by components are analysed.

From the point of view of the analysis, the stand-by components can be classified as:

- repairable continuously monitored,
- repairable periodically tested,
- or non repairable.

To calculate reliability parameters, i.e. averaged or time dependent unavailability of periodically tested components, the following input data are required:

- failure data (failure rate, unavailability per demand)
- repair data (repair time)
- information on testing policy and efficiency (including test induced failure data).

Information on testing policy and efficiency involves the following parameters:

a) frequency of testing,
b) duration of test period, including further repairs (if any)
c) failure detection efficiency during tests; i.e. the probability that the failure mode in question will be identified during the test,
d) test caused failure probability, i.e. probability of failures which can be caused by the testing procedure, (incl. non restoration after test)
e) test override probability; i.e. probability of taking the component under test back to operation in case an actual demand occurs.

The data (a) and (b) can be extracted from actual operational procedures, while data (c) to (e) can be deduced from operational statistics (from the raw data). This kind of data is in general not readily available.

Due to the opposing effects of the above factors, a test frequency optimisation can be performed as mentioned in Section 2.3.1.2.

In practice most of the safety systems or safety-related systems are periodically tested according to a predetermined time schedule. All trains of redundant systems have to be tested, and both the test period and the test stagger have to be considered and taken into account in the analysis of the redundant system.

2.4.4 Repair and Recovery Data

2.4.4.1 Repair Times

Repair time is the time needed to repair a system or component, which has failed during the operation of the plant. It should be noted that the repair time is not necessarily equal to the time during which the system is unavailable due to a failure since repairs do not always start at the time of the detection of the failure. Furthermore repairs must be followed by testing to ensure that the system is functioning again within its technical specifications requirements.

Since the repair time can contribute appreciably to the total time during which the system is unavailable, the repair time must be determined and controlled. There is a general need to calculate the maximum allowable repair times on a more sophisticated basis. Probabilistic safety criteria can be used to specify the maximum time for maintenance or repairs of components and systems during plant operation.

To calculate allowable repair times, the following information is necessary:

- a) meantime to failure
- b) meantime to detect failure
- c) meantime to repair
- d) meantime of test.

Plant operating experience including manpower available for maintenance and availability of spare parts is needed and should be used to complement above information.

<u>Recovery Data</u>

The more recent PSAs (such as Oconee PRA) do not conclude with a list of plant dominant accident sequences but apply on the resulting dominant sequences a set of recovery probabilities. This can significantly change the resulting hierarchy of dominant sequences These recovery probabilities manifest the probability of operator corrective actions to mitigate the accident sequence the plant may experience. The recovery probabilities are derived based on the emergency procedures of the plant and correlate to their effectiveness and operator training in their use.

The determination of recovery data is not yet well established. There is a need for data that will help the analyst to choose recovery probabilities based on the factors presented above as well as other additional factors. Further discussion of this data would be covered in studies of human reliability data.

2.5 Issues on Data Collection, Evaluation and Utilization

Some issues on data collection evaluation and use are detailed in the following sections

2.5.1 Data Collection

2.5.1.1 Availability/Incompleteness of Data

For Reliability Analysis and PSA required data are compiled from generic sources, plant specific statistics, and engineering judgement.

Generic data have been published in various report in different countries The use of such generic data involves a number of problems which include:

- the limited types of components involved and unknown assumptions;
- the inconsistencies between various published generic data banks; (Section 2.5.1.2)
- the difficulty in application of generic data to some specific components (Section 2.5.3.1)

Plant specific statistics are available for plants which have been in operation for long periods. Even in this case difficulties still exist, namely:

- how to extract the information needed for reliability analysis from raw data sources, which very often does not include all the parameters of the required data in existing data collection systems. To avoid this problem it recommended that a reliability analyst should be involved in the development of raw data collection system for a new plant from the earliest stages.

- for old plants the current forms and methods should be revised and extended by reliability analysts.

Data based on engineering judgement very often imply data suggested by one person. Collective opinion and suggestion of an experienced team composed of both operational staff and realiability analysts are preferred (see Section 2.5.1.5 and 4.3).

2.5.1.2 Inconsistency of Data

Since the raw data are usually collected from many different plants on several sites, the data collected can include inconsistencies. Frequently encountered problems are listed below:

- Inconsistent definition of components or systems, especially in the definition of the system boundary and interface points;
- Inconsistency of the component boundary, e.g. interface to the control system, the power supply, and the lubrication system. Therefore, the boundary of each component needs to be clearly defined in order to avoid overlaps or omissions.
- Inconsistent definition of failure. For example, a component success/failure criterion depends on the failure mode and failure severity in the system analysis. Failure due to an inapplicable failure mode or incipient failure should not be considered.
- Inconsistent definition of operational data, including the number of demands estimated and the operating times. These should be assigned for each failure mode separately. For example, the failure rate of "failure to run" should be calculated on the basis of the operational time, but that of "external leakage" should be calculated either on the basis of the time the component or system is under fluid pressure or, if this is not available, on the basis of the calendar time.

2.5.1.3 Statistics and Uncertainties

The following statistical problems are also frequently encountered:

- Assumptions used in the calculation of statistical parameters must be well documented and kept in mind when drawing conclusions. The assumptions must be included in Reliability Analysis or PSA reports.

- Uncertainties are sometimes misinterpreted. They are sometimes mistakenly believed to express objective stochastic variations of given parameters However, because of the lack of sufficient statistical information this is not necessarily so. There is a large amount of subjectivity in the specification of uncertainties. People with operational as well as statistical experience should be involved in the collection, evaluation and use of reliability data.
- Computers in statistical analysis are useful, but must be used cautiously. An understanding of the statistics and the various assumptions is of most importance in manipulation of data.

2.5.1.4 Random/Dependent Failures

In the design of complex, highly reliable systems the designer and the safety analyst must work together to decide how many trains will be necessary to achieve the required reliability.

As it is generally recognized that complex, multiple-train systems can be affected by hidden dependencies, ways must be devised by the designer and the analyst for dealing with them. Generally, they look for guidance by examining operational experience with systems of similar complexity. However, operational data are not always collected in a way which makes such an examination fruitful. Therefore, it is recommendable that, whenever possible, reliability data clearly indicate which ones are truly random and which failures are related.

2.5.1.5 Engineering Judgement

Engineering judgement is sometimes required to obtain reliability parameters for those components for which raw data do not exist. Engineering judgement is necessary on all levels of handling reliablity data in both Reliability Analysis and PSA.

It could be dangerous to use the judgement of either anonymous experts or well known experts with expertise which is irrelevant to the problem in question. It must be always ensured that it is known who the experts are, what are their credentials and what are the bases for their opinion and judgement. It must be remembered that because experts frequently have their knowledge influenced by their background and interactions with each other, their advice may be not completely independent. Engineering judgement should be tested, as with any other source of reliability data, and not accepted uncritically.

Engineering judgement should be provided by skilled engineers who have a considerable experience in designing, constructing and operating NPPs, or in related topics as well as in Reliability Analysis and PSA work.

It is also more advantageous to use engineering judgement to modify available data for similar application rather than to use it to produce new data for which no related experimental data is available.

2.5.2 Data Evaluation

2.5.2.1 Sources of Reliability Data

There are many sources of reliability data that the analyst should consider. However, the various sources must not be used uncritically. The analyst must be aware of the advantages and of the limitations of the various sources.

The available sources can be broadly classified on by how the data were obtained and on how specific they are for a particular application. Classification of the data according to the way the data have been obtained is shown in Table 2.5.2.1.-1. Classification of the data according to how specific they are is shown in Table 2.5.2.1.-2. Which source of data should be used depends on specific conditions and requirements.

Table 2.5.2.1-1:
Classification of data according to the way they have been obtained

SOURCE OF THE DATA	ADVANTAGES	LIMITATIONS
Operational data	most appropriate	full spectrum of data rarely available
Field tests	influence of various parameters can be ascertained	important operational parameters can be missed
Laboratory testing	failure mechanisms can be more easily identified; testing can be accelerated	possible unrealistic and over-simplified operating conditions
Generic published information	data conveniently available; sometimes endorsed by reputable organizations	primary source is not always given and thus not open to scrutiny; use may be misleading; applicability problems
Expert opinion (engineering judgement)	sometimes this is the only source available	can be misleading; credentials of the experts must be known; must not be accepted uncritically

TABLE 2.5.2.1-2:
Classification in accordance to how specific the data are

SOURCE OF THE DATA	ADVANTAGES	LIMITATIONS
own plant specific data	most appropriate	not always available
plant reference data	provided by the vendor of the plant; often checked and validated	are not appropriate if vendors' assumptions and experience are not reflected in the operation of the plant
generic data	most readily available	not always appropriate, must be used cautiously and critically

2.5.2.2 Factors Affecting Failure Data

Reliablity data are characterized not only by stochastic features such as frequencies of random events. There are other factors arising from both environmental and operational conditions of the components and of the systems.

For appropriate use of operational experience data in Reliability Analysis and PSA it is necessary to collect and evaluate a wide range of operational information. The following list of types of information is given as a minimum of what should be taken into consideration:

- date and time of the event and transient sequence as well as the time from the beginning of the normal operation of the component
- operational state of the plant at the occurrence of the event
- operational states of systems involved in the transient
- influence of the event on the state of the unit as a whole
- evaluation of the influence of the event on nuclear safety
- causes of the event as well as their evaluation
- special technological system characterization
- characteristic features of history of inspections, testing and repair of systems and components related to the event
- other information on technology, operation and maintenance, such as the age of the component

2.5.2.3 Categorization of Initiating Events

To help in performing plant-specific PSAs, EPRI (14) and NRC (15) supported studies to prepare generic lists of initiating events. In these studies a list of 37 IE categories for BWRs and 41 IE categories for PWRs was recommended. The latter study reevaluated the IE categories and determined:

(a) a comparison of the PWR and BWR lists shows that they do not have similar detail in part of the categorization and mainly with respect to loss of condenser initiated events.

(b) categories PWR 38 and BWR 34 "cause unknown" are too broad in nature and should be detailed when the list of initiators would be revised.

Another observation of lack of sufficient information for categorization of initiating event can be found in a report (16) which was based on a circulated questionnaire to all utilities operating NPPs. The questionnaire was not defined well enough to screen the correct initiating events. The result was that 109 events were categorized as LOSP, while later studies (17,18) revealed that only half of them could be correctly categorized as LOSP events.

It is recommended that categorization of IEs would be made by use of event descriptions that will well define each IE that is categorized and selected for data collection and frequency evaluation. This would help to avoid misinterpretation of the events.

2.5.3 Use of Data

2.5.3.1 Generic vs. Specific

It is obvious that for plant studies plant specific data are desirable. The appropriate amount and quality of data for such studies will be only available if the plant has been operated for a long period and systematic data collection has been performed. Unfortunately in many cases it is not so; therefore it will then be necessary to extract reliability information from generic sources.

There are three main problems using generic data for plant specific studies:

a) Generality of data. In many cases the generic data are derived from plant specific experiences and from engineering judgement (see WASH-1400), so they are sometimes more specific than really generic.
b) Combination of generic and plant specific data. This combination has to be done on component, sub-system, or system level. Bayes theorem is an approach to this problem.
c) Interpretation of results, from calculations based on a combination of generic and specific data. Results which have a relative nature, e.g. importance factors or ranking ratios (risk reduction worth, risk achievement worth) are less dependent on data used and can be used with greater confidence than absolute values. To illustrate the dependency of results on input data, extensive sensitivity analysis are required.

2.5.3.2 Limitation of Data

As discussed in Section 2.5.2.1 various sources of reliability data are available to the analyst. The analyst must recognise that the reliability data cannot be used universally and in all circumstances. All data are subject to certain limitations and these must be appreciated, recognized and understood.

The analyst must ensure that as far as possible he uses the data only within the limits of their validity. Since the ranges of validities are not always given, the analyst must use his judgement whether the available data are appropriate for his application or if adjustments are needed.

As a minimum, the analyst should recognise that as the operating conditions become more extreme, the uncertainty associated with the generic data increases.

2.5.3.3 Validation

Most studies, especially those related to less common types of reactors and plants in the first years of their operation must use generic data to establish component parameters. In these cases the data have to be extracted from generic information, and, if possible, from the reference plant, to yield more realistic results.

Once a study has been completed, new data may be used to validate or to update the study. In order to carry out such validations new data confirmed by operational experience or simulator use will contribute to this process of validation.

In the processing of the data one of the major objectives is to set up agreed methods for data authentication and validation; this provides increased confidence in the results of PSA studies. Data must be validated in order to show how methods can be applied to problems of common interest and transferred into standardized procedures.

2.5.3.4 Historical or Improved Data

One of the common problems with the use of the reliability data from the analysts point of view is to decide what reliability data he should use when analyzing future systems (as happens in analysis at the design stage).

Two approaches are possible, namely the use of data from historical databases or of data reflecting technological progress. There are good arguments for both approaches.

For example, the use of historical databases will indicate how well proven technology has been used. Such an approach will enable a direct comparison to be made with other designs based on the same or similar technological developments.

On the other hand technology has advanced and the designers do learn from past mistakes. Thus improvements in availability and reliablity of various components and systems are to be expected. These improvement can be taken into account and targets for improved reliability data can be used, provided that:

- there is evidence that over the years the reliability and availability of components and systems have been improving,
- the observed and documented improvements, rather than postulated (or hoped for) improvements, are cautiously taken into account when setting the targets for improved reliability data,
- targets from improved performance, reliability and availability are challenging, but realistic.

These considerations provide an appropriate and legitimate framework for incorporating technological progress. However, it cannot be over-emphasized how important the above conditions are. If they do not apply and if the targets for various improvements become divorced from reality, the credibility of the whole approach is lost.

3. DATA BASE FORMULATION

3.1 Status and Characteristics of Reliability Data Bases in Member States

A general review of some of the data bases in Member States has been undertaken with two objectives, namely:

- to give general information on the nature, the structure and the content of the different data bases.

- to help in defining the specifications of new data bases.

Not all existing data bases are described here, because some countries were not represented. The participants tried, however, to reflect all the information known to them.

The main data bases described here are:

France: SRDF, SRDF-A, Event File
Federal Republic of Germany: TUV - Norddeutschland E.V.
German Democratic Republic: Reporting System
Great Britain: PR/A, NUPER
Hungary: Generic data bank, Component reliability data bank
Italy: SDE, SEME, PACS
Japan: NSIS, FREEDOM - CREDO
Republic of Korea: PUMAS/N
Spain: DACNE, BDIO, DACNF, BDC
Sweden: ATV
CEC-JRC ISPRA: ERDS/CEDB, ERDS/AORS

The status of the different data bases is described in Tables 3-1.

TABLE 3.1. General information

Country:	FEDERAL REPUBLIC OF GERMANY (FRG)
Name of the Data Base	DCURA
Operator/ Manager	TUV-N
Type of information in DB	Component failure
Main purpose	- review of operational plant experience - reliability parameter comparison actual with reference data
Date of start	1986
No. of monitored comp.s/unit	in principal, all safety components of NPP
Status	in operation
No. of records (-end 1987)	~ 5000
Components/ event	all types (see above)
Types and No. of units concerned	2 BWR 3 PWR
Conditions of information release	confidential

TABLE 3.1. General information (Continued)

Country: FRANCE

Name of the Data Base	SRDF	SRDF A	Event File
Operator/ Manager	EdF/SPT Operat.Dept.	EdF/SPT Operat.Dept.	EdF/SPT Operat.Dept.
Type of information in DB	Component failure	Component failure	Events
Main purpose	PSA maintenance	PSA maintenance	Operating Experience Analysis
Date of Start	1978	1984	1980
No. of monitored comp.s/unit	600	> 500	-
Status	in operation	in operation	in operation
No. of records (end 1987)	1500	2500	20.000
Components/ Events	Electro-mechanical	Electronics	Significant Events
Types and No. of units concerned	34 x 900 MW PWR 10 x 1300 MW PWR	10 x 1300 MW PWR	4 x GCR 34 x 900 MW PWR 10 x 1300 MW PWR 2 x LMFBR
Conditions of information release	to be discussed accord. to EdF rules	to be discussed accord. to EdF rules	to be discussed accord. to EdF rules

TABLE 3.1. General information (Continued)

Country:	German Democratic Republic (GDR)
Name of the Data Base	Reporting System
Manager/ Operator	National Board on Atomic Safety and Radiation Protection
Type of information in DB	Abnormal events of safety related systems
Main purpose	Safety evaluation of NPP
Date of start	1976
Status	in operation
No. of Monitored comp.s/unit	10000 components
Status	In operation
No. of records (end 1987)	Data accumulated in 60 reactor years
Components/ Event	Events from off-normal reports
Type and No. of units concerned	PWR - 5 units - 440 MWe (VVER)
Conditions of information release	Not defined

TABLE 3.1. General information (Continued)

Country: GREAT BRITAIN

Name of the Data Base	Plant Reliability/Availability D.B. - Safety Systems (PR/A)	The Nuclear Plant Event Recording System (NUPER)
Manager/ Operator	CEGB Production Planning Department	CEGB Nuclear Coordination Group
Type of information in DB	Component	Event reports
Main purpose	To gather quantitative information on availability and reliability	To collect and transmit information on safety related events
Date of start	January 1988 for safety system reliability	March 1985
No. of monitored comp.s/unit)	Of the of 500 dpending on the complexity of plant	Not relevant
Status	At pilot scheme stage	Fully operational, but evolving
No. of records (end 1987)	zero	500 event reports, (growing by 5 per year)
Components/ Events	Safety related components in systems	Events of safety significance
Types and No. of units concerned	All CEGB Magnox with concrete pressure vessels, and all CEGB AGRs: 14 units in total	All gas cooled reactors in UK (40 reactors), with significant events from other countries
Conditions of information release	Not yet decided	Confidential, but available* to other utilities with a need to know

*controlled dissemination

TABLE 3.1. General information (Continued)

Country: HUNGARY

Name of the Data Base	Generic D.B.	Load Reduction Report System	Incident and Safety Related Event Report	Component Reliabilty D.B.
Operator/ Manager	Inst. for Electrical Power Research (VEIKI)	NPP PAKS	NPP PAKS	NPP PAKS
Type of information in DB	Initiating event frequency Component reliability data	plant unavail. distribution	events	Failure data of periodically tested comp.
Main purpose	PSA (RA)	Analysis of plant availability	- Report to the authority - Feedback to the operation	Reliability data for PSA
Date of Start	1987	1983	1983	1988
No. of monitored comp.s/unit	(100) component categories (30) initiating events			400 500
Status	From 1989 to be extended by generic common mode and human error data			To the end of this year planned to extend to other components
Components/ events	all types of components			Periodically tested components only
Types and No. of units concerned	n/a	4x400 VVER	4x400 VVER	4 x 400 VVER (PWR)

TABLE 3.1 General information (Continued)

Country: ITALY

Name of the Data Base	SDE	SIMF	PACS
Operator/ Manager	ENEL	ENEA/DISP	ENEA/DISP
Type of information in DB	Abnormal event reports	Abnormal event reports	Reliability data on safety related components and systems
Main purpose	Backfitting and operatioal safety improve ment, design improvement	Backfitting and oper ational safety improvement, design improvement	Backfitting, design improvement, reliability parameter for PSA
Date of Start	1978	1978 CAORSO 1985 TRINO	1989
Status	operational	operational	development
No. of record (end 1987)	2000	2020	Diesel generators reliability data from 1978 to 1987
components/ events	safety related events	safety related events	safety related components and systems
Types and No. of units concerned	BWR CAORSO	BWR CAORSO PWR TRINO	BWR CAORSO PWR TRINO
Conditions of information release	to be agreed with ENEL released to CEC JRC Ispra)	to be agreed with ENEA/DISP (major events reported to NEA and IAEA IRS	

TABLE 3 1. General information (Continued)

Country:	JAPAN
Name of the Data Base	FREEDOM/CREDO FBR Reliability Evaluation Data for Operation and Maintenance Centralized Reliability Data Organization
Manager/ Operator	Power Reactor and Nuclear Fuel Development Corporation Oak Ridge National Laboratory (U.S. DOE)
Type of information in DB	Component
Main purpose	Collection, storage, maintenance, and evaluation of LMBRs components
Date of start	FREEDOM (1985) at PNC CREDO (1978) in the US
No. of Monitored comp.s/unit	21.000 components
Status	in operation
No. of records (end 1987)	21.000 components 1.500 events
Components/ Event	Liquid metal reactor specific or related components
Types and No. of units concerned	3 LMFBRs and several test facilities
Conditions of information Release	limited (available only for PNC and U.S. DOE users)

TABLE 3.1. General information (Continued)

Country:	JAPAN
Name of the Data Base	The Nuclear Power Safety Information
Manager/ Operator	Nuclear Power Safety Information Research Center
Type of information in DB	1. Incidents and failures file 2. Operation file 3. Power station data 4. Annual Data 5. Monthly Data 6. Data on individual energy loss 7. Shutdown data 8. Damage data 9. Overseas information
Main purpose	1. Maintenance of nuclear power safety control information. 2. Analysis and evaluation of information on off-normal events 3. Reliability evaluation of plant systems and facilities. 4. Evaluation of the plant characteristics. 5. Compilation of the pertinent information to be distributed to local municipalities.
Date of start	Oct. 1984, but accumulated data for the past 20 years.
No. of Monitored comp.s/unit	n/a
Status	in operation
No. of records (end 1987)	1000 events
Components/ Event	at present events or off-normal report only
Type and No. of units concerned	1 gas reactor, 33 LWRs (PWRs, BWRs)
Conditions of information release	not yet defined

TABLE 3.1. General information (Continued)

Country:	REPUBLIC OF KOREA
Name of the Data Base	PUMAS/N
Manager/ Operator	Kori no. 1,2 Plant Nuclear Generation Dept.
Type of information in DB	1. Incidents and failures file 2. Operation file, 3. Material Data 4. Outage Data 5. Equipment History Data 6. Event Data 7. Efficiency Data 8. Radiation Protection Data
Main purpose	Management.
Date of start	June 86
No. of Monitored comp.s/unit	10000 components
Status	In operation
No. of records (end 1987)	5000
Components/ Event	all important equipment and events
Type and No. of units concerned	2 (PWR) 600 MWe, 1 (CANDU) 600 MWe 6 (PWR) 900 MWe
Conditions of information release	not yet decided

TABLE 3.1. General information (Continued)

Country: SPAIN

Name of the Data Base	DACNE-BDIO	DACNE-BDC-CEDB
Operator/Manager	UNESA/TECNATOM	UNESA/TECNATOM/JRC Ispra
Type of information in DB	Event	Component
Main purpose	PSA Reporting to CSN	Reliability parameter for PSA Data interchange
Date of start	March 1989	March 1989
No. of monitored comp.s/unit	-	500
No. of Records (end 1987)		—
Status	Development	Development
Components/Events	Events	Mechanical and Electromechanical
Types & no. of units concerned	1 GCR 6 PWR Westinghouse 1 PWR KWU 2 BWR GE	1-GCR 6 PWR Westinghouse 1 PWR KWU 2 BWR GE
Conditions of information release	Utilities and CSN use only	CEDB-rules

TABLE 3.1. General information (Continued)

Country: SWEDEN	
Name of the Data Base	ATV
Operator/ Manager	Vattenfall
Type of information in DB	Component failure
Main purpose	Reliability parameter for PSA Plant availability
Date of start	1976
No. of monitored comp.s/unit	7600
Status	In operation
No. of records records	94263
Components/ event	Components
Types and No. of units concerned	12 LWR(PWR+BWR) 2 Finnish BWR
Conditions of information release	To be discussed with utility

TABLE 3.1. General information (Continued)

Country:	VARIOUS	
Name of the Data Base	ERDS – CEDB	ERDS – AORS
Operator/ Manager	CEC JRC-Ispra	CEC JRC-Ispra
Type of information in DB	Component failure	safety significant event
Main purpose	Reliability parameters for PSA Plant availability Operational safety Maintenance evaluation	PSA Backfitting Design improvement Operational safety
Date of start	1984	1984
No. of monitored comp.s/unit	600	Not relevant
Status	In operation	In operation
No. of records (end 1987)	5200 comp.s. 4200 fail. records	30.000 94263 events (incl. USA LER)
Components/ event	mechanical and electro-mechanical comp.s	Plant abnormal events
Types and No. of units concerned	10 LWR(PWR+BWR) conv.part of GCR	LWR(PWR+BWR) GCR FBR
Conditions of information release	To be discussed with JRC and data suppliers	To be discussed with JRC and data suppliers

As far as the component reliability data bases are concerned, a table with the main information given by the data banks has been prepared.

It includes:

- the type of components monitored;
- the main engineering and operating characteristics recorded;
- the operating tables (number of demands, operating times..).

The results are presented in Tables 3.2.

The completeness of information regarding the engineering characteristics of each component monitored is indicated in the data base, including:

- operating conditions;
- size;
- materials;
- manufacturer;
- preventive maintenance (maintenance practice);
- precautions and limits of use.

The operating data included in the tables (i.e exposure time, operating time and /or number of demands), are essential for obtaining correct reliability parameters This information can be obtained from evaluation or from observation.

In the first case, it is possible to estimate the operational data on the basis of the plant operation or from the number of expected periodical tests. This can lead to an underestimation of the exposure times or of the number of demands.

The second case is preferred. Several means can be used to observe directly the operating times and the number of demands; the best being counters installed on actuators. A very efficient solution is to use the plant computers programmed to record practically hour by hour the operating times and the number of demands on the components monitored.

Table 3.2 Main information items provided

Country: FEDERAL REPUBLIC OF GERMANY (FRG)	
NAME OF THE DATA BASE	DCURA
Types of Components	
Pump	y
Valve	y
Breakers	y
Transformers	y
Heat Exchangers	y
Bus Bars	y
Tanks	y
Motors (electric)	y
Turbine	y
Alternators	y
Engine (Diesels)	y
Batteries	y
Electronics	y
Eng. Charac.	n (future plans)
Size	n
Nature of Fluid	n
Pressure and Temperature	n
Other	
Manufacturer	n
Periodical Tests Description	n
No. of Demands	n
Operation Time	n
Unavailability Time Due to Preventive Maintenance	assessed
Number of Test or Preventive Maintenance Acts	outside DB

Table 3.2 Main information items provided (Continued)

Country: FRANCE

NAME OF THE DATA BASE	SRDF	SRDF A
Types of Components		
Pump	y	
Valve	y	
Breakers	y	
Transformers	y	
Heat Exchangers	y	
Bus Bars	y	
Tanks	y	
Motors (electric)	y	
Turbine	y	
Alternators	y	
Engine (Diesels)	y	
Batteries	y	
Electronics	n	y
Eng. Charac		
Size	y	
Nature of Fluid	y	
Pressure and Temperature	y	
Other		y
Manufacturer	y	y
Periodical Tests Description	y	y
No. of Demands	observed	no
Operation Time	observed	operates permanently
Unavailability Time Due to Preventive Maintenance	not direct	yes
Number of Test or Preventive Maintenance Acts	not direct	not direct

Table 3.2 main information items provided (Continued)

Country: GREAT BRITAIN	
NAME OF THE DATA BASE	Plant Reliability/Availability Data Bank
Types of Components	
Pump	y
Valve	y
Breakers	y
Transformers	y
Heat Exchangers	y
Tanks	y
Motors (electric)	y
Turbine	y
Alternators	n
Engine (Diesels)	y
Batteries	y
Bus Bars	y
Electronics	y
Eng. Charac.	
Size	y
Nature of Fluid	y
Pressure and Temperature	y
Manufacturer	y
Periodical Tests Description	y
No. of Demands	y (only for some component
Operation Time	y
Unavailability Time Due to Preventive Maintenance	y
Number of Test or Preventive Maintenance Acts	y

Table 3.2 Main information items provided (Continued)

Country: ITALY

NAME OF THE DATA BASE	SDE	SEME
Types of Components		
Pump	y	y
Valve		
Breakers	y	y
Transformers		
Heat Exchangers	y	y
Tanks	y	y
Motors (electric)	y	y
Turbine	y*	y*
Main Generator	y*	y*
Engine (Diesels)	y	y
Batteries	y	y
Bus Bars	y	y
Electronics	y	y
Eng. Charac.		
Size	n**	n**
Nature of Fluid	n**	n**
Pressure and Temperature	n**	n**
Manufacturer	n**	n**
Periodical Tests Description	n**	n**
No. of Demands	n**	n**
Operation Time	n**	n**
Unavailability Time Due to Preventive Maintenance	n	n
Number of Test or Preventive Maintenance Acts	n**	n**

*only if safety related parts are affected as consequential event
**available in other plant documentation

Table 3.2 Main information items provided (Continued)

Country: JAPAN

NAME OF THE DATA BASE	FREEDOM	NSIS
Types of Components		(safety related components only)
Pump	y	
Valve	y	
Breakers	y	
Transformers	y	
Heat Exchangers	y	
Tanks	y	
Motors (electric)	y	
Turbine	y	
Alternators	y	
Engine (Diesels)	y	
Batteries	y	
Bus Bars	y	
Electronics	y	
Eng. Charac.		
Size	y	
Nature of Fluid	y	
Pressure and Temperature	y	
Manufacturer	y	
Periodical Tests Description	y	
No. of Demands	y	n
Operation Time	y	n
Unavailability Time Due to Preventive Maintenance	y	n
Number of Test or Preventive Maintenance Acts	y	n

Table 3.2 Main information items provided (Continued)

Country: REPUBLIC OF KOREA	
NAME OF THE DATA BASE	PUMAS/N
Types of Components	
Pump	y
Valve	y
Breakers	y
Transformers	y
Heat Exchangers	y
Tanks	y
Motors (electric)	y
Turbine	y
Alternators	y
Engine (Diesels)	y
Batteries	y
Bus Bars	y
Electronics	y
Eng. Charac.	
Size	y
Nature of Fluid	y
Pressure and Temperature	y
Manufacturer	n
Periodical Tests Description	n
No. of Demands	n
Operation Time	n
Unavailability Time Due to Preventive Maintenance	n
Number of Test or Preventive Maintenance Acts	n

Table 3.2 Main information items provided (Continued)

Country: SPAIN	
NAME OF THE DATA BASE	DACNE-BPC-CEDB
Types of Components	
Pump	y
Valve	y
Breakers	y
Transformers	y
Heat Exchangers	y
Tanks	y
Motors (electric)	y
Turbine	y
Alternators	n
Engine (Diesels)	y
Batteries	y
Bus Bars	n
Electronics	n
Eng. Charac.	
Size	y
Nature of Fluid	y
Pressure and Temperature	y
Manufacturer	n
Periodical Tests Description	n
No. of Demands	observed*
Operation Time	observed.*
Unavailability Time Due to Preventive Maintenance	n
Number of Test or Preventive Maintenance Acts	n

*as much as possible

Table 3.2 Main information items provided (Continued)

Country: SWEDEN	
NAME OF THE DATA BASE	ATV
Types of Components	
Pump	y
Valve	y
Breakers	y
Transformers	y
Heat Exchangers	y
Tanks	y
Motors (electric)	y
Turbine	y
Alternators	y
Engine (Diesels)	y
Batteries	y
Bus Bars	y
Electronics	y
Eng. Charac.	
Size	y
Nature of Fluid	y
Pressure and Temperature	y
Manufacturer	y
Periodical Tests Description	y
No. of Demands	y
Operation Time	y
Unavailability Time Due to Preventive Maintenance	y
Number of Test or Preventive Maintenance Acts	y

Table 3.2 Main information items provided (Continued)

Country: VARIOUS

NAME OF THE DATA BASE	CEDB
Types of Components	
Pump	y
Valve	y
Breakers	y
Transformers	y
Heat Exchangers	y
Bus Bars	y
Tanks	y
Motors (electric)	y
Turbine	y
Alternators	y
Engine (Diesels)	y
Batteries	y
Electronics	n
Eng. Charac.	
Size	y
Nature of Fluid	y
Pressure and Temperature	y
Manufacturer	not available
Periodical Tests Description	Planned for future
No. of Demands	y
Operation Time	y
Unavailability Time Due to Preventive Maintenance	Not observed
Number of Test or Preventive Maintenance Acts	no prev. maint recorded(*)

* Average values, estimated by operators, will ble given (future improvements)

3.2 Component failure description

In this section the basic information necessary for an adequate failure description is listed and commented on. In addition, the special requirements for PSA use are specified.

Table 3.3 correlates the information collected for failure description and its use, primarily from the point of view of the safety analyst.

Table 3.3. Description of information to be collected and relation to the safety analysis use

FAILURE DESCRIPTION	INTEREST
date of failure detection	for studying time to failure
date of start of the repair	for assessing average time before repair starts
date of restart to operation (end of tagging-out of the component)	for assessing actual repair time
duration of the failed (or unavailable) state (from the failure occurrence detection until the restart)	duration of the degraded performance or loss of performance for assessing total downtime (unavailability) of component
duration of repair	gives a useful indication of technical aspects of repair (useful for maintenance purpose)
manpower for repair, radiation exposure	mainly for the optimization of maintenance team activity
plant status at failure detection (e.g. in operation, during maintenance or shut-down for refuelling)	evaluation of the impact of the function loss or degradation on the plant safety. Failures during shutdown are not usually taken into account for PSA purpose

Table 3.3 (Continued)

FAILURE DESCRIPTION cont'd	INTEREST
plant status during the unavailability time	impact on plant safety
system status at failure detection (e.g. one train in operation in a two train system, components sharing load)	provide data for calculating conditional probabilities or for state transition probabilities in Markov models
effect by which the failure is observed (failure mode in the CFDB classification)	
I) Failures occurrs while component is operating	
a) degree of suddenness of the component failure [The component is suddenly unavailable or the unavailability can be deferred.] * immediate (e.g. catastrophic failure) * progressive (e.g. incipient failure)	for characterising whether the component failure initiation is prompt or gradual (in the latter case credit can be given sometimes for emergency intervention)
b) degree of seriousness (of the degradation of the component function) e.g.: -complete loss -partial loss -no consequence on the function (minor imperfection or the repair can be made while maintaining the component available)	it is correlated to the previous (a) and at the same level of importance

Table 3.3. (Continued)

FAILURE DESCRIPTION cont'd	INTEREST
II) Failures occurrs on demand (including testing) * complete failure to start * it starts, but with a certain delay or the required performance is not reached	in some cases this is not a catastrophic failure and should not be considered as such in emergencies.
description of the failure mechanism (failure mode in SRDF classification which has a failure mode matrix in accordance with each component type)	important according to the degree of detail of the modelling used in the PSA: it also helps in identifying the possibility of recovery actions. The descriptors of the failure mechanism are useful information items to understand what has really occurred with the component
causes of failure (immediate and root causes) It reveals (identifies) the root cause of the failure (design, manufacturing, installation, operator error or maintenance error, abnormal service/ environment conditions; failure induced by external influence).	from the "cause" analyst understands if it is spontaneous (random) failure (component hardware fault) or caused by misoperation or by external influence.
parts failed: it characterizes the parts of the component involved in the failure and in repair	it is of primary importance for the maintenance aspects
effect of the comp. failure on the plant; on the system it belongs to or on other system/components	it is of interest mainly for operational safety evaluations. it can provide some useful information in case of dependencies between failures
common cause failure identification (related failures)	it allows for assessing probabilities of common cause failures

Table 3 3. (Continued)

FAILURE DESCRIPTION (cont'd)	INTEREST
corrective actions taken or planned (incl. human factors)	it describes the repair action and it indicates the need for modifications
administrative actions taken	it specifies the repair schedule; consequences on plant operation schedule, on failure reporting to safety authority. Important to the plant management.
number of demands at the time of the failure	to calculate reliability parameters for particular components. it is also used to verify the adequacy of constant unavailability on demand model, to study variation of failure probability on demand component ageing, etc.
number of operating hours at the time of the failure	to calculate reliability parameters for particular components. it is used to verify the adequacy of a constant failure rate model, to study failure rate time dependency, component ageing, etc.
Free (uncoded) description* proposed minimum information: plant and system status description of what happened, direct causes, consequences	it helps the data collector to better describe what occurred. it is useful for the safety analyst who examines and possibly discards some failures from the set before making the statistics.
reference to similar events in the same plant or similar plants	

* a list of keywords is convenient for easy retrievals

3.3 Modes of Data Collection

Several modes of data collection can be used according to the objectives and the resources available to the data base manager.

Tables 3.4 indicate the modes of data collection in various Member States. Some options have been included in the list as possible modes despite their present limited use.

Table 3.4 Modes of data collection in several Data Bases

Country: FEDERAL REPUBLIC OF GERMANY (FRG)	
DATA BASE	DCURA
MODES OF DATA COLLECTION	
Technician(s) on site in charge of data collection	-
Maintenance tech. in addition to their current activities	-
Operation tech. in addition to their current activities	-
Automatic Data Collection (plant computers, maint. comp.)	-
Specialists (PSA analysts) send on site for limited mission	-
Investigations by questionnaires	-
Others (to be specified)	TUV specialist in charge of L.A.

Table 3.4 Modes of data collection in several Data Bases (Continued)

Country: FRANCE

DATA BASE	SRDF	SRDF A	Event File
MODES OF DATA COLLECTION			
Technician(s) on site in charge of data collection	x	-	x
Maintenance tech. in addition to their current activities		x	-
Operation tech. in addition to their current activities			-
Automatic Data Collection (plant computers, maint. comp.)	on study	N4 type plants	-
Specialists (PSA analysts) send on site for limited mission	for PSA project		
Investigations by questionnaires	n	n	n
Others (to be specified)		-	-

Table 3.4 Modes of data collection in several Data Bases (Continued)

Country: GREAT BRITAIN

DATA BASE	Plant Reliability/ Availability D.B. Safety System	NUPER
MODES OF DATA COLLECTION		
Technician(s) on site in charge of data collection		–
Maintenance tech. in addition to their current activities	x	–
Operation tech. in addition to their current activities	x	–
Automatic Data Collection (plant computers, maint. comp.)	(in the future)	–
Specialists (PRA analysts) sent on site for limited mission	-	–
Investigations by questionnaires		–
Others (to be specified)	x (Specialist information engineer)	x

Table 3.4 Modes of data collection in several Data Bases (Continued)

Country: HUNGARY

DATA BASE	
MODES OF DATA COLLECTION	
Technician(s) on site in charge of data collection	
Maintenance tech. in addition to their current activities	-
Operation tech. in addition to their current activities	x
Automatic Data Collection (plant computers, maint. comp.)	—
Specialists (PRA analysts) send on site for limited mission	-
Investigations by questionnaires	--
Others (to be specified)	

Table 3 4 Modes of data collection in several Data Bases (Continued)

Country: ITALY

DATA BASE	SDE	SEME
MODES OF DATA COLLECITON		
Technician(s) on site in charge of data collection		–
Maintenance tech. in addition to their current activities	x	–
Operation tech. in addition to their current activities	x	–
Automatic Data Collection (plant computers, maint comp.)	-	–
Specialists (PRA analysts) send on site for limited mission	–	x
Investigations by questionnaires	-	–
Others (to be specified)	–	reported by the utility and revised

Table 3.4 Modes of data collection in several Data Bases (Continued)

Country: JAPAN

DATA BASE	FREEDOM	NSIS
MODES OF DATA COLLECTION		
Technician(s) on site in charge of data collection	x	-
Maintenance tech. in addition to their current activities	x	--
Operation tech. in addition to their current activities		-
Automatic Data Collection (plant computers, maint. comp.)		-
Specialists (PRA analysts) send on site for limited mission		-
Investigations by questionnaires		--
Others (to be specified)		By head office or safety department of utilities

Table 3.4 Modes of data collection in several Data Bases (Continued)

Country: REPUBLIC OF KOREA

DATA BASE	PUMAS/N
MODES OF DATA COLLECTION	
Technician(s) on site in charge of data collection	X (2/unit)
Maintenance tech. in addition to their current activities	
Operation tech. in addition to their current activities	4 computer specialists/unit
Automatic Data Collection (plant computers maint. comp.)	terminal, automatic collection (plant network)
Specialists (PRA analysts) send on site for limited mission	10
Investigations by questionnaires	
Others (to be specified)	-

Table 3.4 Modes of data collection in several Data Bases (Continued)

Country: SWEDEN

DATA BASE	AIV
MODES OF DATA COLLECTION	
Technician(s) on site in charge of data collection	x
Maintenance tech. in addition to their current activities	–
Operation tech. in addition to their current activities	-
Automatic Data Collection (plant computers, maint. comp.)	–
Specialists (PRA analysts) send on site for limited mission	periodically
Investigations by questionnaires	periodically
Others (to be specified)	

Table 3.4 Modes of data collection in several Data Bases (Continued)

Country: VARIOUS

DATA BASE	CEDB
MODES OF DATA COLLECTION	
Technician(s) on site in charge of data collection	
Maintenance tech. in addition to their current activities	
Operation tech. in addition to their current activities	
Automatic Data Collection (plant computers, maint. comp.)	
Specialists (PRA analysts) send on site for limited mission	some JRC specialists preferred several data collection campaigns
Investigations by questionnaires	-
Others (to be specified)	assistance given for training in data collection

3.4 Structure and Organization for Data Collection

Data collection requires considerable effort at different levels. While the overall responsibility for the data base typically lies with a specified group of people at the headquarters of the utility, the ultimate quality of the job is strongly dependent on the personnel actually collecting data.

The prime mechanism for enhancing the quality of data is a daily follow up of plant activities. This can only be accomplished by a person well integrated in the operating team and able both to detect the failures of components of interest and to record the information required.

While the detection of failures can be done solely from maintenance work orders, reliability reports will require some additional information for which specific training of personnel is needed. Dedicated of a person for this task is thus recommended.

An extremely important point to be aware of is the fact that any loss of information which is not detected on site will afterwards be very difficult, if not impossible, to recover. That makes it almost essential to include a first level review by some plant staff member to ensure completeness of the records and to detect some of the quality problems. This can be accomplished by Q.A. people on site.

People at the headquarters should also provide some spot checking of the information with the main purpose of maintaining consistency between different plants and to ensure that the objectives of the data base are being satisfied by the actual data supplied. These people will normally rely on computer verification of the information, and might also be able to cross check between event and component data banks.

Tables 3.5 describes the manpower and computer resources allocated in some data bases.

Table 3.5 Manpower and computer resources used in data collection

Country: FEDERAL REPUBLIC OF GERMANY (FRG)

DATA BASE		DCURA
MEANS ASSOCIATED WITH DATA COLLECTION		
On Site	Manpower	–
	Computers	--
At corporate level	Manpower	3 engineers
	Computers	IBM PC

Table 3.5 Manpower and computer resources used in data collection (Continued)

Country: FRANCE

DATA BASE		SRDF	SRDF A	Event File
MEANS ASSOCIATED WITH DATA COLLECTION				
On Site	Manpower	2 tech./4 units + 0.5 eng./4 units	0.5 tech./2 units	
	Computers	Personal Computer IBM PC/AT type	Personal comp. IBM PC type	Personal comp
At corporate level	Manpower	2 engineers	0.5 engineers	> 2 engineers
	Computers	Mainframe	Mainframe	Mainframe

Table 3.5 Manpower and computer resources used in data collection (Continued)

Country: GREAT BRITAIN

DATA BASE		Plant Reliability/Availability DB Safety System	NUPER
MEANS ASSOCIATED WITH DATA COLLECTION			
On Site	Manpower	1 (part time) for 2 units but with assistance	1 (part time)
	Computers	No (direct VDU input planned ~2 years)	VDU connected to remote mainframe
At corporate level	Manpower	~2	3
	Computers	General purpose mainframe	

Table 3.5 Manpower and computer resources used in data collection (Continued)

Country: HUNGARY

DATA BANK		Component Reliability D.B.
MEANS ASSOCIATED WITH DATA COLLECTION		
On Site	Manpower	1 part-time engineer operators filling form
	Computers	IBM PC network
At corporate level	Manpower	1 part-time engineer
	Computers	

Table 3.5 Manpower and computer resources used in data collection (Continued)

Country: ITALY

DATA BASE		SDE	SEME
MEANS ASSOCIATED WITH DATA COLLECTION			
On Site	Manpower	about 10	1 (integration of data received from utility)
	Computers	Mainframe	
At corporate level	Manpower	1 (data revision)	1 (data revision)
	Computers	Mainframe	Mainframe IBM-PC

Table 3.5 Manpower and computer resources used in data collection (Continued)

Country: JAPAN

DATA BASE		FREEDOM	NPSI
MEANS ASSOCIATED WITH DATA COLLECTION			
On Site	Manpower	More than 300 man months at PNC (unknown for the US)	
	Computers	FACOM M380 at PNC	
At corporate level	Manpower		about 30
	Computers	FACOM M380 at PNC	-

Table 3.5 Manpower and computer resources used in data collection (Continued)

Country: REPUBLIC OF KOREA

DATA BASE		PUMAS/N
MEANS ASSOCIATED WITH DATA COLLECTION		
On Site	Manpower	10
	Computers	Westinghouse H-8000 on site
At corporate level	Manpower	—
	Computers	-

Table 3.5 Manpower and computer resources used in data collection (Continued)

Country. SWEDEN

DATA BASE		AIV
MEANS ASSOCIATED WITH DATA COLLECTION		
On Site	Manpower	1 contact person/unit
	Computers	mainframe terminal (UNISYS) IBM, VAX
At corporate level	Manpower	2 engineers
	Computers	UNISYS mainframe

Table 3.5 Manpower and computer resources used in data collection (Continued)

Country: VARIOUS

DATA BASE		CEDB	AORS
MEANS ASSOCIATED WITH DATA COLLECTION			
On Site	Manpower		
	Computers		
At corporate level	Manpower	1 engineer + 3 technicians 1,5 computer engineers	1,5 engineers +2 technicians (*) 0.5 computer engineer
	Computers	AMDAHL 470/V8	

* for day to-day maintenance and control only. For a continuous operation with a data in-flow of 3000 event reports per year, 8 engineers and 5 technicians would be necessary. This high effort is due to the special pre analysis and pre-processing of data before storage.

3.5 Quality Assurance (QA) in Data Collection, Validation and Screening

QA is of utmost importance in any data collection system.

The first step in data collection is obviously to define clearly and unambiguously what is to be collected. Detailed procedures and clearly set out forms should prevent incompleteness or misinterpretation of data. Furthermore, people collecting data should continually be made aware of the importance of their work and periodically retrained.

Responsibility and interfaces should also be clearly defined and periodically confirmed. Verification of all collected data at plant level should be recommended when the possibility still exists to revise data instead of relying on memory. At this stage data should be verified for completeness and content according to well defined validation procedures. Technicians can assure good data collection, but engineers should be involved in verification and validation activities. Validation should mostly be based on a clear view of the scope of the activity concerned and of the final utilization of the data.

Knowledge of the characteristics of components and of their functions in the plant and in-field experience helps to avoid misinterpretation of data and in validation.

At corporate level an independent verification is recommended, but only on a sample basis. At this level a background reliability analysis and quality control are necessary for a successful review.

Validation would benefit an effective graphical representation and "trend and pattern" analysis, aimed at discovering wrong, incoherent or incomplete reporting in some areas. Finally, cross checking of the collected data and comparison with results of generic data bases is also helpful.

Data should be screened not only for consistency and content, but also for possible negative tendencies to allow prompt backfitting actions when necessary. Computers should be used in connection with automatic techniques in the processes of data verification and validation.

Data verification in the collection phase should be employed mostly to check that the collected data are consistent with the objectives of the data collection program and analysis.

There are two main mechanisms for the verification of the data:

- Q.A. programme (set up beforehand)
- cross-check verification (carried out afterwards)

For the first mechanism, a written procedure should be specified in order to define the target and acceptance criteria. The reporting should be mandatory for everything that is required, with no optional items. This is needed to prevent minimum reporting becoming the norm. Likewise, definitions should not be so broad that the data supplier has trouble deciding in which category to report Checklists are one means to enhance the quality of this phase. Periodic meetings between everyone involved in data collection and analysis are recommended in order to give everyone a clear idea of the overall work. It is very important that data suppliers interact on a regular basis with the data base managers or even with the users, and make use of the data base themselves. Otherwise, there is a tendency to lose sight of the purpose of the work and ultimately there is likely to be a reduction in the quality of the data supplied. Frequent retraining of all parties is also necessary.

Finally, the plant personnel responsible for collecting the raw data should ensure that they are free from ambiguities which could cause misunderstandings.

For the second mechanism, if many failures are observed on one type of component, the performance of similar components in different systems could be compared. If the performance is similar, the reporting is confirmed. If the performance of this component is different, it might be because of different conditions of operation of the other system; or alternatively, the reporting might be incorrect, or different procedures might have been used. Similarly verification could be made through comparison with generic data bases on the same component from other plants.

3.6 Characteristics of Computer Systems for Data Transmission and Storage

According to the size of the utility and the objectives of the data base, several means can be used to collect, to transmit and to store the data. Two distinguished set-ups are:

- Centralized organization with computer network, and
  local organization

The more classical solution (and probably the more efficient) is the use of a mainframe computer at corporate level connected in a network with local microcomputers for file interrogation by the users.

Tables 3.6 summarize means of data acquisition, transmission and storage as well as means of release of data to the users in several established data bases.

Table 3.6 Means of data manipulation

Country: FEDERAL REPUBLIC OF GERMANY (FRG)

DATA BASE	DCURA
Data acquisition	on PC-screen
Data transmission	-
Data storage	IBM-PC Diskettes Tape
Data release to users	Reports
Software used for data treatment and retrieval	DCURA (dBase III)

Table 3 6 Means of data manipulation

Country: FRANCE

DATA BASE	SRDF	SRDF A	Event File
Data acquisition	direct on Terminal CRT	direct on Terminal CRT	direct on Terminal CRT
Data transmission	network	network	network
Data storage	IBM 3080 Mainframe	IBM 3080 Mainframe	IBM 3080 Bull DPS 7 Mainframe
Data release to users	network and diskettes	network and diskettes	network and diskettes
Software package for data treatment and retrieval	FdF soft and Dbase 2	FdF soft and Dbase 2	FdF soft and Dbase 2

Table 3.6 Means of data manipulation (Continued)

Country: GREAT BRITAIN

DATA BASE	Plant Reliability/ Availability DB Safety Systems	NUPER
Data acquisition	Forms	Direct on screen
Data transmission	post to computing centre, then onto magnetic tape	private line
Data storage	IBM 3081 Mainframe Disk	IBM 3081 Mainframe Disk
Data release to users	Network	Network
Software package for data treatment and retrieval		

Table 3.6 Means of data manipulation (Continued)

Country: HUNGARY

DATA BASE	Load Reduction Report Systems Component Reliability D.B.
Data acquisition	Load reduction report directly on computer; failure data manually on written forms (computerization in progress
Data transmission	IBM PC network in written form
Data storage	IBM PC streamer in written form
Data release to users	in plant by IBM PC network; outside in written form
Software package for data treatment and retrieval	

Table 3.6 Means of data manipulation (Continued)

Country: ITALY

DATA BASE	SDE	SIME
Data acquisition	written form	
Data transmission	magnetic tape	
Data storage	Mainframe	Mainframe (Olivetti, Hitachi) Personal computer IBM
Data release to users	Magnetic tape local network (inside utility organization)	Local Network (inside ENEA/DISP organization), diskettes
Software package for data treatment and retrieval	Ad hoc software for sequential files treatment	IBM STAIRS text search (Datamet)

Table 3.6 Means of data manipulation (Continued)

Country: JAPAN

DATA BASE	NPSI
Data acquisition	From utilities to MITI by written reports and telefax
Data transmission	Terminals are connected by telephone lines
Data storage	Magnetic disk, tape, optical, disk
Data release to users	on line is official use only JAPEIC and MITI
Software used for data treatment and software	-

Table 3.6 Means of data manipulation (Continued)

Country: JAPAN

DATA BASE	FREEDOM
Data acquisition	Written forms
Data transmission	Postage (partially computer network)
Data storage	FACOM M380 (Japan) IBM 3033 (U S)
Data release to users (by network diskettes, listings)	listings or network
Software used for data treatment and retrieval	-

Table 3.6 Means of data manipulation (Continued)

Country: REPUBLIC OF KOREA

DATA BASE	PUMAS/N
Data acquisition	Written forms
Data transmission	At present, each unit has one computer
Data storage	Westinghouse N 8000, 2500
Data release to users	Written reports and CRT
Software used for data treatment and retrieval	KEPCO soft

Table 3.6 Means of data manipulation (Continued)

Country: SPAIN

DATA BASE	DACNE BDIO	DACNE BDC
Data acquisition	written forms and screens	written forms and screens
Data transmission	diskettes and modem	diskettes and modem
Data storage	personal computers	personal computers and JRC Ispra computer
Data release to users	diskettes and listings	network
Software used for data treatment and retrieval	personal computer software	personal computer software and ADABAS

Table 3.6 Means of data manipulation (Continued)

Country: SWEDEN

DATA BASE	AIV
Data acquisition	Direct on terminal CRT
Data transmission	Network, tapes
Data storage	UNISYS 1190
Data release to users	Network, printouts, special topic report
Software package for data treatment and retrival	DMS 1100

Table 3.6 Means of data manipulation (Continued)

Country: VARIOUS

DATA BASE	CEDB
Data acquisition	The data collector can use filled in forms or an informatic support (magnetic tape, diskette) to send data to JRC-Ispra
Data transmission	JRC computer network, telecommunication network, post, diskettes
Data storage	AMDAHL 470/V8 (IBM compatible) JRC-Ispra mainframe, running under OS/MVS, DBMS ADABAS (Software A.G.)
Data release to users	by network, diskettes, printouts, etc.
Software package for data treatment and retrieval	- ADABAS with an enhanced version of ADASCRIPT (user-oriented software from Software A.G.) - Other software developed by J.R.C.

3.7 Potential Applications of the Data Bases

The importance of data on reliability analysis has already been discussed in Chapter 2. At this point, it should be emphasized that one of the most obvious and direct applications of both the component and event Data Banks is to provide the necessary means for information exchange between different plants. A good information gathering and retrieval system is the basis for a sound analysis of the observed malfunctions and an effective exchange of information.

Table 3.7 further outlines some additional applications of data bases

Some applications are mainly of interest to utilities, others mainly to regulators, general industry or research organizations. Waste of time and resources could be avoided by considering various needs when establishing data base characteristics. In addition, the general methodology used to collect and assess data in the nuclear field could be effectively used in activities other than nuclear. In this respect, mutual exchange of data sources between the nuclear and the conventional field could bring mutual benefits. Different design, manufacturing and operation aspects should be, however, properly considered. The CEDB, developed by JRC Ispra, already contains data on components of the power conversion systems of conventional power plants

TABLE 3.7. APPLICATION OF DATA BANKS

APPLICATION \ DATA BANK	PLANT EVENT DATA BANKS: AORS	EDF EVENT FILE	DACNE BDIO	Hungary Inc. D.B.	SDE	SEME	NUPER	COMPONENT EVENT DATA BANKS: CEDB	SRDF	SRDF-A	DACNE BDIO	Hungary Comp. D.B.	PACS	CEGB CRDS	ATV	FRG TUV	FREEDOM CREDO
PSA																	
Initiating events	X	X		X	X	X	q										
Failure data																	
– component	X	X			X	X		X	X	X	X	X	X	X	X		
– human errors	X	X	X	X	X	X		X	X	X			X		X		
– CCF	X	X	X		X	X		o	o	–			X				
– repair time					–	X		X	X	X	X	X	X	X	X		
Maintenance																	
– optimization of maintenance period					–			X	X								
– test intervals opt.					X	X		X	X								
– spare parts manag.					–			X	X	X							
– plant outages manag.					–			X	X								
– component ageing					X	X		X	X					X			
Operation																	
(real time) exchange of information	+	X	X		–		+	+	X	X	+	+		+			
plant performance comparison	X	X		X	–												
incident prevention	X	X		X	X	X	X	X	X					X			

X – included
o – not directly
+ – not real time
q – only qualitative

3.8 The Component Event Data Bank (CEDB) of the European Reliability Data System (ERDS)

One of the most widely known data bases containing component events is the Component Event Data Bank managed by the EEC JRC Ispra. Its size in the sense of the number of records and of the number of countries participating requires that one chapter of this document be devoted to CEDB only.

The CEDB is a centralized bank, managed by the CEC-JRC Ispra, collecting operational data (failure reports, annual operating times/demands) of components of NPPs operating in various European countries. The bank's main objectives are to supply data suitable for reliability/availability evaluations and the promotion of operational safety.

The bank receives data either from national data bases, such as the EDF/SRDF and the VATTENFALL ATV, in their original format (and language) or directly from some NPPs (in the CEDB format).

Where necessary information is expressed and homogenized in a common structure (i.e. the CEDB structure) through a computer aided transcoding process and translated to a single language (English). The main peculiar features of the bank are its classification scheme, data retrieval capabilities and on line statistical processing programmes. The bank can be interrogated both by JRC internal users, through the JRC T.P. network, and by external users through national telecommunication networks.

As of December 1987, the CEDB contains data from about 5200 components (well identified by their engineering and operational characteristics), pertaining to 21 component types (mechanical and electromechanical equipment pieces) and 51 engineering systems, monitored for an average time of 5 years in 10 LWR units and in the steam-water cycles of units of other reactor types and of conventional power units. These units are located in seven European countries (the number of components monitored in each unit is about 600). The failure events reported are about 4200.

3.9 Current Problem Areas

Countries which have developed their own data bases have often addressed the problem in different ways and with different ideas of how the information is to be used. New data collectors can benefit from the experience already gained, and avoid repetition of some of the initial difficulties. IAEA assistance is desirable in co ordinating this activity. The problems for every new data collector will be different to some degree. In countries in which there is no plant in operation but only a plant under construction, there is more time to gain information and experience concerning data collection methods, computerization of data collection and storage, etc. After a systematic examination of international experience, the means (manpower, computer techniques, etc) and structure of the future data bank can be determined. For countries with plants in operation but with no reliability data banks established there is not much time to plan concepts, to develop the computer codes and then to start collecting data. The data collection might well be started before the data bank structure is fully defined, in which case it will be necessary to record manually every detail that is available in order to leave open all options. The information would then be included in the data bank retrospectively, when its structure had been finalized In such cases data collection and the development of the bank itself may go in parallel.

As for the relative convenience of continuing an ongoing but limited reliability data collection system or starting a new more comprehensive one, the following remarks can be made:

- it is necessary to ensure continuous monitoring of the performance of the equipment of a plant in order to control ageing problems and to monitor maintenance effectiveness and the correct application of operating procedures;

- for new plants the use of more sophisticated equipment technically requires a more sophisticated reliability data acquisition system.

remarkable differences are often identified in the reliability performance of apparently very similar equipment. This requires thorough analysis of all the factors that might influence such performance, these analyses are greatly assisted by the availability of detailed in field data

For a component data base, the two main difficulties are to ensure the coherence and homogeneity of the data collection between the different plants; and the completeness of the data collection

The first problem can be solved by giving the data collectors on site simple and precise criteria to define failures. Increasing the use of existing plant computers used for maintenance records, tagging out, and process control are other ways to give the collector additional means to catch failures. Encouraging contact between safety or PSA analysts and data collectors on site, and organizing joint meetings for the data collectors from each of the plants in order to exchange information and to discuss difficulties and practices, are additional means of ensuring coherence and homogeneity in data collection

The second problem can be handled by the same approach, but in addition great emphasis must be placed on motivating data collectors to seek the pertinent details of the failures and to ensure that the associated descriptions are comprehensive.

4. DATA MANAGEMENT AND ANALYSIS

4.1 Computerized Data Base Management Systems

Reliability databases have a tendency to become very large after a few years of reporting by several power stations at the component level. The size of such databases demands that state of the art software for information handling be used. There are examples of databases which are only a few years old but which suffer from the considerable inefficiency of overnight batch processing for even the simplest enquiry. This should be avoided in any new system, by giving thought at the design stage to system upgrades. Interactive user friendly access is most desirable.

Another consideration is to build in flexibility and spare capacity at the design stage of the data base. The requirements usually change with time and even essential items can be overlooked at the beginning. A system that cannot accommodate such contingencies will have to be abandoned; possibly with the loss of much useful and expensively obtained data. When computer network is being used for input and retrieval of data, message facilities between users at different locations are very desirable.

In most cases, only part of the information contained in the event data base is needed in the calculation of the reliability parameters. The interface problem could be solved by assigning the same identification codes for the corresponding information in the event data base as well as in the reliability parameter data base. However, there will be instances when it is not possible to use the same identification codes. In such a situation, it is necessary to develop an identification code translation routine; this is fairly straightforward in most circumstances.

In the specific case of the utilization of the data base retrieval and manipulation system in a PSA study, the minimum information required by it from the collected event data base system are the component type, the failure mode, and the associated information on component reliability. As a next step, another computerised system can be used to generate the appropriate input for the specific PSA application which can be, for instance, the generation of numerical values to be used for quantifying basic events in the fault trees.

The database, be it reliability or event type, needs to be searched according to the requirements of the users. User's requirement are difficult to include in some existing data handling systems, but is less of a problem if modern user-friendly information software or the expert system interface is used. There are databases in existence which take a long time for the novice to learn how to use, and are cumbersome even for experts, and thus should be avoided.

4.2 Data Base Security

The first stage in ensuring the security of the data base is a clear definition of who can have access to the data, both within the organization and outside. Even then, the norm for such access should be "read only". Where there is a requirement for derived or altered data then this should be achieved by the use of a data base which is separated from the primary one, and under the control of the user.

The methods needed to maintain the security of the data will differ depending on the data base support. An administrator of the data is an obvious requirement, in order that a centralized responsibility is clearly defined. The administrator should be responsible for:

- avoiding unauthorized manipulation or inadvertent destruction of the data base;
- ensuring that data cannot physically be lost by such means as fire, computer crash, or other "hardware" or human activity. For this purpose at least one backup copy of the data file should be made and stored in a controlled area, as is normal computer practice;
- ensuring release of confidential data only to authorized people and organizations;
- being the only one that is allowed to access the file for writing.

For computerized data bases, passwords to access the data base are an obvious requirement and should be regularly changed; for written data bases, a distribution list should be defined, with the level of confidentiality indicated.

Finally, because of the high cost of setting up and maintaining a data base it is advisable to avoid any unnecessary limitation of the "read only" access to the data by external organizations in order to maximize the benefit to everyone, but taking into account, obviously, any sensitive commercial aspects, and ensuring proper utilization of released data.

4.3 Statistical Treatment to Generate Usable Parameters

4.3.1 Statistical Treatment of Data

Data Bases can provide users with a wide body of information on the results of the operation of systems, components and piece parts (part of components). This information must be processed in a suitable manner before it can be used for different purposes such as PSA, special reliability studies, planning of maintenance activities, optimization of plant features such as availability or reliability.

The treatment of data must be performed through the application of appropriate statistical techniques However, these statistical techniques must be carefully selected and applied according to the characteristic of the data acquired and to the requirements of the ongoing analysis.

Point and interval estimation, test of hypothesis and data analysis techniques like discriminant function analysis and correlation analysis are mentioned in the field of classical statistical approach. Bayesian approach (parametric and non-parametric) is also applied despite some criticisms that have periodically arisen, concerning the practical applications of this statistical approach in deriving failure and reliability parameters.

4.3.2 Assessment of Running Times and Number of Demands

In addition to the component failure events which are recorded in event data bases, other operational information like number of demands and operational time is essential to derive usable reliability parameters.

In the absence of fully computerized maintenance records and/or process control computers, the running times of rotating components, or the number of starts experienced by them, is going to be very labour intensive or even sometimes impossible to acquire. In some instances it will be possible to fill in a significant part of the needed information by application of logical algorithms to the data base, providing certain rules are obeyed.

A requirement to input exposure times or running times for all safety system components would increase the task of the data gatherers at the stations by an order of magnitude. Thus what is needed is a clear statement to the data gatherers of all of the default options. For instance, a running reactor implies that a large proportion of the components in the safety systems are in a clearly defined state, unless it is reported otherwise. These lists of assumed plant configurations should cover all likely operating states, and the data collector at the station will need to be familiar with them.

The number of demands experienced by certain components are much more difficult to obtain, if not impossible, unless of course they are put in explicitly. Any number derived is likely to be an underestimate This problem is solved by having dedicated counters on selected components, or by recording demands on plant computer.

4.3.3 Bayesian Data Updating

The Bayesian approach can be used for deriving reliability parameter for systems, components or parts of components. Such reliability parameters could be failure rates, failure on demand probabilities, maintenance and repair rates, etc.

Bayes' theorem is used to derive an a posteriori probability distribution of these parameters from an a priori probability distribution and related data which can be used as evidence (Ref. 2). The Bayesian approach can be applied in the so-called one step methodology as well as in the two step methodology (Ref. 3)(Ref. 4). The information from data bases can be used to assess the a priori probability function as well as the likelihood function. For example, parameters defining the reliability of systems or components pertaining to PSA of specific plant can be considered as a priori information. Plant specific data bases are also the main source of data on failure behaviour of systems and component. Information from different data bases, or the information derived by aggregation of data from different plants (or other sources) can also be used to assess the a priori probability distribution for some of the reliability parameters.

When the information has been obtained from expert opinions, the methodology of pooling this information have to be analyzed with care. Principally it must be assured that the dependencies, or bias among these experts, have been identified and removed

Data from the data bases can be provided in different forms, namely:

a) set of pair of values, for each component, of the number of demands and corresponding number of failures that occurred.

b) set of pair of values, for each component, of the total time in operation (or in test) and corresponding number of failures that occurred.

c) set of values for components or system maintenance duration and/or repair times.

Once the a priori probability distribution and the likelihood function has been obtained, the a posteriori probability function is calculated.

According to the type of a priori functions, the a posteriori probability can be assessed through analytical or numerical methods. If the a priori probability function is the conjugate distribution to the likelihood function, then the calculation of the a posteriori distribution can be performed analytically, because the a posteriori distribution will be of the same type as that of the a priori function. When this is not the case, numerical approximations can be used for practical purposes One of the methodologies currently used is the discretization method (Ref. 5)

4.3.4 Combining Data From Various Sources

The lack of sufficient and/or relevant data on component failure rates or failure probabilities is a problem often encountered in PSA studies.

Nowadays there are quite a few reliability data sources available. The power plant utilities, suppliers, and national as well as international organizations collect operating experiences from both nuclear and conventional plants or other industrial or military facilities. In order to broaden the population base and the number of failures recorded, data bases are sometimes combined Some methods to combine data sources, which consider uncertainties in the parameters have been published. Some of them are also discussed in the PRA Procedures Guide (Ref. 6) and its references. The methods have been applied in some of the PSA's as well as in reliability data books (Ref. 7). The methods are mainly based on empirical Bayesian methods. Subjective methods for combining data from various sources are presented in Ref. 8 and Ref. 9. (The phrase "subjective method" is used to emphasize that the estimates or the distributions obtained by using such a method are not completely based on statistical observations.)

The use of data from various sources can be appropriate when a PSA is being performed for an installation which is in the design, construction or initial operational phases, and consequently where plant specific data are not yet available or are unreliable because of their sparseness.

Reference 6 presents some of the pooling methods. This methods require that for each source both a point and an interval estimate for the failure rate is provided. Furthermore, it is assumed that the data sources are statistically independent and of equal importance A "consensus" estimate for the failure rate is then obtained by means of simple geometric averaging techniques.

If the data sources are not of equal importance, a weighted geometric mean can be used with weights chosen to reflect the importance of each source The resultant maximum likelihood consensus point estimate is then a weighted geometric mean of the individual estimates. The weights assigned to each source could be simple functions of the uncertainty bounds of particular data source (Ref. 6).

Another method presented in the PRA Procedures Guide (Ref. 6) is the so called "mixture method". It involves fitting a suitable prior distribution to each generic source and then combining the individual prior distributions by forming a mixture. Several methods are suggested for determining the weights for the prior distributions in Ref. 6. If the mixture is used as a prior distribution, the corresponding posterior distribution will also be a mixture of the individual posterior distributions with the new (updated) weights.

In the subjective Bayesian method (Ref. 3, 4) the data from various sources are used as the evidence data in the likelihood function of the Bayesian equation, and the usability (or relevance) of the data from each source is evaluated It is assumed that the usability of the data from each source, to be used as evidence for the specific case being analysed, can be interpreted in a probabilistic way This feature enable combination of distinct data from various sources.

The subjective probabilities of the weights can be, in principle, determined rather arbitrarily to describe the analyst's confidence in the usability of the data from each source.

If the data sources come from different nuclear power plants, the weights can be interpreted as indices to describe the similarity between these plants and the specific plant being analysed (Ref. 9). One way to obtain such indices is to apply the analytic hierachy process (AHP), which is a systematic procedure for representing elements of any problem; in particular, the decision problem hierarchically (Ref 10).

4.4 Expert Opinion

4.4.1 Artificial Intelligence Applied to Data Bases Enquiry and Treatment of Reliability Parameters Derived from Expert Judgement by Making Use of Fuzzy Sets and Possibility Theory

Literature sources very often give component reliability parameters which were derived from expert judgement For instance, some of the reliability data given by IEEE Std 500 were produced through the aggregation of estimates of experts collected following the Delphi method For a given component and for each failure mode, IEEE gives a recommended value for the failure rate, a low (optimistic) value, a high (pessimistic) value These above-mentioned failure values were obtained through a geometric average of the corresponding values given by each expert

The recommended and limit values given for each failure rate by IEEE can be interpreted as identifying a fuzzy set of failure rate values (a fuzzy set with the associated membership function). Fuzzy sets, possibility theory and fuzzy logic (Ref. 11)are effective mathematic tools which are particularly suitable to deal with the human way of thinking, expressing judgements and taking decisions.

The CEC-JRC Ispra, in collaboration with the Paul Sabatier University (Toulouse, France) are studying the possibility of developing a Reliability Parameter Data Bank (RPDB) by exploiting fuzzy logic. According to this approach, the future RPDB should consist of an intelligent interface capable of interrogating on line data bases representing the IEEE and other similar reliability sources including, the CEDB. It should also combine the answers obtained from the various sources to generate a unique answer. The request can also be expressed by making use of fuzzy attributes for the identification of the component type and its mode of failure.

As a first step the JRC and the Paul Sabatier University are developing an intelligent interface capable of understanding a request containing some fuzzy expressions, and of translating this request into a sintax developed for interrogating the IEEE source. More precisely, it would be useful for retrieving and processing the relevant data contained in a data base representing the IEEE tables and giving as an answer a failure rate expressed as a fuzzy set.

In this first application only one component type, the electric motor family, are considered. Some new and complex problems (such as dealing with reliability parameters expressed by fuzzy sets on one hand and reliability parameters expressed by probability distributions on the other) are still to be investigated at the theoretical level before they are considered as viable for developing a RPDB.

4.4.2 Aggregation of Expert Opinion

As the statistical data on operating experience are not always available, or are available in a non compatible form, it is necessary to use expert opinion as the source of credible reliability parameters.

The expert opinion is usually collected from individuals familiar with operating and failure history or even with design and manufacturing of particular component or generic types of component. To avoid substantial errors in estimation of a reliability parameter by a single expert, or small group of experts, several methods were developed to collect and evaluate expert opinion. In addition, dealing with a larger group of experts requires the establishment of a feedback system for the written exchange of data and information among the group. A widely known method for this purpose is the

Delphi method, successfully used in IEEE Standard 500 and in a number of other cases.

Regarding the credibility of a data source compiled from expert opinion it is obvious that compilation of more expertise (larger number of knowledgeable, independent experts) guarantees better results.

If the data sources are ranked in accordance with the methodology used to gather or to aggregate expert opinions the following options exist:

1. Consensus (larger group of knowledgeable experts is required)
2. Nominal group technique (similar to Delphi)
3. Delphi procedure (used in IEEE 500)
4. Aggregate individuals (statistical aggregation)
5. Individual judgement
6. Absolute probability judgement

4.5 Reliability Parameter Data Bases

4.5.1 Information on Reliability Data Bases

A number of data bases for component reliability parameters are available in the literature. Of most importance for use in PSA's are data bases which were compiled from nuclear experience or from experts with knowledge in the nuclear area.

Reliability parameters found in PSA studies are either plant specific (derived directly from plant specific operating experience); generic scenarios, updated with plant specific experience or data coming from outside the plant but assesssed as being applicable to particular components.

There are examples of single plant PSA data bases which were derived from operating experience directly or generic data updated with plant specific operating experience. These include Oconee PSA, Zion PSA, Ringhals PSA.

There are also several data bases compiled for single plant PSA studies in which the reliability parameters mostly rely on general nuclear, industrial and military experience, or on experts opinion. This approach is usual when the PSA is being performed on a pre operational phase, or early in plant lifetime when the operational experience is limited. Examples of this category includes Shoreham PSA, Sizewell B PSA and the German Risk Study.

When a PSA programme with several studies is being performed, usually a single data base is used for several plants (although some plant specific data is also used). These types of data bases usually draw parameters either from a group of nuclear plants, from expert opinion or from non-nuclear sources like industrial or military experience. Data bases of this kind are found in the U.S.A. in IREP data base (NUREG 2728), NREP data base (NUREG 2815), and ASEP data base (NUREG-4550, Vol.1).

It is important to mention the French approach where data used in a PSA study currently being performed comes from a collection of operating experience from many identical plants. Having such wide operating experience for various components, PSA studies can be carried out using operational data only.

The most widely used source of data from expert opinion is IEEE Standard 500, (1977 and 1984 editions) which contain the reliability parameters for a large number of NPP components. The reliability parameters compiled from only nuclear experience are found in several NUREG reports (1205 for pumps, 1362 for valves, 1363 for DGs) and are based completely on analysis of raw data from LER reports.

Several other NUREG reports or industry publications are also available presenting parameters derived from the operating experience of a limited group of plants and which consider LER's, plant internal documentation or information provided by the plant personnel.

Outside the US, the Swedish Reliability Data Book is a widely known source, compiling data from operational experience of 7 Swedish PWRs.

There are a number of industrial and military sources available, most of which cover a very specific area of application. Some of the better known are: Military handbook 217 E, a widely used source for electronic component; UKAEA SRD operates a large data base consisting mainly of industrial data. As the reliability is extremely important in off-shore operations, a great deal of data on off-shore components was collected and reliability parameters were published in the OREDA Handbook.

The aerospace industry dealt with reliability problems for a number of years, and it has produced very comprehensive data sources, although rarely applicable to nuclear components

Military sources provide reliability parameters on many components some of which are not used in nuclear plants, but data on common components like motors, diesels, pumps, etc., are usually applicable. As the military data bases usually incorporate on relatively large amounts of operating experience, sometimes it is wise to consult this source as well. An example of a military data source is the Non electric Parts Reliability Data (NPRD-3), 1985, Reliability Analysis Center (RAC) Rome Air Development Center.

A comparison of several reliability parameter data bases are given in the following tables 4.1.

Table 4.1 Comparison of main characteristics of data sources

DATA SOURCE / Characteristics	PSA Study generic data updated with plant operation experience (e.g. /ion NPP)	P S A Study operating experience only
no. of component types	35	60
level of detail in component description (no. of sub-divisions of general component type)	limited	moderate to high
component boundary definition	n/a	n/a
component operating mode (o m)	n/a (directly assessed from system o.m.)	n/a (directly assessed from system o m)
component operating environment	n/a (sometimes assessed from location)	usually n/a (sometimes assessed from location)
no of failure modes defined per component	1-3	1-5
reliability parameters	mean value; variance;	mean value distribution confidence intervals
repair time	sometimes available	available
ultimate source of data	generic prior, updated with plant specific operating experience	operating experience only
other relevant information		

Table 4.1 (Continued)

	Programme of PSA Studies		Nuclear Experience LERs
	IREP	NUREG 1205	Sweden I Book
no. of component types	35	10	50
level of detail in component description (no. of sub divisions of general component type)	limited	moderate	moderate to high
component boundary definition	n/a	yes	yes
component operating mode (o.m.)	n/a	yes	sometimes defined
component operating environment	defined for a few compon. only	not directly defined	not directly defined
no. of failure modes defined per component	1-3	2-4	1-3
reliability parameters	mean, median error factor	mean, distribution	mean, upper bound
repair time	n/a	n/a	yes
ultimate source of data	assessed from several different sources incl. expert opinion	event reports from US plants	event reports; data collection system; plant sources
other relevant information			

Table 4.1 (Continued)

	compilation from different sources	
	WASH 1400	IEEE 500
no. of component types	30	>200
level of detail in component description (no. of sub-divisions of general component type)	limited	relatively high (vary with component types)
component boundary definition	no	no
component operating mode	n/a directly	sometimes defined
component operating environment	defined for few components	multipliers available for most components
no. of failure modes defined per component	1 3	1 9
reliability parameters	median; distribution; confidence interval; error factor	recommended value, maximum and minimum of assessed sources
repair time	no	sometimes available
ultimate source of data	assessed from variety of sources	assessed from nuclear industrial and military sources and aggregation of expert opinion
other relevant information		

Table 4.1 (Continued)

	Military data Rome Air Development Centre (Reliability Analysis Center)	
	MILHDBK217D	NPRDS 3
no. of component types	>1000	100
level of detail in component description no. of sub-divisions	high	limited
component boundary definition	n/a	n/a
component operating mode	n/a	n/a
component operating environment	yes	yes
no. of failure modes defined per component	1-5	1-3
reliability parameters	mean value	mean, 5%, 95%
repair time	n/a	n/a
ultimate source of data	field data	field data
other relevant information		

4.5.2 CEDB Reliability Data Book (RDB)

Reliability data derived by a statistical processing of the primary data stored in the CEDB are being collected in the CEDB RDB. This book is a collection of tables of component reliability data; these tables are considered to be an organized output of the base. Updating is made by an "ad hoc" informatic programme.

The RDB has been conceived as an easy to use tool for the analyst in safety and reliability/availability studies. By merely consulting the book, without accessing the base, he can quickly obtain reliability data for the item he is interested in, and at the same time associated information on the sample size from which these data were derived can be seen. This associated information enables the analyst to evaluate the credibility of the reliability data given in the relevant tables for any particular item.

The book structure is described next:

The component is identified on the basis of a code specifying five component features:

- country, from which the raw data originates
- plant type, to which the raw data refers
- plant engineering system, to which the component pertains
- component family (i.e. type) such as "pump"
- 1st component engineering characteristic (for a pump it could be the combination type/medium handled, e.g. centrifugal/water)
- 2nd component engineering characteristic: (e.g. operating pressure range)
- 3rd component engineering characteristic (e.g. operating flow range)

Some background data for the statistics (i.e. data characterising the sample observed) are given. For each failure mode, number of items observed, cumulative operating time and/or number of demands, number of components which had failures, number of failures occurred, cumulative observation calendar time, etc. is given.

Information on the component application/function performed, operating mode, external environment, test/maintenance; boundary specification (by a definition and a sketch).

For each type of failure during operation, the Mean Time Between Failure (MTBF) and the related standard deviation (STD) are given; in the case of failure time, the distribution can be assumed to be exponential (the test of exponentiality, provided by the standard CEDB data treatment procedure, will have been satisfied); the failure rate best estimate, its STD and the 90% symmetrical confidence interval are given in addition. The same applies for failures on demand

Repair data are also given: background data for the statistics (number of repaired items, number of repair, etc.) Mean Time To Repair (MTTR) its STD, and minimum and maximum repair time experienced.

Comment: the choice of the three engineering features characterizing the item is to be made on the bases of previous analysis of the influence of each attribute on the component behaviour.

4.5.3 IAEA Generic Component Reliability Data Base

A compilation of the published component reliability parameters was undertaken at the IAEA to facilitate the use of a computer code package for fault tree and event tree analysis (currently under development at the IAEA named PSAPACK and to facilitate a comparison of the various data bases available in the literature.

Currently the IAEA generic reliability data base consists of about 1000 records, providing reliability parameters for 450 different components, grouped in about 100 component types. Each record provides as much information regarding respective components as possible (limited by the source of information), including operating mode and environment, repair time, source and ultimate source, component boundary definition and all appropriate comments found in the original source.

For the generic data base a common record form was developed, and also the unique 5 alphanumeric characters coding system, characterizing component types, failure modes and sources of information. The data base was developed for use on an IBM compatible PC computer using the dBASE III software.

The IAEA generic data base includes reliability parameters from 21 sources including data from PSA studies, industry reports, and the major data sources available from Europe and the USA.

4.6 Special Studies to Evaluate Data for Initiating Events (IE)

Initiating event frequencies are needed in PSA's. In many cases the operating experience (LER's) cannot be used directly without some sorting, grouping or modification. Therefore, determination of IE frequency requires special studies on the raw data, which may be either generic (Ref. 12) or plant specific (Ref. 13).

Generic studies of initiating events have been performed to determine frequency of anticipated transients (Refs. 14-15). In this case two lists of anticipated transients for BWR and PWR were used, and each LER in the basic event data base was allocated to one of these IE categories. Thus IE frequencies for each separate category were obtained. For this evaluation to be effective very well defined event descriptions are needed for each one of the IE categories.

Another example is the development of the methodology to quantify the Loss of Offsite Power (LOSP) frequency. The earlier studies (Refs. 16-17) used LER's almost directly The only treatment made was the grouping of the LER's into geographical regions. However, a more recent study (Ref. 18) has further examined the entire data base of LOSP from LER's. They were put in categories based on geography related features as well as plant specific features, such as number of incoming power supplies from the grid; number of switchyards; number of transformers; number of switchgear buses and the availability of "black start" capability. This approach provided factors based on operating experience which could be used with the above plant-specific features to obtain a more closely tailored frequency of LOSP for any type of plant and according to its specific design.

For PSA analysis of LOSP, the recovery time after LOSP is also of great importance. The same special studies discussed above derived recovery times on the basis of the LER's for each of the above features. It is obvious that geographical features (severe weather conditions) had longer recovery times than plant specific features such as failure in the switchyard.

A third example of special studies for deriving data for IE frequency determination is the study of LOCA IE frequency. Such a recent study is given in Ref. 19. In this study LER's up to December 1984 incorporating 800 reactor years were analysed. This study considered specifically three pipe sizes, two leak rates, and different systems (LOCA sensitive and non-LOCA sensitive). It provided failure rates for pipes for the above categories based on the LERs survey. A similar special study of pipe break LERs, was performed for Seabrook PRA and facilitated the evaluation of the frequency of interfacing LOCA (Ref. 20).

In the Oconee PRA a special analysis was made and cutsets that lead to loss of instrument air were identified in the plant fault trees and then quantified to obtain the frequency of this low frequency IE. This is also commonly used to evaluate other frequencies of IEs which are cases of support system failure.

In summary, raw data of LER type or generic failure data of components are needed to estimate IE frequencies required for PSA studies.

4.7 Special Studies to Evaluate Reliability Data and Reliability Indicators

4.7.1 Generation of Safety System Reliability Indicators

Reliability data are often used as an expedite approach to keep track of the reliability of safety systems and components.

One of the ways component event data can be used is in conjunction with safety system reliability indicators. It involves the condensation of the large amount of information on fault and maintenance duration of safety system components, into a few quantitative parameters These can then be compared with the various design and license conditions in order to see if any remedial action is required. It might be that equipment has to be replaced, or that the practice and frequency of maintenance, needs to be altered. This activity is in no way a replacement for a comprehensive PSA study; it is rather something that can be accomplished quickly and frequently as part of the surveillance activity.

The data that is needed consists essentially of the availabilities of the components of the various safety systems. Unavailability will be due to faults, maintenance, and testing. The simplest indicator will thus be the time fraction that one or more components of the safety system were out of action, for whatever reason. This safety system partial unavailability indicator suffers from a total lack of precision on the degree of impairment;

there is no distinction between a fault that completely disables the safety system, and one that is inconsequential.

An assessment can be made, however, at the time of the fault, or unavailability, on the degree of impairment of the relevant safety system, taking into account the redundancy inherent in such systems. This is subjective and raises the question of who is competent to make such an assessment in an impartial manner. However, it has the advantage of being very simple in that only one extra number needs to be recorded into the data base for each event. The indicator to be produced is just the time fraction weighted by the assessed impairment for each unavailability period. The result is a rough quantification of the degree of impairment, but the treatment of redundancy is weak.

The next level of refinement is to take into account the fact that there are many occasions when a safety system does not need to be entirely functional, such as when the reactor is shut down. Indeed, maintenance and testing activities will deliberately be scheduled for such periods. This is quite easy provided the power level of the reactor, or the times when it is critical, is known to the data base. This gives the actual safety system unavailability. It is, however, more meaningful in the case of failures, other than from maintenance and testing activities, to derive the indicator taking no account of any fortuitious non requirement for failed components, on the grounds that failures will occur randomly.

The most refined safety system availability indicators will be those that take account of the configuration of the components. This is the only rigorous way of allowing for redundancy and multiple overlapping failures or unavailabilities. This requires some kind of system model; not in the same level of detail as those used in PSAs, but at least accurate down to the component level. The algorithm being used to generate the indicator will need to search the data base for the availability record of all components in each system; the default being that the component was available. Obviously, the subjective assessment of the degree of impairment of the safety system is now redundant, but there is great advantage in replacing it with an estimate of the degree of impairment of the component when the data is put in. This is a much less demanding task than previously as no account of redundancy is needed. The indicator produced will now be a genuine measure of the degree of impairment of the safety system.

The final requirement might be to distinguish the various causes of unavailability by producing an indicator similar to that described above, but for each possible cause, namely: component failure, testing, maintenance, human error (if this is logged) etc. This refinement would obviously be of great value to the station management.

Another concept for the treatment of reliability data to reflect each country's nuclear power performance is introduced in a paper presented during the meeting.

4.8 Data Base Network for Reliability Improvement

Improvements in availability, maintainability and reliability are of utmost importance in nuclear power plant safety. PSA studies indicate that importance of improving the reliability of safety system and their support systems as well as other non safety equipment which if failed can cause turbine trips, reactor scrams and other plant disturbances.

The most important information that are needed for reliability improvement are the operating experience, accident reports and material analyses. However, one of the most difficult problems encountered in data analysis is the lack of a consistent procedure to correlate symptoms, trends and actual failure probabilities.

Table 4.2 compares some of the features of several event data bases.

Table 4.2: Comparison of International Event Data Base

NAME	PRIS	IRS	ERDS	USERS
Organisation	IAEA	NEA (OECD)	EC	UNIPEDE
Users	Participants	Participants	Partic. (USA and Sweden incl.)	Participants
Report Term	1/year	Actual accidents	Actual abnormal events	Actual accident
Start Date	1970	1979	1979	1985
Coverage	90% of all plants in the world	80% of all plants in the world	About 130 plants for AORS	138 plants
Contents	- Unit Basic Data - Outage Significance - Causes of Outage	- Significant Incident	- Component Events DB - Abnormal Plant Occurrences - Unit Status - Reliability Parameters	- Significant Events
Limitations	- Above 100 MV - Outage >100 hrs.	- Incident		- All types event - Outage >3 hrs.
Reports	- Annual performance reports - Outage Statistics (Pie Charts)	- Analysis Report	- Event Report	- Event Description
Advantages	- World Record Comparison	- Similar Accident Correction Actions	- Broad analysis possibilities	

The combined experience of various nuclear power plants provide a much richer source of information for availability and maintainability improvement than that of any single plant. A computer network connecting NPPs around the world greatly facilitate this exchange of experience

The primary objective of that network would be to identify generic problems in component quality, recurring failures, and to some extent, the quality of nuclear plant management, as shown in the flowchart in figure 4 1.

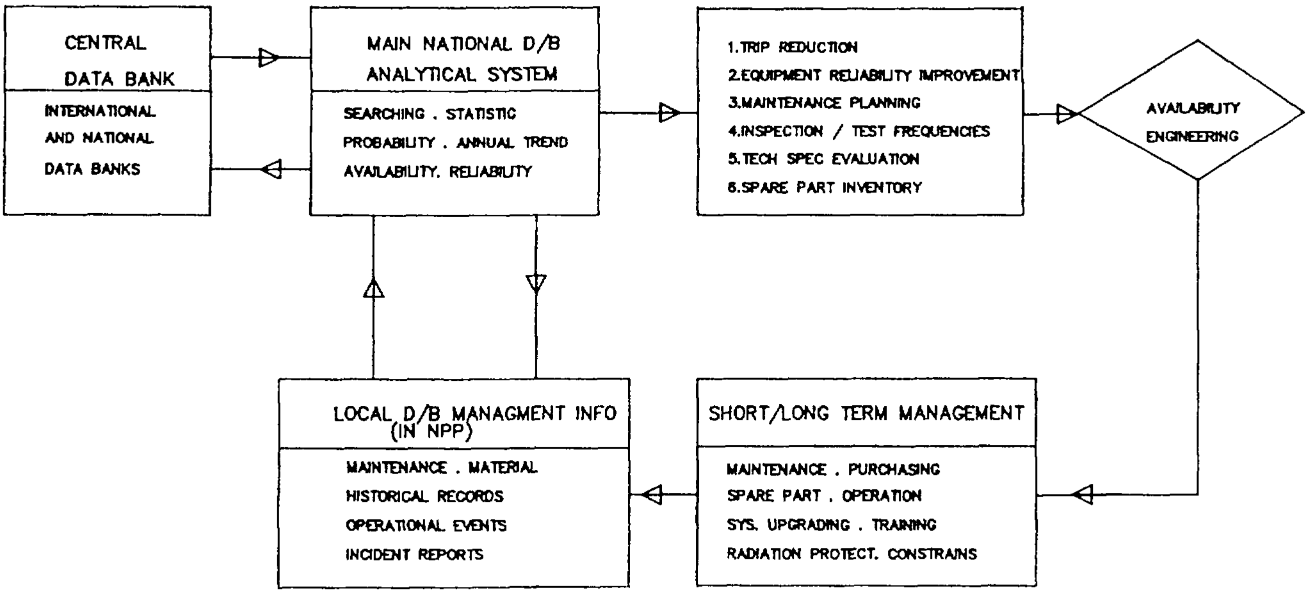

FIG. 4.1. Suggested D/B flowchart for reliability improvement.

In this framework a more detailed exchange of experience including mechanical, electrical and instrumentation, design aspects, failures, spare parts, tools, engineering, operation history, training, procedures, outage reports, analysis programming, radiation protection and plant management. For effective exchange of information between plants, the International Data Bank should consider:

- Common Data Bank Format
- International Data Bank Network including PRIS, IRS, FRDS, USFRS and others banks.
- On line communication system between the Data Bank and the nuclear power plants

The operational predictions based on the operating experience from NPPs in one country is limited, except for those countries, which have nuclear plants in service for a long time. Operational data collected internationally with high level of aggregation (not raw data), are in principle easily obtainable through the INPO, NEA, EC and UNIPEDE, but access is limited to members of these organizations. Furthermore, these libraries are not yet interconnected and event reports have to be sent or interrogated separatetely on each of these library systems.

Regarding the present situation, possibilities for a systematic exchange of information among countries are limited in the near future due to the lack of a common language, the differences in evaluating environmental conditions, the variety of reactor types, etc.

Various international data bases which collect operational events in NPPs are in operation. Among them are:

PRIS: IAEA
USERS: UNIPEDE
IRS: NEA (OECD)
ERDS: EC

ABBREVIATIONS USED IN THIS SECTION

PRIS	:	Power Reactor Information Systems
IAEA	:	International Atomic Energy Agency
IRS	:	Incident Reporting System
NEA	:	Nuclear Energy Agency (OECD)
ERDS	:	European Reliability Data System
EC	:	European Community
USERS	:	UNIPEDE Significant Event Reporting System
UNIPEDE	:	Union International des Producteurs et Distributeurs d'Energie Electrique
INPO:	:	Institute of Nuclear Power Operation

4.9 Current Problem Areas

a) The special requirements which are placed on information systems for data handling and storage are:

- capability of handling large amounts of data
- easy to use
- feasibility for further development;

b) In the area of data verification special attention should be given to:

- standardised component identification coding system;
- standarised method for checking the credibility of raw data;

c) Every data base collecting raw data should be revised continuously to reflect the actual nuclear power plant operating experience.

The next subsections raise several additional problem areas in more detail.

4.9.1 Information Barrier   The need for a Communications Network System

In recent years, there has been a marked increase in the interest in improving the reliability of nuclear power plants. PSA and other similar analysis techniques have been developed and data from NPP operating experience and engineering analysis were extensively used. They are usually used to predict the frequency of certain events like core damage accidents including those caused by equipment malfunctions or operator errors.

To generate or predict a maintenance management program from the available operational experience is difficult due to the lack of hardware reliability information and inherent uncertainties. Some international organizations (IAEA, EC, UNIPEDE) have Data Banks, tailored to specific purposes. The access to raw data collected here is, however, rather limited.

The need exists to establish a good communication network between plants, utilities, national as well as international data bases free of any barriers (language, interpretation). International Organizations like IAEA could contribute to the establishment of a good data base network for effective sharing of experience between countries.

4.9.2 Problem areas connected with use of generic data bases

When using generic data base there are several areas where misinterpretation of the data supplied by generic sources can occur. This results sometimes in unacceptable variation in component reliability parameters. Four general problem areas have been identified in the following order of importance:

- component boundary definition
- failure mode definition
- operating mode definition
- operating environment definition

Component boundary definition is the main source of misinterpretation. There are no firm rules or even similarities in the way various sources deal with this problem. Most sources do not even provide detailed definitions of component boundaries; sometimes components are only defined as "off the shelf" items. Inclusion or exclusion of, for example, the driver on the pump boundary, can sometimes change reliability parameters in orders of magnitude

There are three major interfaces to be specified for an unambiguous component boundary definition, namely:

- mechanical interface;
- power supply interface;
- control system interface.

Failure mode is another problem area where difficulties are caused by lack of a common nomenclature. One way to solve the problem is to define generic failure modes, as it was done in the IAEA Generic Data Base compilation.

Operating mode is of importance for active components like pumps. There are three distinct categories:

- standby
- alternating
- running (continuously operating)

Each of the above can influence the reliability parameter substantially. For components such as valves, normally only two operating modes, open or closed, needs to be considered.

Component environment can also change reliability parameters substantially (2-5 times). The main environmental factors which can influence reliability are:

- humidity
- temperature
- pressure
- radiation

The operating environment is not usually mentioned in sources. This can lead to errors when calculating the system reliability in the environment of an accident.

4.9.3 Problem areas connected with combination of data sources

When several data sources are to be combined to obtain reliability parameters one has to be aware of the possibility of them not being independent. As there are a limited number of ultimate data sources (either experience or experts) it is almost always the case that at least part of the data base relies on the same population of components, even when the data sources are PSA studies performed separately in different countries. A solution for limiting the problem is the assignment of proper weighting factors to the different sources, after the ultimate source has been investigated and fully identified.

5. FUTURE TRENDS

In reviewing problem areas in data collection, analysis and retrieval the group of participants in the TCM have identified the following aspects as deserving special attention by those involved in collecting and using reliability data. Some of the issues highlighted require further investigation and coordination of research in an international framework (e.g. IAEA's Coordinated Research Programme on Data Collection and Analysis for PSA).

- Data collected in recent years have been done in a more systematic way and are therefore to be preferred for use in reliability studies. Historical records contain important evidence to be incorporated in data bases. It is however very time consuming to dig out information in plant records and to extract the required data. In reality, the effort does not always lead to usable results because of the way records have been originally established.
- Operating or exposure times and number of demands are in many cases estimated to compensate for the lack of actual records. This procedure can lead to gross errors
- Failure data collected does not reflect, in many cases, hidden dependencies between parts or components. It should be clearly stated in the data base, which data have been gathered assuming that failure occurred independently.
- Indication of the root causes of failures are an important source of information which can be used to unfold dependencies.
- The assumption that failure rates collected during tests of short duration are representative of longer operating times can be highly conservative, because failure rates are normally higher in the initial period of operation. Some preliminary studies indicate a factor of 10 difference. Further work is needed in this area
- Data collection is an expensive undertaking. It is therefore of utmost importance that plant managers are convinced of the benefits of this activity. Managers have to be given enough information on how collected data can be used to improve plant safety and availability. PSA is only one of the possible uses of reliability data.
- A close liaison between plant personnel, data collectors and data users is necessary. Meetings or workshops can be an effective way of promoting this interface.

Manpower requirements for data collection are in general very difficult to assess and can be misleading. For example, in some cases the actual number of people stated to be involved in running the system can be low because the involvement of the plant personnel needed is not reflected. Different data bases may have different structures depending on their specific objectives. It is, however, important that common requirements are established to allow for communication between data bases.

It is furthermore recommended that IAEA will use its international role to lead an effort for constructing an international network of national and international data bases

- Standardization of information in data bases is highly desirable for sharing of international experience. Aspects which will require standardization include, for example, the failure description. Countries embarking on reliability data collection should take advantage of the available international experience and should attempt to establish standard structure and content in the planning phase of their data bases.

  The aerospace and military industries report their failure records to a centralized data base. The available information is then shared among users for various purposes, including an alert system in cases when action has to be taken to prevent reccurrence of a reported failure. An extension of some of the existing systems into the nuclear power field, should be considered.

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Appendix

PAPERS PRESENTED AT THE MEETING

RELIABILITY DATA AND SAFETY EVALUATION OF NUCLEAR POWER PLANTS — STATUS AND TRENDS IN THE GERMAN DEMOCRATIC REPUBLIC

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Abstract

Nuclear safety depends strongly on the knowledge about faulty conditions of components and systems. Therefore a systematic approach is necessary to get comprehensive safety-relevant informations.
System analysis and the computation of processes governing design basic accidents and severe accidents are the main tasks of such an approach. The solution of problems covered by these tasks are strongly connected.
System analysis covers structure investigations and methods of reliability analysis, which are the main points within the framework of probabilistic safety analysis.
Collection, analysis and retrieval of reliability data had been found to be a field of increasing importance to safety analysis. Therefore both constructor and operational organizations in the GDR are legally bound to collect and analyse reliability data as well as to make changes of the technology of a NPP if data indicate a necessity.
An independent system of data collection has been established by the National Board on Atomic Safety and Radiation Protection. Current data bases of the GDR involve generic data as well as plant specific reliability informations.
The paper presents the general approach used in the GDR to introduce reliability data into licensing decisions.
Status and plans of data manipulation are discussed as well as the structure of data bases.
In addition experiences gathered in using reliability data are presented. Some of them are the following:

i) At the present status of reliability analysis and probabilistic safety analysis respectively generic data play an important role.
ii) Plant specific data bases are to be improved to get more sophisticated probabilistic saftty informations. Special attention has to be paid on human errors and common mode failures.
iii) Engineering judgement is the most important point in manipulating reliability data and using them in probabilistic safety analysis.

1. Introduction

Since the very beginning of commercial use of NPP in the GDR reliability data have been considered to have a great influence on safety evaluation. Therefore the evaluation of reliability data has become an integrated part of every safety evaluation of a NPP.
Reliability of a component or a system is defined as the probability of performing a specified function under given conditions for a given time interval. Defining reliability in that way gathering and analysing of quality data become a very difficult work.
Reliability data must as well as possibly specify the statistic population which is to be taken into the probabilistic consideration. Therefore reliability data bases must cover a wide range of properties of components and systems as well as environmental conditions which may influence failures.
Data aspects are carried by special regulations of the Regulatory Body of the GDR. Laws and national standards contain general requirements on safety evaluations. /1/,/2/,/3/.
Guides and special decisions of the National Board on Atomic Safety and Radiation Protection give exact informations on how to perform safety evaluations in a prescribed manner.
The main part of safety evaluation is established by accident analysis. Their results are the base for performing all measures of nuclear safety.
To get possible initial events and sequences of accidents operating experiences of NPP must be gathered and analysed.
According to /2/ institutions incolved in implementing NPP in the GDR are legally bound to collect and analyse operating experiences as well as to make changes of existing technology if data indicate a necessety.
In the 20 years of using commercial NPP in the GDR a data base has been established containing experiences of roughly 60 years of reactor operation.

2. Reliability Data

Two main tasks of safety evaluation must be distinguished by their ways of solving problems:

i) accident analysis which is to be done under given initial and boundary conditions and

ii) system analysis which is to define possible initial and boundary conditions.

An accident analysis contains calculations of accident sequences taking into consideration general physical laws. It covers both deterministic and probabilistic results. Generally computer-aided calculations are the main parts of an accident analysis.

There is a general approach used in probabilistic system analysis which neglects uncertainties of the accident analysis compared with possible changes of initial and boundary conditions such as failures of safety relevant systems.

That means a functioning component (or system) will fulfill its given function under given conditions.

Therefore the performance of reliable accident analysis is one of the main conditions to be successfull in probabilistic safety analysis. As a result special requirements must be fulfilled by an accident analysis.

For requirements on computer codes used in licensing in the GDR see e.g. /4/.

System analysis has to determine the behavior of a NPP under possible conditions using a describing approach.

There are no general physical laws to determine system failures from former conditions. Failures of every system under consideration must be calculated from operational experiences gathered over a long time period. Supposing that conditions won't change by time the operational experiences may be assumed to be valid also in future. That is a general assumption of PSA. Besides there isn't a large number of elements with identical properties to get well defined statistical results. That means there won't be samples of a poor statistical population which can be used in the system analysis. Therefore a system analysis must contain a wide range of informations including plant specific as well as generic data and engineering judgement.

For that reasons a probabilistic safety analysis is a more subjective approach.

A system analysis covers two main tasks:

i) structure analysis and

ii) reliability analysis.

In both fields methodical investigations and developments have been done for some years in the GDR, e.g. /5/,/6/.

Figure 1 shows the structure of safety system analysis as it was established by the Regulatory Body of the GDR.

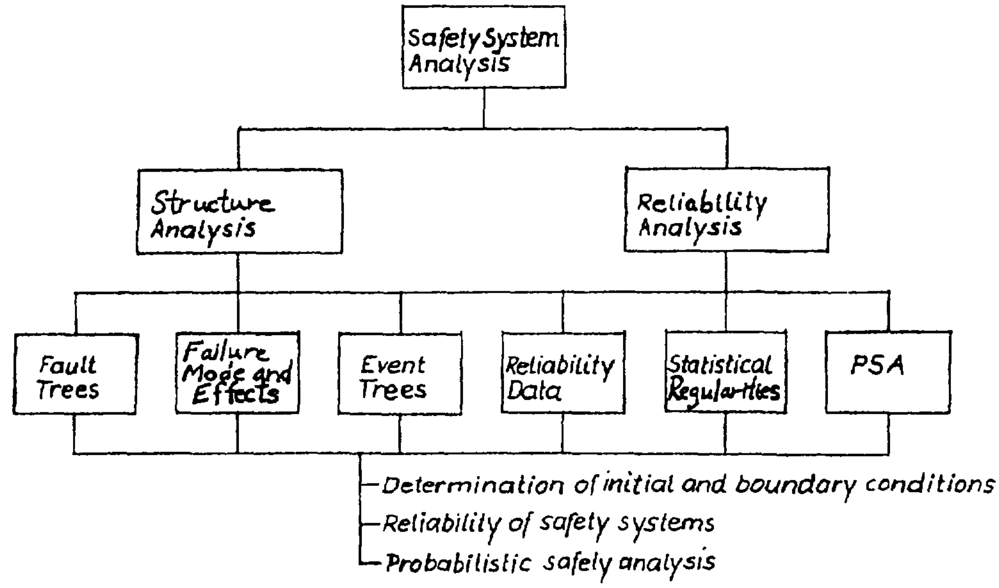

FIG.1. Structure of the safety system analysis of the Regulatory Body of the GDR.

Data bases have been established by both operating organizations and the Regulatory Body. Figure 2 shows some characteristic features of reliability data base of the Regulatory Body of the GDR. Generic data are considered to be a good measure to evaluate results of reliability analysis.

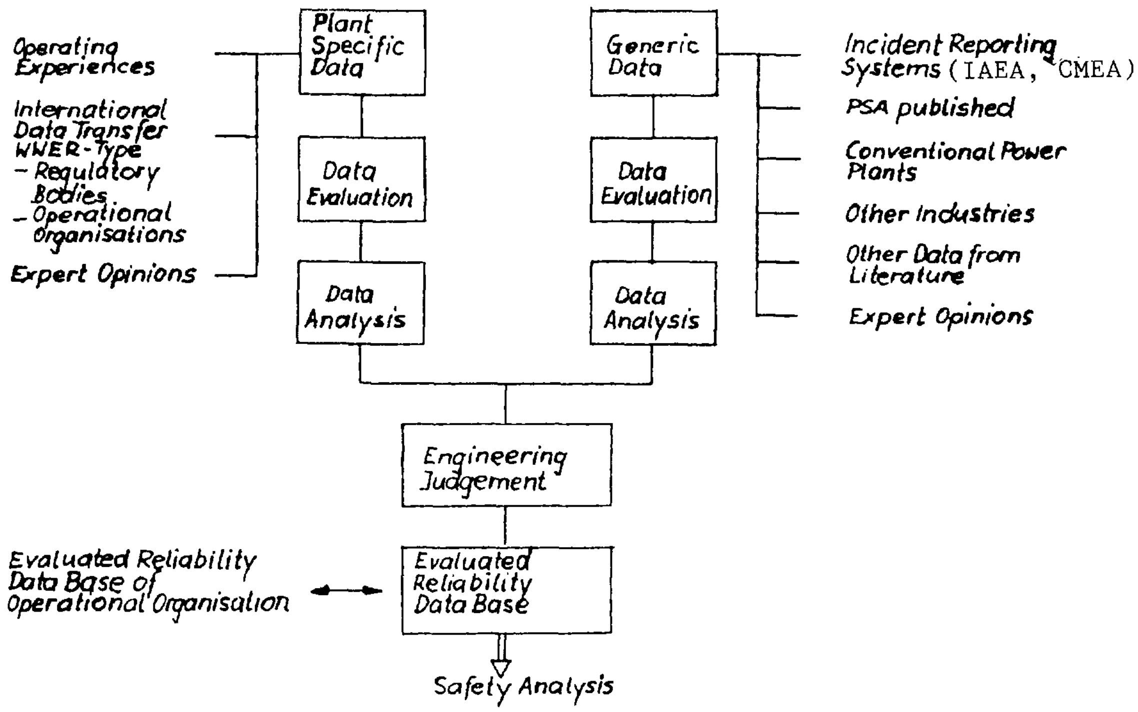

FIG.2. Structure of the reliability data base of the Regulatory Body of the GDR.

Operational experiences gathered up to now throughout the world give an average frequency of core damage accidents in thermal reactors of $1 \times 10^{-3} y^{-1}$. This number is for the ability to use thermal reactors in a safe manner, resulting from 4,500 years of reactor operation. Looking at 22 PSA published in the last years and using a lognormal distrubution you will get a median value of core damage frequency of about $7 \times 10^{-5} y^{-1}$. That is shown by figure 3. (Results from /7/)

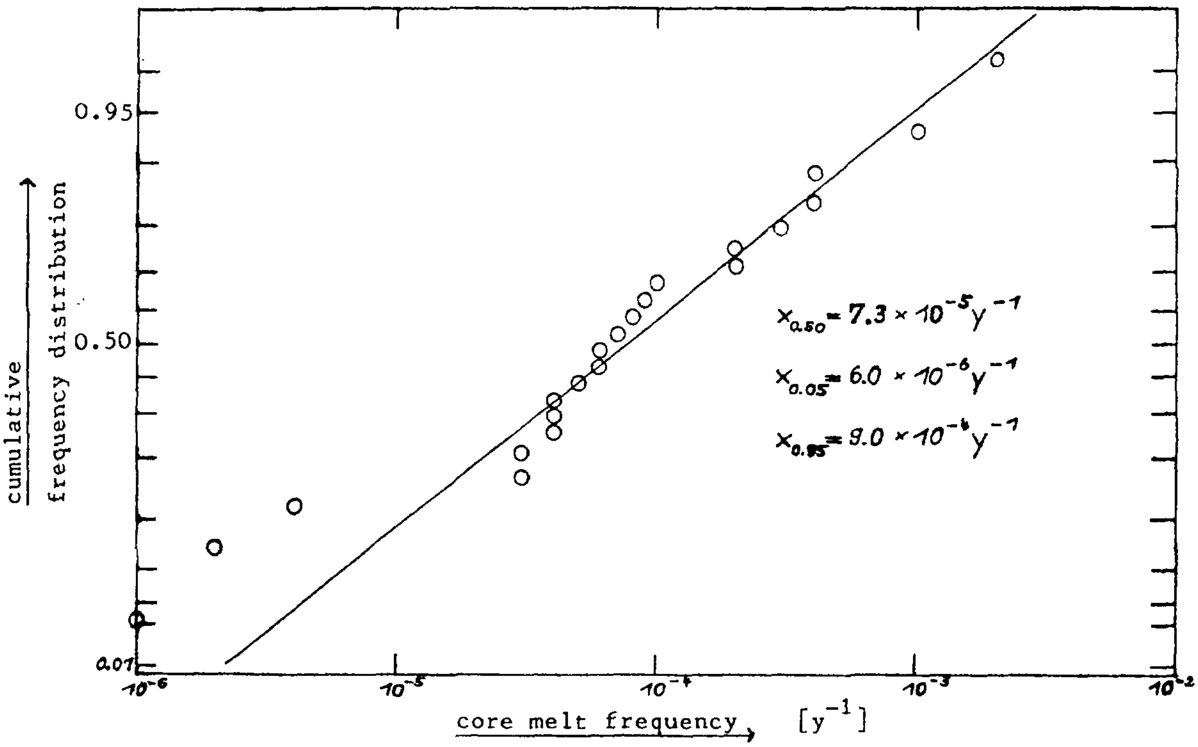

FIG.3. Distribution of core melt frequencies.

A 90% confidence interval is got by

$$6 \times 10^{-6} y^{-1} \leq x \leq 9 \times 10^{-4} y^{-1}.$$

Though the results of the PSA had been got by using different methods and assumptions they may be considered as if they indicate the possibility of core damage accidents as it is expected for modern NPP.

If PSA or reliability analysis are done their results must be carefully compared with generic data mentioned above taking into consideration the characteristic features of construction and operation of the analysed NPP.

Simular measures may be established for single components and systems from literature.

Operational experiences e.g. indicate a mean value of failures per demand of roughly 10^{-3} for proofed automatically working redundand safety relevant systems containing active mechanical components such as pumps and motor valves. Without redundancy a probability of failures per demand of roughly 10^{-2} is received. Collecting and evaluation of generic data are considered to be an important part within the framework of system safety analysis as it has been estabilished in the GDR.

Nevertheless main interest is focused on the improvement of plant specific data. They are collected not only from operational experiences of NPP in the GDR but also from other NPP of the same type as they are used in other countries.

Recently a special incident reporting system was established by the Regulatory Bodies of socialist countries characteristic features of which are focused on PSA data needs.

One of the most important reliability data sources used by the Regulatory Body of the GDR is a special abnormal event reporting system. It covers a wide range of informations.

According to /8/ there are three classes of abnormal events. They differ in the frequencies of the events and their influences on nuclear safety. Special requirements on what information is to be reported were estabilished for every class of events.

Class 1 is to cover core damage accidents as well as conditions which can lead to core damages with high probability. This class involves e.g.

- transients with failures of reactivity control,
- critical states caused by transport, storage of fuel elements,
- failures of removing heat from the core.

Class two contains other transients where core damage is expected to be impossible but given conditions of safe operation aren't met, such as lack of safety functions being not necessary and deficiencies of safety relevant systems as they are defined in the licence.

Class 3 contains all other events influencing failure conditions of safety relevant systems. Such events are e.g.

- decreased availability of safety systems
- deficiencies found during inspections and testing
- increased time of in-service inspection and repair of safety relevant systems.

Reports on abnormal events have to carry information about the following items:

- name of NPP and unit
- designation of the abnormal event
- number of the investigation report
- date and time of the event and transient sequence as well as the time when the normal state was established again
- operational state of the plant when the event was starting
- operational states of systems involved in the transient
- influence on the state of the unit at all
- transient sequence
- influence on nuclear safety
- causes of the event as well as their evaluation
- special technological system charakterisation (failure modes, causes, precursors, failure effects)
- date and mode of latest inspection, testing and repair of relevant systems and components
- other information about technology, operation and maintenance.

Special inspections are carried out by the Regulatory Body to retrieve data.

3. Concluding Remarks

Data management as described in this paper is under continuous development.

Especially the improvement of data bases is a main task at present. Therefore data exchange with countries which are operating WWER-type reactors too was enforced recently to improve specific data bases.

Special attention is paid to the evaluation of data including generic data as well as a carefull engineering judgement at all levels of safety evaluation.

Regularities obtained from data bases are used to avoid mistakes.

The further improvement of reliability data depends strongly on a sophisticated evaluation of human errors and common cause failures. A special programme was initiated by the Regulatory Body to investigate the influence of operators on abnormal events.

The main topics of evaluation of reliability data as they were found in performing safety analysis are:

i) The evaluation of reliability data is an integrated part of safety analysis including accident analysis as well as system analysis.

ii) Generic data are an important part of reliability analysis. They represent a measure to compare results as well as methods and data of reliability analysis with the international standard.
iii) Nevertheless there is a general need of improving plant specific data taking into consideration that system analysis is a describing approach.
iv) Engineering judgement is a very important topic of safety analysis.
v) Special attention must be paid to some important aspects of safety analysis such as evaluation of human errors and common mode failures.

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Mitteilungen des Staatlichen Amtes für Atomsicherheit und Strahlenschutz 25 (1988) Nr. 1

PRESENT STATUS OF THE NUCLEAR POWER RELIABILITY DATA BANK IN JAPAN

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Abstract

Since its establishment in October,1984, under instruction by the Ministry of International Trade and Industry, Nuclear Power Safety Information Research Center has built up the nuclear information data base on the construction, operation and maintenance of the nuclear power stations (NPSs) to aid in the effective fulfillment of national safety administration.

The outline of the data base will be introduced herein. Based on the incident and failure information in Japan stored in the data bank to date, the Center has started the evaluation program of unavailability of the engineered safety systems as system level failure rate (FR) and of FRs of the major valves, pumps and heat exchangers as component level FR since JFY 1987. The outline of the results of the program will be also described.

I. Nuclear Power Information Data Base in Japan

1. Role of the Nuclear Power Safety Information Research Center

Since its establishment in October, 1984, the Nuclear Power Safety Information Research Center has been maintaining the computer system and keep building up the Japanese nuclear safety information data base with the aim of enhancing the safety and reliability of the nuclear power plants and the effective fulfillment of the national safety administration by analizing and evaluating the safety and reliability of the nuclear plants, based on the information of their construction, operation and maintenance.

Main activities of the Center are as follows:

(1) Maintenance of nuclear power safety control information

(2) Analysis and evaluation of information on off-normal events

(3) Reliability evaluation of plant systems and facilities

(4) Evaluation of the plant characteristics

(5) Compilation of the pertinent information to be distributed to local municipalities

2. Information on Nuclear Power Safety management

(1) Incidents and Failures file

This data file was prepared primarily for the purposes of pattern retrieval and statistical analysis of incidents and failures.

Approximately 1,000 reports are filed, which include incidents and failures required to be reported by law and minor failures by regulatory notification.

The main items of filing are: serial number; station and unit name; title of incident and failure; date of occurrence or detection; operating status; description of incident or failure; affected systems, equipment, and components; trip signal; cause; countermeasures; influence on safety and plant output; method of detection; duration of shutdown (hours), etc.

In addition, narrative accounts of the incidents and failures and their causes are input to magnetic storage devices, enabling data retrieval by designated key words. The contents of the reports are also stored on optical disks as image data available for detailed study. In addition, OECD/NEA-IRS and USNRC LER information are also compiled as the separate data base for reference.

(2) Operation file

This file serves as the basis for an understanding and analyses of the state of nuclear power plant operation. An accurate recognition of operating status is necessary for analysis and evaluation of incidents and failures, evaluation of reliability, analysis of availability, etc. The contents of input to the file have been selected to support the various analysis accompanying the ongoing studies and evaluation, especially those concerning the electrical energy loss and its specific causes. The main input items are as follows.

(a) Power station data:
Date of initial criticality, initial connection to grid, and inauguration of commercial operation for individual unit.

(b) Annual data:
Total electricity generation (gross, net), generating hours, and capacity factors, etc. for individual fiscal year.

(c) Monthly data:
Electricity generation (gross), generating hours for each month.

(d) Data on individual energy loss:

Date of occurrence, type of cause, quantitative energy loss, reduced power level, and other items for each case of power reduction or shutdown; causes are classified into 23 categories, including periodical inspection, incidents and failures, minor incidents or failures, surveillance testing, control rod pattern adjustment, and external causes, etc.

(e) Shutdown data:

Date and time of beginning of power reduction, disconnection from grid, connection to grid, and of resumption at full power, as well as energy loss during power reduction, shutdown, and power increase, etc. for the operation cycle.

(f) Image data:

Annual operation graphic curves and monthly reports for each fiscal year.

(g) In addition, overseas information including IAEA PRIS data, France CEA ELECNUC are also compiled as the separate data base for reference.

(3) Radiation exposure and radioactive waste

(4) Inspection (experienced periodical inspection data, analyzed data on critical path schedule of periodical inspection, ISI data, welding data)

(5) Plant basic information (design values, operational limits, acceptance criteria)

(6) Equipment

(7) International power plant data (plant basic data)

(8) Documents and other materials including drawings

(9) Newspaper topics

II. Provisional Calculations of Unavailability at System Level

In case when any of maintenance and other works will be performed on the plant important safeguard system, etc. whose availability is controlled by the safety regulations, such work should, regardless of its nature be reported to MITI.

By analizing such reports about the maintenance or events, we have started the provisional calculation of unavailability at system level.

1. Status of Occurrence of troubles (standby release) of Safeguard systems

   (1) Annual trend of the occurrence of unscheduled standby release

   Unscheduled standby releases frequency of LWRs was 88 (63 for BWR, 25 for PWR) cases during the period between FY 1980 through 1985.

   Fig. 1 shows the annual trend of the occurrence; the lower side of the bar graph is for BWR, and the upper side is for PWR; the axis of ordinate represents relative values (by percent) to the value of BWR in FY 1982 which is the maximum among others.

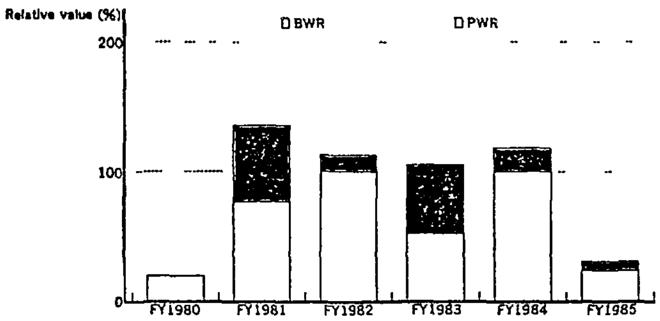

Fig 1 Unscheduled Standby Release-LWR

(2) Distribution of unscheduled standby release on a unit basis

Fig. 2 shows the distribution of the frequency of unscheduled standby releases where the total numbers (63 for BWR, 25 for PWR) of unscheduled standby releases for BWR and PWR during the above duration are respectively taken as 100%.

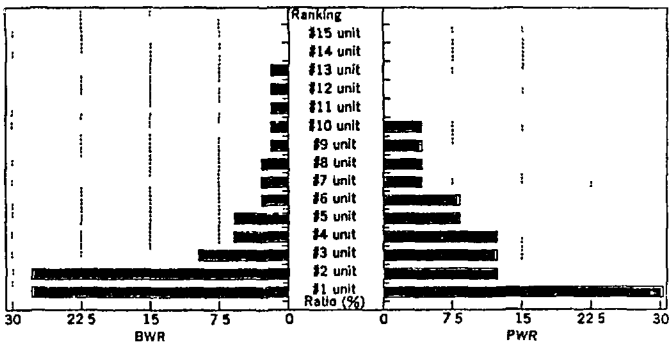

Fig. 2 Unscheduled Standby Release - Unit Base (from '80 through '85)

2. Calculation results of system unavailability

In Japan, the unavailability of the engineered safeguard systems has been zero, but calculation value consideration of without repairing failed system could be obtained using the average testing interval and in-service time. Figs. 3 and 4 show average values (in logarithmic scale) of the unavailability of engineered safeguard system.

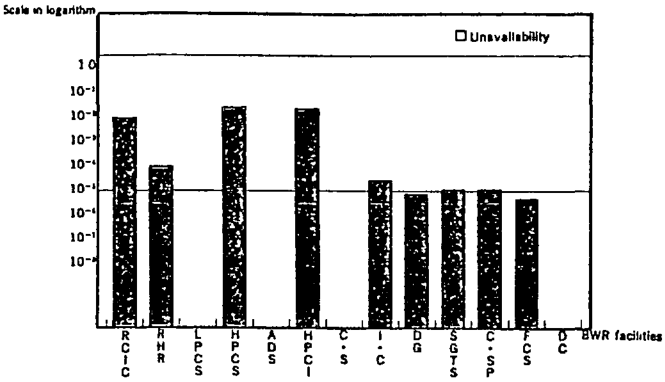

Fig. 3 ECCS Equipment Unavailability

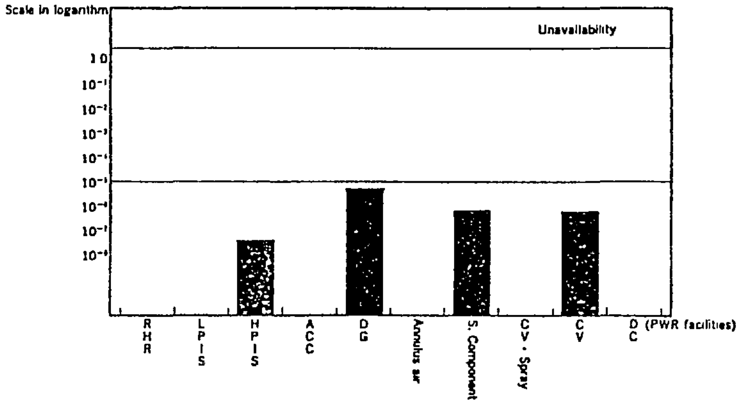

Fig. 4 ECCS Equipment Unavailability

III. Failure Rate Data Construction Program at Component Level

1. FY 1987 Program

A long-term reliability evaluation data at the component level in Japanese nuclear plants is being constructed utilizing information formally reported to MITI such as incidents and other operational events.

In the FY 1987, the following items of pumps, heat exchangers and valves are surveyed, reviewed and calculated on a trial basis;

(1) A survey of the number of incidents and events based on the formal reports on LWRs

(2) A survey of the number of specific components by type of LWR

(3) Review of calculation method of the service interval hours (frequency of demands) of specific components

(4) Review and provisional calculation of the failure rate calculation model of specific components

2. Basic Concept for establishing the Scoping of components corresponding to Failure Data

To keep appropriate accuracy of calculation, the scoping of in service components is limited to the extent of the formal reports. Since formal incident and failure reports cover those components failures of which affect plant availability and safety, the scope of components counted in this program is limited to the followings:

(1) Components constituting main piping systems or directly connecting to them

(2) Components directly affecting the functions of the engineered safeguard systems

Since the number of plant components depends on the plant capacity, components are surveyed on plant output basis (500 MWe class, 800 MWe class and 1100 MWe class)

3. Review of Calculation Methods of operating hours (frequency of demands) of Specific Components

(1) Operating hours (frequency of demands) of pumps

Operating modes of pumps are grouped into three:

a. Normally operating pumps

b. Alternately operating pumps

c. Normal standby pumps

(2) Operating hours (frequency of demands) of heat exchanger

Because of the operational characteristic of the heat exchangers are static, it is believed that there is no clear difference between operation and standby conditions. Therefore, it is judged that the annual plant operation hours may be appropriately evaluated by combining operation and standby hours.

(3) Operating hours (frequency of demands) of valve

Except for some control valves, valves are in steady (constant) conditions during most of the operation hours.

4. Failure Rate (FR) calculation

As an example, the derivation of a macroscopic FR calculation equation for pumps is as follows. The macroscopic FR of pumps of the FY 1986 is obtained by quotient of the cumulative number of events of pumps from FY 1969 through 1986 divided by the cumulative operating hours of pumps, thus expressed by the following equation:

Macroscopic FR ( )

$$= \frac{\text{Cumulative number of pump failures from 1969 to 1986 (F)}}{\text{Cumulative operating hours of pumps from 1969 to 1986 (T)}} \quad \text{---(1)}$$

The cumulative operating hours in the denominator is the sum of the cumulative operating hours of normally operating pumps, alternately operating pumps and normal standby pumps.

IV. Summary

We have constructed and been maintaining data of the operating experience of Japanese nuclear power plants, including the accumulated data for the past 20 years. We are intending to utilize all the information statistically treated or compilated. Though we have just started the provisional calculations for failure rate by utilizing Japanese data, we believe it is important to consider the following in our continuing effort in the calculation:

1. Since failure rates to be applied to PSA will reflect the quality and reliability of manufacture and maintenance, such rate should be inherently specific to each country.
2. In calculating failure rates based on LWR operation data, a simple and realistic model commensurate to the level of accumulated failure information should be used.
3. Failure rate data must be continuously revised based on data for operating experience being added. Calculational methods and preconditions, etc. should be continuously reviewed for further refinement.

ANNEX

The general outline of a treatment technique for reliability data is introduced. It is a very preliminary concept which will reflect the actual nuclear plant performance of each country.

It should be appreciated that there is a need for designated specialists to present constructive comments on this concept in order to develop this method.

General Outline

a) Get the macroscopic failure rate of each country in a strictly specified and standardized way.

b) Classify the operational records level according to the priority of the failure rate (Refer to Table 1).

c) Using a compensation factor, modify the appropriate data sets according to the special PSA purpose. The macroscopic failure rate should be continuously revised, as data for operating experience is accumulated.

Figures

Figure 1 shows the rate of capacity loss classified by causes for BWRs in Japan.

Figure 2 lists incidents and failures.

Figure 3 depicts the basic data flow chart concerning nuclear power generation in Japan.

Figure 4 indicates the general configuration of the computer system.

Table 1

$$\text{Macroscopic Failure Rate} \quad \lambda = \frac{\sum_{\text{plant}} \text{Number of failure cases}}{\sum_{\text{plant}} \text{Operating hours (or Frequency of demands)}} = \frac{Fn}{Tn}$$

Classification	Fn	Tn
Total 1. System level	By detail operational records o Number of failure cases o Failure mode o Failure cause	By detail operational records o Operating hours o Maintenance hours o Operation demands o Testing hours
Total 2. Component level (includes every supporting sub-components and parts)	By operational records o Number of failure cases o Failure cause o Scope of compornent for Fn and Tn	By Operational records with preconditions o Preconditions depend on plant capacity factor o Standardized operating hours o Standardized frequency of demands
3. Parts level (Component with-out accessories or parts)	Compensate By engineering judgement o applied to λ of non-nuclear field o alpha of specific nuclear plant operational data (e.g. Wash-1400)	Utilize Fn, Tn derived from experimental data, factory data, specific nuclear plant data or non-nuclear operational experience

Note: It is desirable to calculate *system level failure rate by detail operational records. The failure rate (FR) of *system level, *compornent level and parts level should be reviewed for consistency by using FTA or some other engineering tools to verify that reasonable engineering judgement was applied in the correction factors.

* total

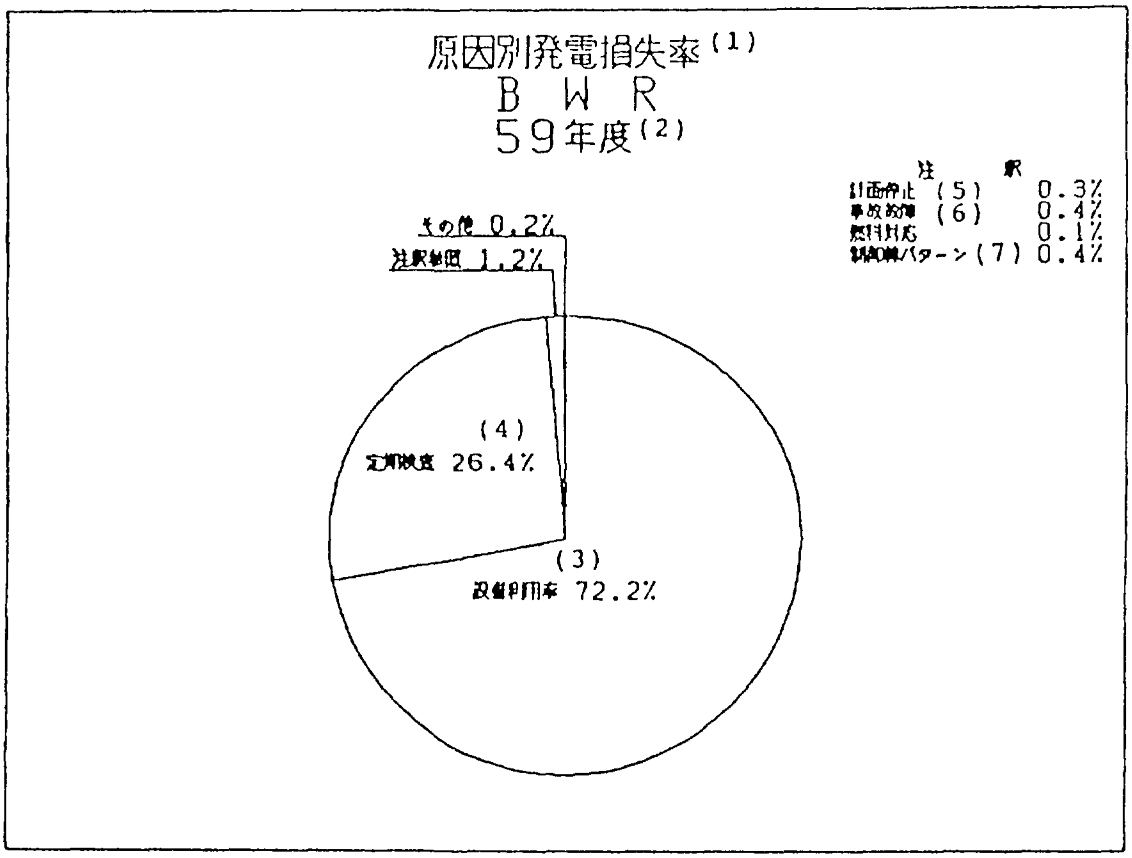

Fig. A-1 (Notes) (1) Rate of Capacity Factor Loss Classified by Causes
(2) F.Y. 1984, (3) Capacity Factor
(4) Periodical Inspection (Annual Inspection & Refueling)
(5) Short Planned Outage
(6) Incidents & Failures
(7) C/R Pattern Exchange & Ajustment

```
                                                         (1)
       1        << 事 故 故 障 件 名 一 覧 >>
NO.  整理番号          ユニット          発生年月日
 (件名)                    (2)                 (3)
===========================================================================
 1   116            XXXX           S53 01 26
原子炉再循環ポンプの異常(4)
---------------------------------------------------------------------
 2   121            XXXX           S53 06 22
発電支障事故（冷却材ポンプトリップによる原子炉自動停止）
---------------------------------------------------------------------
 3   122            XXXX           S53 08 18
発電支障事故（１Ｃ－冷却材ポンプ異常振動による原子炉手動停止）
---------------------------------------------------------------------
 4   128            XXXX           S53 11 09
Ｄ－冷却材ポンプ用モータ損壊事故（建設中）
---------------------------------------------------------------------
 5   133            XXXX           S54 01 26
発電支障事故（再循環ポンプ異常による原子炉手動停止）
---------------------------------------------------------------------
```

Fig. A-2 (Notes) (1) List of Incidents & Failures
(2) Name of Unit, (3) Date of Occurrence
(4) Outline of Incident & Failure

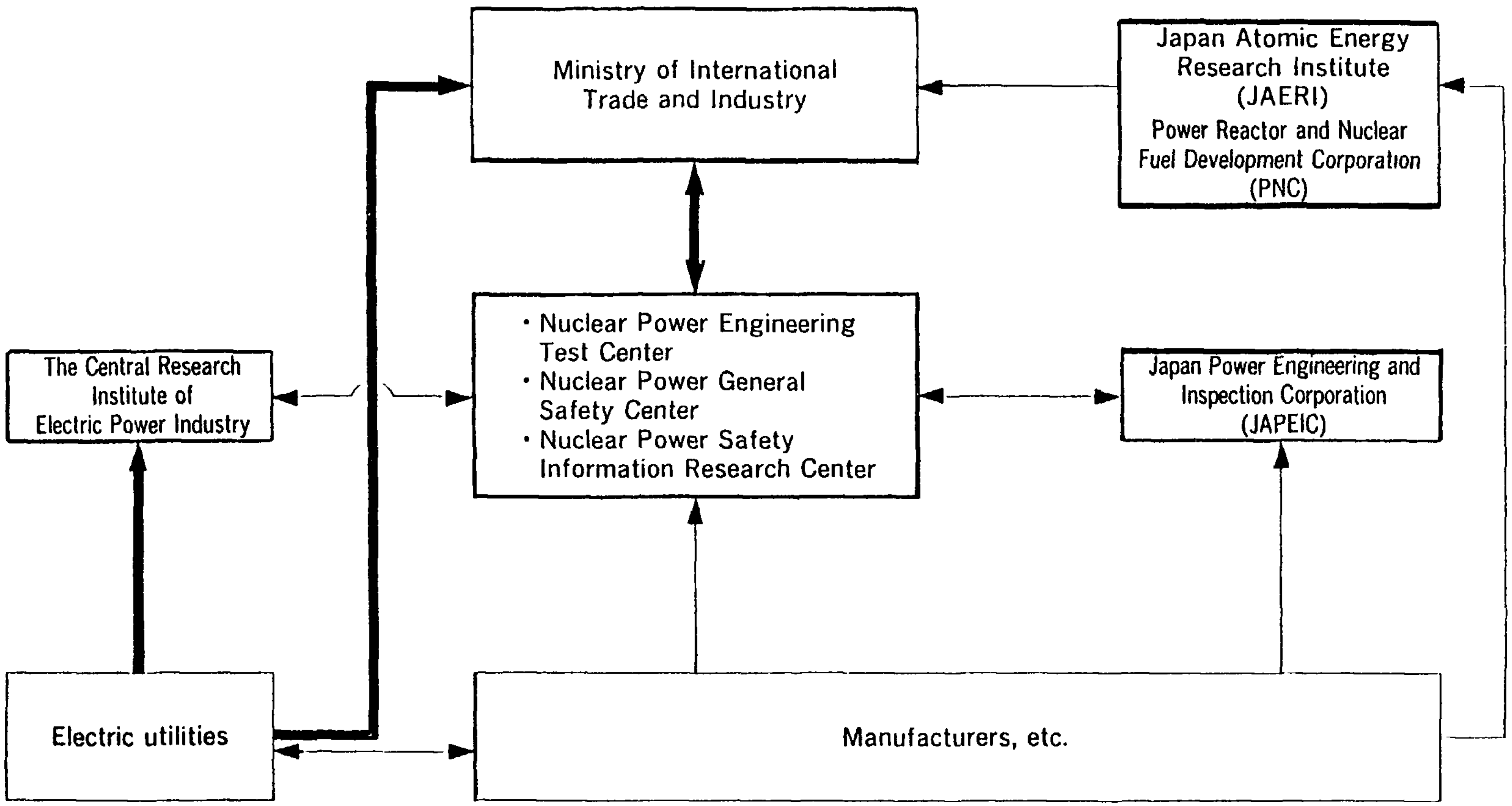

Fig. A-3 Basic Data Flowchart concerning Nuclear Power Generation in Japan

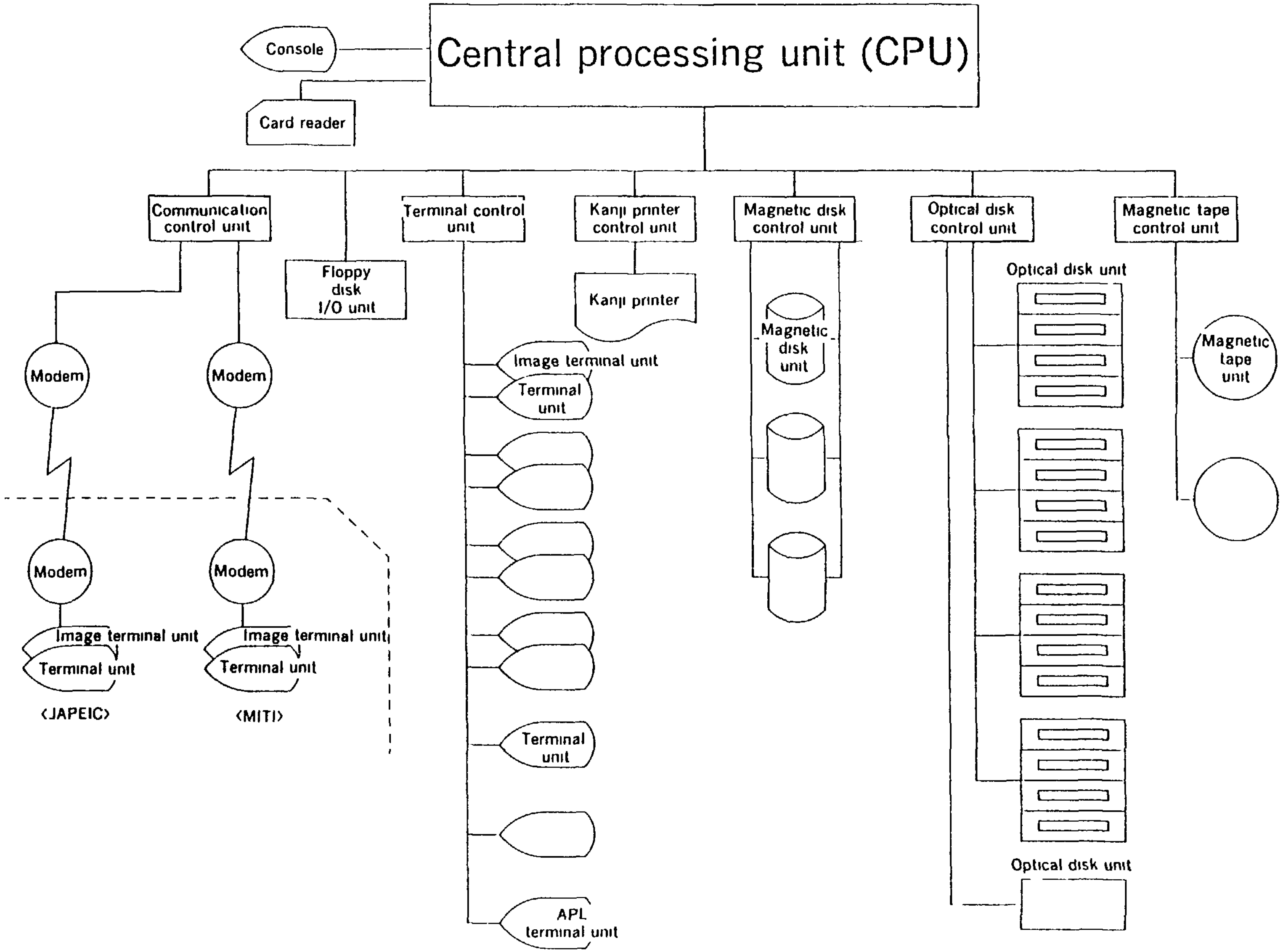

Fig. A-4 General Configuration of the Computer System

DEVELOPMENT OF FREEDOM/CREDO DATABASE FOR LMFBR PSA

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Abstract

A probabilistic safety assessment (PSA) is being performed for the liquid-metal cooled fast breeder reactor (LMFBR), MONJU, which is currently in the construction phase. FBR Reliability Evaluation Database for Operation and Maintenance (FREEDOM) has been developed at the Power Reactor and Nuclear Fuel Development Corporation (PNC) for the collection, storage, maintenance, and evaluation of the data to support the MONJU PSA. The FREEDOM system includes the data contents and structure of the Centralized Reliability Data Organization (CREDO). The CREDO system was initially developed in the United States at Oak Ridge National Laboratory (ORNL), and is jointly sponsored by the U.S. Department of Energy (DOE) and PNC of Japan. Data has been collected from the experimental fast reactor, JOYO, the Fast Flux Test Facility (FFTF), and the Experimental Breeder Reactor-II (EBR-II). In addition, data has also been collected at several important test facilities in the U.S. and Japan.

The utilization of the CREDO database has recently been initiated for the MONJU PSA. This paper describes the structure of the database and the approach of CREDO application to the LMFBR PSA. For better use of reliability data, the effects of component boundary, component failure modes and failure severity were investigated. A failure trend analysis was performed to examine the applicability of these data to the MONJU PSA. The accumulation of operational experience in LMFBRs and a detailed investigation of the existing reliability, availability, and maintainability (RAM) data would contribute to the reduction of the uncertainty of PSA.

INTRODUCTION

A PSA is being performed for the Japanese prototype LMFBR, MONJU, which is currently in the construction stage. A PSA requires an extensive database in component RAM data as well as initiating event frequency data. Available LMFBR-specific component reliability databases were relatively limited compared to those of the light-water reactor (LWR) industry. In support of the MONJU PSA, FREEDOM was developed at PNC for the collection, storage, maintenance and evaluation of the data. The FREEDOM system was originally developed for the purpose of the improvement in operation and maintenance of the experimental fast reactor, JOYO.

Furthermore, in order to perform more realistic reliability analysis, which includes the minimization of the parameter uncertainty and the exclusion of conservativeness, expansion of operational experience in LMFBRs was necessary. PNC therefore participated with DOE in the further development of CREDO. The purpose of the CREDO system is the collection, evaluation, and analysis of data associated with the operational experience of advanced reactor components. The CREDO system produces not only RAM measures of performance, such as the failure rate and the mean-time-to-repair, but can also produce enhanced statistical analysis outputs and customized analyses.

DATABASE DEVELOPMENT

The primary database, FREEDOM, has been under development at PNC since 1985. Two major purposes of FREEDOM are to provide input to CREDO and to compile information on operation and maintenance improvement for existing PNC facilities. The FREEDOM system has the following functions:

(1) Data storage
(2) Verification of input data
(3) Compilation of CREDO input
(4) Formatted output of stored data for reporting
(5) Primitive retrieval

Data has been collected from JOYO, 50 MW Steam Generator Test Facility, Sodium Exposure Test Loop, Control Rod Drive Mechanism Loop, and Sodium Flow Test Loop (now decommissioned) at O-arai Engineering Center (OEC) of PNC. A list of components collected is shown in Table 1. Three types of data are collected: (a) event data, (b) engineering data [component descriptions], and (c) operating data [hours of reactor or test loop operation per reactor mode]. All relevant events related to the reliability evaluation were extracted from operators' log books. One of the most time consuming aspects of developing a comprehensive data system such as FREEDOM/CREDO is the process of getting actual field data into the computerized database management system. The data collection work for more than 10^4 components and past operating experience took more than 300 man-months at PNC. Statistical analysis of the failure data is currently being performed.

Table 1 Component List

	CREDO		FREEDOM
1	Annunciator Modules		
2	Batteries		
3	Circuit Breakers and Interrupters		
4	Cold Traps and Vapor Traps	1	Cold Traps
		2	Vapor Traps
5	Contactors and Starters		
6	Control Rod Drive Mechanisms	3	Control Rod Drive Mechanisms
7	Demineralizers		
8	Electrical Buses		
9	Electrical Conductors		
10	Electric and Electronic Connectors		
11	Electric Heaters	4	Heaters
12	Electromagnetic Pumps	5	Electromagnetic Pumps
13	Filters/Strainers	6	Filters
14	Fuses		
15	Gas dryers		
16	Gas movers	7	Blowers
		8	Vacuum Pumps
17	Generators		
18	Heat Exchangers	9	Intermediate Heat Exchangers
		10	Dump Heat Exchangers
19	Indicators		
20	Instrument Controllers		
21	Internal Combustion Engines		
22	Liquid Rheostat		
23	Logic Gates	11	Logic Circuits
24	Mechanical Control Devices	12	Drive Unit for Vanes
25	Mechanical Pumps	13	Mechanical Pumps
26	Motors	14	Motors
27	Nonnuclear Sensors	15	Process Instrumentations
		16	Leak Detectors for Sodium
		17	Load Cell for Cotrol Rods
28	Nuclear Detectors	18	Nuclear Detectors
29	Penetrations		
30	Pipe and Fittings	19	Pipe
31	Plugging Meters	20	Plugging Meters
32	Power Supplies		
33	Pressure Vessels and Tanks	21	Tanks
34	Reactor Control Rods		
35	Recombiners		
36	Recorders		
37	Relays		
38	Rupture Devices		
39	Signal Modifiers		
40	Signal Transmitters		
		22	Preamplifiers and Nuclear Inst.
		23	Monitors for Nuclear Inst.
41	Support and Shock Devices	24	Pipe Supporters
42	Switches		
43	Transformers		
44	Turbines		
45	Valves	25	Valves

The structure of the FREEDOM/CREDO system is shown in Figure 1. The CREDO system is a mutual data-sharing and cost-sharing database between the U.S. DOE and PNC of Japan.

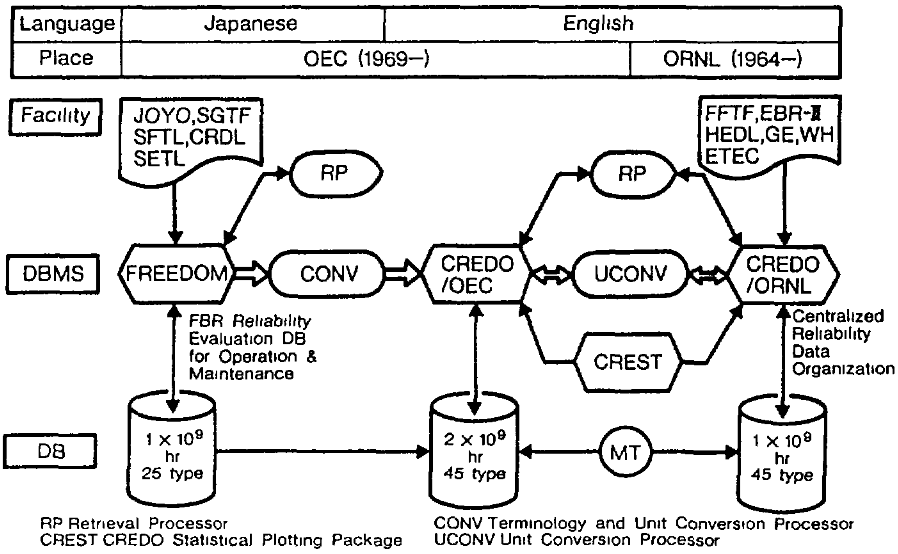

Figure 1 LMFBR COMPONENT RELIABILITY DATABASE AND DATABASE MANAGEMENT SYSTEM

The CREDO system is currently operational at both PNC/OEC and ORNL. The CREDO system is a component-based system and collects data on components that are liquid-metal-specific, associated directly with a liquid-metal environment, contained in systems which interface with liquid-metal environments, and are important critical safety-related components. Data sources in the U.S. are the FFTF and EBR-II, and several important test facilities. The data sources are shown in Table 2. The CREDO system contains information on a population of more than 21,000 components and approximately 1,500 event records. The total component operating time is approaching 2.2 billion hours.

The objectives of the CREDO system are categorized into three parts: (a) the collection of engineering, operational, and failure event data from liquid-metal, nuclear-related facilities, (b) the organization and structuring of the data into an efficient database management system, and (c) the dissemination of the data and information in the form of RAM analyses to various DOE and PNC users.

CREDO's database management system (DBMS) was designed to catalog and store data in three types of files - engineering, event, operating. The engineering data file contains a unique description of each component as reported by each site. The event data file contains detailed data concerning any CREDO-reportable event that occurs to components being tracked by the CREDO system. The operating data file consists of a set of chronological reports that give the accumulated operating history of a reporting unit. The event data include a description of the event, the method of detection, the failure mode, the failure cause, corrective action, etc. An engineering data supplement is

Table 2 CREDO Data Sources

Liquid Metal Reactors
Fast Flux Test Facility (FFTF)
Experimental Breeder Reactor-II (EBR-II)
Experimental Fast Reactor JOYO
Test Loop Sites
Energy Technology Engineering Center (ETEC)
Westinghouse Advanced Energy Systems Division (WAESD)
General Electric Advanced Reactor Systems Department (GE/ARSD)
Hanford Engineering Development Laboratory (HEDL)
O-arai Engineering Center (OEC/Japan)
50-MW Steam Generator Test Facility
Sodium Exposure Test Loop
Control Rod Drive Mechanism Loop
Sodium Flow Test Loop
Other Data Sites
FERMI-I
Hallam Nuclear Facility
Sodium Reactor Experiment (SRE)

used in conjunction with the engineering data file to collect such information as: (1) a description of the component in terms of its engineering parameters, (2) materials data for critical parts of the component, and (3) design and operating parameters for the component. This information enable a detailed analysis concerning the design specification.

CREDO's standard statistical output is composed of two separate analyses. The primary metric generated is failure rate. It is calculated by taking the ratio of the total number of failures for a specified population, to the total number of operating hours for the same population. This definition inherently assumes that the failure population is exponentially distributed. Another item of interest is a listing of the separate failure modes. This listing provides the number that failed, the percentage of failures per mode, and a 5% and 95% confidence interval around the failure rate mean. In addition to failure data, repair data is also provided in the form of mean-time-to-repair based on a log-normal distribution.

One of the most important aspects of the reliability database is quality assurance of the data entered. The data gathered must provide a consistent picture of the operation of components and systems being addressed. In order for CREDO to provide multi-site data, it is obvious that an underlying consistency and uniformity in both nomenclature and component definition is essential. The CHECKER program is used for the systematic checking such as verification for spelling of keywords and the range of numerical value. This program flags any missing or incorrect data. A data screener re-checks in addition to the computerized checking. This is necessary because much of the supportive information concerning the description of a component and its history of operation are reflected in the narratives included in all three types of data collected. Such logical checking requires the attention of a specialist employing good engineering judgement.

DATABASE APPLICATION

The systems model of the MONJU plant has been constructed based on a large fault tree-small event tree method. The system fault trees which involve more than 10^4 fault events, were reviewed to obtain a list of components and failure modes. The resulting list was categorized and used to request reliability data. The reliability data for LMFBR-specific components are extracted from the FREEDOM/CREDO system. In addition, extensive data gathering and processing of generic components such as electrical components and water/steam components have been performed from other existing databases. The collected data is compiled to quantify the failure events included in the systems model. For multiple data sources, the geometric averaging technique has been applied.

In the FREEDOM/CREDO database application to the MONJU PSA, the following treatment is utilized: the fault tree model is developed to the detail of that used in the database. Consistency of the component boundary is essential to avoid overlap or omission. Therefore, in some cases, the fault tree was adjusted to conform to the developed database. This kind of feedback is necessary for better use of reliability data. The CREDO staff has ensured consistency in the reporting of data by defining a set of 45 generic components since the definition of the term "component" may vary from site to site.

Also, the definition of failure is an important aspect. A failure is sometimes caused outside of the component boundary and defined as secondary failure. A secondary failure is often caused by a loss of driver power, flow control valve abnormality due to variation in the media pressure, etc. A primary/secondary failure flag is attached in the database. This information is useful for searching cascade types of dependent failures.

The component failure/success criterion in a fault tree analysis depends on the failure mode and failure severity. The failure modes are defined by 35 types of keywords in the CREDO system. Since there might exist some inconsistency between the failure event modeled in a fault tree and the keyword in the system, careful examination of both are required. Effective matching and categorization are performed based on keyword definitions and sometimes involves reading the narratives of event data. The failure severity information are classified into three conditions: complete, partial, and incipient. Such a categorization allows the user to apply the most appropriate failure/success criteria.

Failure trend analysis has been performed in order to examine the applicability of the database to the MONJU plant. The environmental effects and engineering parameter effects are of primary interest. The effects of environment on the failure rate have been examined for three types of valves: manual, motor-operated, and pneumatic valve. The medium processed was chosen as a primary concern of the environmental effect because utilization of sodium is specific to LMFBRs. The comparison of failure rates is shown in Figure 2. The bar represents 5% value, mean, and 95% value from the left to the right. The 5% and 95% values are obtained by the chi-square estimation. The bar length represents uncertainty range which is controlled by the failure population. It is observed that the failure rates are different between sodium valves and gas valves. It is considered that this evidence is due to the difference of the design specification and operational environments. Another observation is the difference of the failure rate of valves with different actuator types. This difference in failure rate due to the actuator type is more distinct. The pneumatic valve is less reliable than the manual valve by an order of magnitude. The motor-operated valve is less reliable than manual valve by a factor of two or three. It is recognized that the difference is due to the failure causes in the actuator parts. It is judged that these two

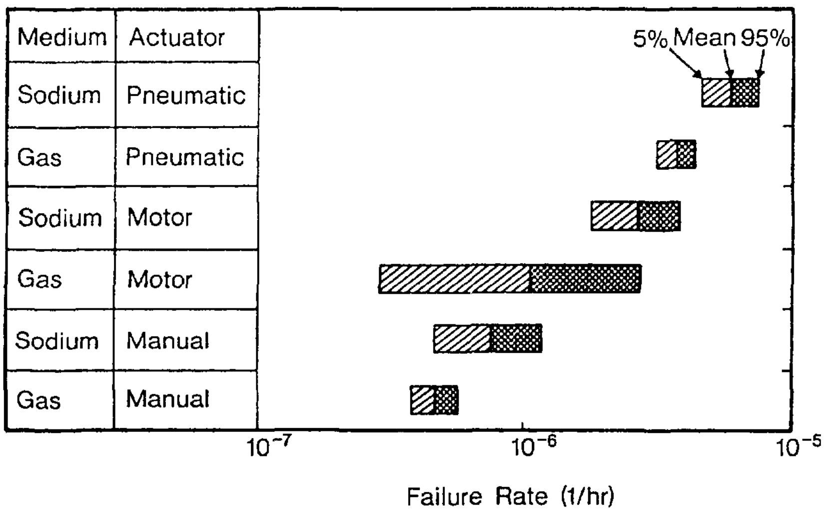

Figure 2 COMPARISON OF FAILURE RATES FOR THE VARIOUS TYPES OF VALVES

effects are independent. Therefore the appropriate component failure rates are used in the reliability evaluation.

The effect of design temperature on the failure rate of the motor-operated sodium valve has been examined. The results are shown in Figure 3. Almost all of the failure rates reside within a small range. A slight trend is indicated that higher temperature valves accompany a higher failure rate. This is particularly for the valves whose design temperature is above 650°C. They tend to have a failure rate which is an order of magnitude higher than those designed for lower temperature operations. Since the maximum design temperature of sodium valves for the MONJU plant is less than 650°C, these data can be excluded.

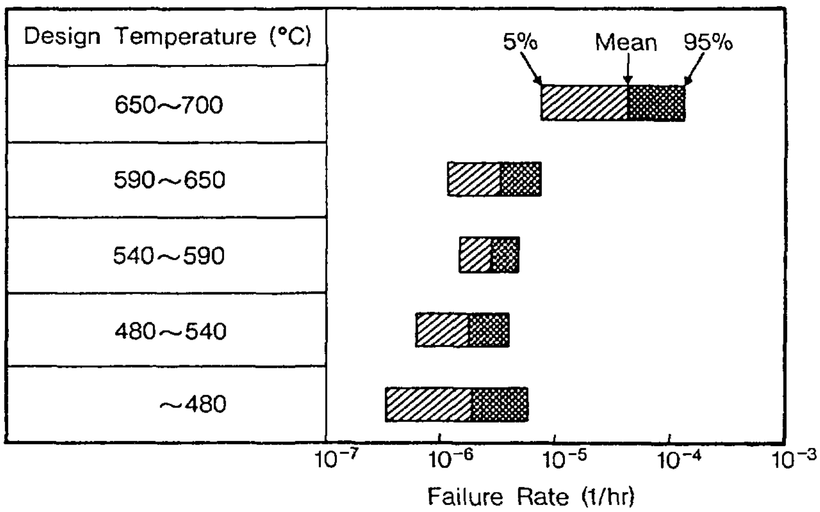

Figure 3 COMPARISON OF MOTOR OPERATED SODIUM VALVE FAILURE RATES FOR THE VARIOUS DESIGN TEMPERATURE

The effect of valve size on the failure rate has also been examined. It is observed that only a small failure rate trend exists for valve size. Therefore all the data can be applied to the MONJU valves.

With respect to data sources, no apparent trend was found. Because a data population size of a specific component and its distribution vary from site to site, it is difficult to separate such an effect. At this stage, the whole data sources are included to obtain the reliability data.

In the course of the MONJU PSA, some importance measures are evaluated to interpret the results. The uncertainty and sensitivity analyses show that the human error factors and dependent events have a significant effect on the results and have a relatively large uncertainty because of a relatively small population of data. The key components contributing to the total core damage frequency are also identified. Hence, these components should be given higher priority in data collection work. Collection of human error event data is initiated in the FREEDOM database based on the human error check sheet. Though explicit common cause failures were not identified in the database, the efforts on the search will be performed for the potential common mode failures.

Because of a relatively sparse data population, the use of generic data, and the inclusion of expert judgement, there exists a relatively large conservatism and uncertainty in the analyses performed. The utilization of the specific component failure rate enables one to reduce the error factor of the results. The FREEDOM/CREDO database allows for a detailed description of components and associated failure definitions It is judged that the parameter uncertainty has been reduced by the application of the FREEDOM/CREDO database. The failure rate of a sparse component is controlled by its operational experience. The accumulation of the operational experience is expected to refine the component failure rates in the FREEDOM/CREDO system.

CONCLUSIONS

The FREEDOM database has been developed for the purpose of LMFBR PSA and improvement of operation and maintenance in existing facilities. The internationalization of the database has been achieved by participation in the CREDO program operated by ORNL and now jointly sponsored by the U.S. DOE and Japan's PNC.

Throughout the application of the CREDO system to the MONJU plant, it is essential to assure the quality of the database, to keep consistency of data definition, and to investigate the failure trends. The detailed information included in the FREEDOM/CREDO system allows the user to analyze such multiple effects on reliability data. The accumulation of operational experience for LMFBRs and a detailed investigation of the reliability database aids in reducing the uncertainty in PSAs.

ACKNOWLEDGEMENTS

The authors are thankful for the encouragement of R. J. Neuhold (DOE), H. Alter (DOE), and A. Watanabe (PNC).

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RELIABILITY DATA SOURCES IN THE PAKS NUCLEAR POWER PLANT

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Abstract

In the Paks Nuclear Power Plant with four unit in operation there is a 10 reactor years operational experience. This fact makes possible to use the results of the operational experience for probabilistic safety evaluations.

For safety evaluations up to now we used datas given by the supplier - what was a limited source -, and datas obtained from different existing data banks in the world.

In the near future for safety performances we would like to use plant specific datas, gained from the Paks Nuclear Power Plant.

This paper describes the existing data sources in the Paks Nuclear Power Plant, which contains information and statistical datas, necessary for safety evaluations. These sources are

Incident investigation report
Safety related event report
Report of events with reduction of plant power
Component failure data system (for a limited nunber of components).

Our future task is to define how to extract the necessary information and datas from the different reports and to extend the component failue data system for all the main equipment of the plant.

1. Introduction

In Hungary activities in probabilistic safety assessment of nuclear power plant started approx. 4 years ago. As part of the level I PSA of Paks NPP several case studies have been performed. Parallely with performing level I PSA different particular system reliability problems have been analysed during the last two years, such as optimal test interval of the safety systems, possibility of maintenance of the condenser cooling pumps during operation of the plant. There are some other problems which have to be solved in the near future by means of reliability analysis.
The safety of our plants was aproached in a deterministic way. In our operational manuals there are several technical specifications and limiting conditions of operation which are, we consider, very pessimistic. Some of these techspecs and LCOs might be overviewed using PSA technics.
For the above mentioned reliability and safety evaluations up to now we have mainly used reliability data taken from publications.
Since the start up of the first unit of the Paks NPP (28.dec.1982) 10 reactor years operational experience have been gained .(Now there are four units in operation.)This could have been sufficient to feed back the operational experience and as a form of feed back, to use plant specific data for PSA. Unfortunately not a total reliability data collection was initiated immediately after the start up of the plant.

Especially many problems we had concerning the reliability data of the components.
In this paper we briefly describe those data sources which contain information necessary for PSA calculations and our future tasks in the field of data collection and evaluation of reliability data sources.

2. Existing Data Sources

At the present time in the Paks Nuclear Power Plant there are several incident and malfunction recording systems. These report systems were established to provide information for the operational organizations. At the time of the formation of the requirements for the form and contents of these reports the PSA was not in general use in Hungary. Therefore such points of view which could guarantee the usability for PSA, could not have been taken into consideration. Nevertheless these reports contain a lot of useful information for safety evaluations. Just the proper way should be found, how to extract and evaluate the information from the incident reports.

As far as the component failure reporting system is concerned, the situation is worse. The component (or equipment) failure recording system, established at the Paks Nuclear Power Plant after the start up of the first units was, one could say "too simple" and inhomogenous. All the different maintenance organizations (mechanical, I& C, electrical) recorded failure information in different way concerning both the form and the content. The basic information for the calculation of reliability parameters of the components were not recorded or were recorded in an insufficient way. For this reason a fairly new system must be established, to get full failure statistics of the components. The concepts of such a reliability data system were described by the authors of this paper about two years ago/Y/ Unfortunately during the last two years the development of the above mentioned system has not gone in a way and at a rate as we had planned it. This fact has first of all reasons concerning organization and interest.

The different kind of information originates within different (most of the information at maintenance) organizations. It is very difficult to force and teach maintenance people to collect and record information which to their opinion is mostly unnecessary. The more organizations are envolved in the data recording, the less the accuracy of the recorded data is.

For these reason in January of 1988 we started a limited scope data recording system for a limited number of components (so called stand by safety systems), which may be operated entirely by only one organization which is interested in the accuracy of the collected data.

Now we would like to describe very briefly those types of reports which in some way could be used as reliability data sources, including the new limited scope component failure recording system.

2.1. Incident investigation report
Safety related event report

Incident investigation must take place in case, when an occurence is classified by the electrical network dispatcher as an "incident" (loss of power-production event).

The requirements to the content of the report is described in Appendix A. The "incident" classification is not defined by the fact, whether the nuclear safety of the plant is concerned or not.

It is defined by the consequences from the operational point of view of the electrical network.

Therefore for those cases when the occurence is not classified as an incident by the network dispatcher, but the nuclear safety is concerned, a so called safety related event investigation must take place and a report must be written.

The content requirements of the report is described in Appendix B.

As we can see both kinds of reports contain practically the same information. In both reports there may be found the description of the initial state, a detailed description of the event sequences, evaluations of the event, description of the operator actions, human responsibility and the necessary measures to be undertaken.

2.2. Load reduction report

The load reduction report system actually is the first form of the equipment failure reporting system. But un fortunatelly this system is expanded only for those equipments whose failure causes load reduction. Therefore good statistics may be obtained only for those component, any failure of which causes load reduction. As far as this recording system - like the incident reporting system - have existed since the start up of the first unit of the plant, they are suitable for evaluation of the probability of some initiating events.

But for calculation of component reliability parameters the data collected in this system are still not adequate.

2.3. Failure report

As it was mentioned earlier, in Jan. 1988. a new failure report system was started for a limited number of systems and components, with limited possibilities. For basis of this systems serves the periodical test of the safety systems and, some automations and protections.
The safety systems and the main automations and protections are tested once during six weeks according to a schedule for a whole year. After each test performed a test report has to be written (see Appendix D), wich practically serves as an official document. Report must be written in those cases as well, when the test is not a periodical but a special one, for example after the repair of a main component of the safety system.

Special test must be perfomed for two operable safety systems in the case, when one of the three systems is inoperable during a period of time not exceeding 24 hours, too.

In case, when during the test any equipment fails, a failure report form has to be filled in (see Appendix E). By means of this data recording system the reliability of those components may be evaluated which are characterised by failure per demand. Systematic documentation of the performed tests and detected failures will assure the accurate number of failures and demands for all components.

3. Use of Generic and Plant Specific Data

As was indicated in preceeding chapters for safety evaluations up to now data obtained from different existing data banks and generic data files have been used. Plant specific data given by tbe supplier or extracted from plant operational statistics were used in few specific cases.

For compilation and automatised retrieval of generic data a computerized data base is being set up Z . Presently the data base involves INITIATING EVENT AND COMPONENT FAILURE DATA files.
INITIATING EVENT cathegories are: LOCA and TRANSIENT ones.
COMPONENT FAILURES are grouped on

- types, as TECHNOLOGICAL, ELECTRICAL, and C/I failures,
- modes, as STAND-BY and OPERATIONAL ones.

For combination of data from different sources simple avarage rule using weighting factors is used. Files listed presently contain information publisbed in following reports:

WASH-1400	Reactor Safety Study	, 1975
GRS	German Risk Study	, 1979
NUREG-2815	PSA Procedures Guide	, 1985
NUREG-2728	IREP Procedures Guide	, 1983

NUREG-3862	Transient IE Frequences	, 1985
EPRI-801	ATWS Frequencies	, 1978
ANO-1	ARKANSAS IREP Study	, 1982
RKS 85-25	Reliability Data Book	, 1985
NKA SAK-1	PRA Uses and Techniques (A Nordic Perspective)	, 1985
EUR 10696	CEC Benchmark Exercise	, 1986

For illustration of compiled generic file structures a sample case sub-file is given in Appendix F.

Plant specific operational and failure data have limitedly been used in the following areas:

- to define mean time between failures and average repair times, e.g. for condenser cooling pumps within their maintenance policy study,
- to formulate auxiliary assumption in numerical data form for given calculation, e.g. for FRANTIC code to define q. unavailability of test override capability within safety system test period optimization study,
- to screen causes of plant outages and load reductions, to estimate distribution of failures within different failure modes and systems,
- to verify a - priori assumed event sequences through operational tests or real cases , e.g. to make clear mission times within reduction of feedwater event sequence study,
- to establish realistic success criteria for redundant safety system trains, e.g. within small and large LOCA event sequence studies.

It is emphasized that plant specific data from Paks NPP were used in above listed cases only in a rather limited scoppe for PSA purposes, and in the future a more systematic data acquisition, processing and retrieval system is necessary.

4. Conclusion

Data contained in different plant reports described in Chapter 4. involve a lot of information which are necessary and/or useful for PSA purposes. These types of data potentially available can be summarized as follows.

Type of Report	Involved Data Available for PSA
1. Incident Investigation Report, Safety Related Event Report Limitation: Only for - network operation or - safety related sequences	a/ Initiating Event - identification - frequency b/ Event Sequence - identification - time scaling c/ Human Errors and Recovery Action
2. Load Reduction Report Limitation: Only for component failures which result load reduction	a/ Initating Event (see 1.a.) b/ Component Failure Paramerters - failure rate, outage time - failure mode (Limited) c/ Personnel Action
3. Test Report, Failure Report Limitation: Only for tested safety systems and components	a/ Component Failure Parameters - failure per demand probability - failure mode and cause - failure identification mode b/ Component Repair Rate

Considering data listed above two main conclusions can be drawn:

1. Operational experience recorded in differnt plant reports during the past 10 reactor years involve more information useful for PSA purposes then what was used up to now. The main reason for limited application is the huge manpower required to manually retrieve information from existing data sheet documents.

2. There is a need for computerized storage, processing, and retrieval of data contained in the reports described in present paper, as well as for extension of scope of compiled data useable for reliability and probabilistic safety assessments. For this purpose a computerized reliability data base software is under development within VEIKI and PAKS NPP cooperation. This work is partly supported by the IAEA through a research contract, (see Ref.X.)

REFERENCES

Y - E.Holló, J.Tóth: Current Activites and Future Trends in RA and PSA in Hungary, IAEA TC Meeting, Vienna, Austria 25-29 Nov. 1985.

X - Development of Methods and Procedures for Collection and Analysis of Data for Probabilistic Safety Assessment, IAEA-VEIKI Research Contract No. 4855/RB, 1987

Z - A.Vánkai, E.Holló: GRDB - Generic Reliability Data Bank, IAEA Worhshop, Seregélyes, Hungary, 7-11 Sept. 1987

Incident investigation report form

1. Identifcation number:
2. Unit number (s)

3. Title of the incident
   - reactor protection   I.   II.   III.   IV.
   - turbine protection
   - electrical protection
   - other:

4. Beginning of the incident:
   End of the incident:
   Duration of localization :
   Duration of restoring :

5. Loss of electr. production: ...... MWh
   Loss of heat production : ...... GJ
   Power deviation from the plan
   average : ...... MW
   max. : ...... MW

6. Classification of the incident

7. Discription of the incident
   7.1. Initial state (deviation from the normal state)
   7.2. Way of detection
   7.3. Discription of the event (detailed)

8. Evaluation of the incident
   8.1. Cause of the incident
   8.2. Actuation of the controllers, automations, protections, alarms
   8.3. Evaluation of the operator actions

9. Equipment failure

10. Method of the equipment repair
11. Personnel responsibility
12. Measures to be taken
13. Necessary special expert investigation to be taken
14. Other comments
15. Investigation is finished (date)
16. Investigation commission (names, signatures)
17. Appendices

Appendix B

Safety related event report

Ident. number:

1. Unit :
2. Title:

3. Beginning of the event
   End of the event

4. Initial state (thermal power, electrical power, core average temperature increase, boron concentration, control rod position ...)

5. Description of the event (detailed)

6. Evaluation of the event
   Cause of the event
   Actuation of alarms, controllers, automations, protections

7. Operator action
8. Measures to be taken
9. Investigation commission: (names, signatures)
10. Apendices

NPP PAKS	Load reduction report		Unit

Event title:

Protection: Signal:

Beginning of the event:	yr.	month	day	hr.	min.
End of the event :	yr.	month	day	hr.	min.
Duration :				hr.	min.

Reduction max:	MW	aver.:	MW	Loss of production	MWh

Classification

1. Incident
2. Malfunction
3. Operative maintenance
4. Planned preventive maint.
5. Other
6. Network demand
7.
8.
9.
0. No classified

Field:

1. Primary side
2. Secondary side
3. C I
4. Electrical
5. Auxiliary
6. Chemistry
7.
8.
9. Other
0. No field concerned

Cause:

1. Design
2. Manufacturing mounting
3. Failure of equipment
4. Quality of documentation
5. Human error
6. Test
7. Repair failure
8. Other
9. Unknown
0. Network demand

Main equipment concerned:

1. MCP
2. SG
3. Turbine, overheater
4. MIV
5. HP preheater
6. Diesel generator
7. 6 kV equipment
8. Generator, exiter
9. Transformer
0. No equipment concerned

Equipment actuation:

Personnel activity:

Restoration of the initial state:

Comments, proposals:

Filled in by; Checked by:

Unit :
Interlock.
Protection:

TEST REPORT

PERIODICAL - SPECIAL: Cause:....................
Interlock.
Protection:...............
Ident:....................

1. Date of test : .... yr. ....month. .....day

2. Failure found during

Maintenance before test	Measure undertaken
....	....
....	....
....	....
....	....
....	....
....	....

3. Failure found during test	Measure undertaken
....	....
....	....
....	....
....	....
....	....
....	....

4. Qualification of test results:

5. Necessary further measures:

....

....

....

6. Initial state restored:

Techn. Dep:....... C I:.............. Electr:..............

Signatures

F A I L U R E   R E P O R T

Component: Ident. code:

Date of failure :	yr.	month	day	hr.	min.
Beginning of repair :	yr.	month	day	hr.	min.
End of repair :	yr.	month	day	hr.	min.
Start of operation :	yr.	month	day	hr.	min.

Failure mode:

Active:
* Fails to start
* Fails to stop
* Spurious start
* Spurious stop
* Fails to open
* Fails to close
* Spurious openning
* Spurious closing

Passive:
* Leakage
* Rupture
* Deformation
* Plugging

Cause of failure:

* C I
* Electrical
* Mechanical
  * Desing
  * Manufacturing
  * Mounting
  * Operation
  * Repair
  * Test
  * Metallurgical
  * Unknown
  * Other

Failure detection:

* Test
* Preventive maintenance
* Sound/light alarm
* Abnormal state indication
* Routine checking

Measures undertaken

* Change of part
* Repair of part
* Total change
* Desing modification
* Temporary repair
* Other

Consequence:

* Loss of system function
* Loss of subsystem function
* Failure of other component
* Load reduction MW
* Loss of production MWh
* Generator switch off
* Reactor shut down
* No consequences
* Other

Date:
Signature:

Appendix F

COMPONENT STAND-BY FAILURE DATA

Component Type: TECHNOLOGICAL COMPONENTS

No.	Component Name	Failure Mode	Type	Mean	EF	MTTR	Source
1.	Pumps						
1.1.	Motor-driven						
1.1.1.		Failure to start	3	3.00E-03	10		
			3	3.00E-03	10		NUREG-2728
			2	1.00E-05			NUREG-2815
			2	4.00E-05			NKA SAK-1
			3	3.00E-03	5		EUR 10696
			3	1.20E-03	3		WASH-1400
1.2.	Turbine-driven						
1.2.1.		Failure to start	3	3.00E-02	10		
			3	3.00E-02	10		NUREG-2728
			2	1.00E-04			NUREG-2815
			3	1.20E-02	3		EUR 10696
1.3.	Diesel-driven						
1.3.1.		Failure to start	3	1.00E-03	3		
			3	1.00E-03	3		NUREG-2728
			2	1.00E-06			NUREG-2815
1.4.	Centrifugal,hor/vert,75-250kg/s,0.3-0.9MPa						
1.4.1.		Failure to start	3	3.90E-03			
			3	3.90E-03			RKS 85-25
1.5.	Centrifugal,hor/vert,30kg/s,2.2-6.7MPa						
1.5.1.		Failure to start	3	1.40E-03			
			3	1.40E-03			RKS 85-25
1.6.	Centrifugal,hor/vert,120-240kg/s,1.2-1.8MPa						
1.6.1.		Failure to start	3	5.10E-03			
			3	5.10E-03			RKS 85-25

INITIATING EVENT ANNUAL FREQUENCY

IE Type: LOCA INITIATORS

No.	IE Definition		Mean	EF	Source
1.	LOSS OF RCS FLOW				
1.1.	Loss of RCS Flow in 1 Loop	1	4.40E-01	4	
			4.40E-01	4	NUREG-2815
			3.90E-01		EPRI-801
1.2.	Loss of RCS Flow in All Loops	14	2.80E-02	4	
			2.80E-02	4	NUREG-2815
			2.00E-02		EPRI-801
2.	LEAKAGE IN PRIMARY SYSTEM				
2.1.	Leakage from Control Rods	4	2.30E-02	4	
			2.30E-02	4	NUREG-2815
			3.00E-02		EPRI-801
2.2.	Leakage in Primary Circuit(no size)	5	1.10E-01	4	
			1.10E-01	4	NUREG-2815
			9.00E-02		EPRI-801
2.3.	Large Leak in Primary Circuit		1.00E-04		
			2.70E-04		GRS
	0.5 - 2 inch		1.00E-04		WASH-1400
	10 - 13.5 inch		1.20E-05		ANO-1
	greater than 13 5 inch		7.50E-05		ANO-1
2.4.	Medium Leak in Primary Circuit		3.00E-04		
			8.00E-04		GRS
	2 - 6 inch		3.00E-04		WASH-1400
	4 - 10 inch		1.00E-04		ANO-1
2.5.	Small Leak in Primary Circuit		1.00E-03		
			2.70E-03		GRS
	0 5 - 2 inch		1.00E-03		WASH-1400
	0.38 - 1 2 inch		2 00E-02		ANO-1
	1 2 - 1 66 inch		3 10E-04		ANO-1
	1.66 - 4 inch		3 80E-04		ANO-1

RELIABILITY DATA ACQUISITION IN CEGB NUCLEAR POWER STATIONS

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Abstract

The Central Electricity Generating Board (CEGB) has two computerized databases currently in use for obtaining reliability data; the scope of one of these is about to be substantially increased. The Plant Reliability/Availability (PR/A) system logs predominantly availability data, in a very comprehensive manner, for all conventional and nuclear power stations (approx. 50) within CEGB. Some reliability data is logged, but it is of little relevance to Probabilistic Safety Assessments (PSA). The other database is the Nuclear Plant Event Reporting System (NUPER) which records safety related events, but only those at the important end of the significance spectrum. The former is entirely quantitative, whilst the latter is mainly text.

Starting in January 1988, detailed reliability data on all nuclear safety related plants will be logged by the first of these systems. To begin with this will be on only one power station, but it is intended that it will be extended to all concrete pressure vessel stations (12 nuclear units) by the end of the year.

A unique safety system list is needed for each station, as these are all substantially different from each other due to design evolution. Each list is ordered according to importance, with the intention of beginning the data collection only for the top items but gradually extending downwards. Each list contains approximately fifty items, each of which is made up of an indeterminate but large number of components. It is intended that the reliability information be collected at the "item" level, albeit with the component intitating each fault identified. This is done to contain the magnitude of the task to one that is practicable.

All faults occurring during operation or discovered by surveillance activities are to be logged. Additionally, all unavailability due to maintenance and testing will be recorded. These activities depend strongly on the enthusiasm of the station personnel; namely the operations and the maintenance engineers. A specially designed reporting form is being produced for these two groups, to be filled in by the respective supervisors at the end of every shift. This will be obligatory, but even so it is anticipated that some of the information will be incomplete. In order to encourage the better reporting of failures, and to obtain any missing data retrospectively, a new appointment will be made at each station. This "Information and Feedback Engineer" will have no responsibilities other than the handling of safety related information.

It is intended that the fault and availability information will be used in the production of safety system indicators, in the first instance, but ultimately in PSA's for each of the stations.

SOME PROBLEMS WITH COLLECTION, ANALYSIS AND USE OF RELIABILITY DATA

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Abstract

Typical problems with the collection, analysis and use of reliability data are discussed.

It is argued that the collection of reliability data has to be selective, and that insufficient attention to this selectiveness is responsible for the majority of problems with the collection of data. The collection of reliability data must be carefully planned and undertaken by dedicated, well-trained and well-motivated staff.

The reliability data must be analyzed, tested and used as carefully and cautiously, and under the same discipline, as other engineering parameters.

---

1. INTRODUCTION

In this paper I will highlight some of the common problems associated with the collection, analysis and use of reliability data. This is not a comprehensive survey; rather this is a selection of the various problems I have encountered during my work in safety assessment.

Reliability data are generally required for three different, but related purposes:

(i) to learn from the past, i.e. to ensure that past problems are not repeated,
(ii) to choose at the present, i.e. to ensure that adequately reliable components and systems are used, and
(iii) to forecast the future, i.e. to develop models of component and system failures and to assess their reliability and the risk they impose.

I believe that because of the different requirements there cannot be a universal method for data collection, analysis and use, but only an appropriate method, and this should be devised to deal with our particular application or problem. Thus, for example, quantitative reliability data may be appropriate in reliability and risk assessment, but they are not particularly useful for design development and product improvement. On the other hand, qualitative data are suitable for design development and product improvement, but useless in reliability and risk assessment.

I will discuss the various problems under three different headings:

- problems due to collection of data,
- problems due to analysis and evaluation of data,
- problems due to retrieval and use of data.

2. COLLECTION OF DATA

It must be appreciated that we cannot collect all the data all the time. Such an approach would not be practicable; it would place an unbearable strain on the collection system. It is inevitable that we will have to be selective, and it is the insufficient attention to this selectiveness which is responsible for the majority of problems with the collection of data. Since it is impracticable to have an all-embracing, universal collection scheme, we have to select - we have to decide on which aspects to concentrate: which plant to collect the data from, which components and systems to include, how to define failures, which non-failures to report, etc. This selectiveness must not be ad-hoc or considered only as an afterthought. It must be regarded of fundamental importance in devising and designing adequate collection systems.

Thus, I do not believe that the objective of data collection is to collect the maximum amount of information, but rather the objective must be to collect the relevant information. Hence, we have to decide at the beginning why we are collecting the data and for what purpose. Obviously, computerisation enables more and more data to be handled, but it should be appreciated that data collection is much less ameanable to computerisation than, for example, data analysis and retrieval.

Insufficient consideration of what to collect and why can lead to the following problems with the collection of data:

(i) insufficient information collected,
(ii) inconsistent information collected, and
(iii) unreliable information collected.

2.1. Insufficient information collected

This follows practically always from a badly executed preliminary analysis of the need for the data. Typical omissions, which may make a particular data collection less than useful, are as follows:

- the underlying causes of the failures and the failure mechanisms,
- the consequences of the failures,
- the operating and environmental conditions,
- the period over which the data have been collected and the behaviour and history of the component or system in question.

The last aspect is of a special importance if, for example, quantitative reliability data are required. It is not sufficient to know how many times a particular component has failed, we also have to know how many times it has not failed when called upon to operate, etc. This is obvious, but unfortunately not always appreciated or collected.

2.2. Inconsistent information collected

This, once again, usually follows from a less than thorough preliminary analysis. Comprehensive, but inconsistent collections may appear superficially adequate, but a more detailed analysis of the data (which is invariably undertaken much later) then reveals many hidden shortcomings. Unfortunately, it may then be too late to amend the collection system.

Typical problems are as follows:

- inconsistent definition of components and systems (especially in the definition of the boundary),
- inconsistent definition of component failures. For example, what constitutes a failure - a pump not starting on the start-signal or not starting within 30 seconds of initiation.

Problems can also arise if a particular collection system is based on a physically incorrect model. The collection requirements may be so strongly driven by the demands of this particular model, that if the model is then shown inadequate the collected data may not be appropriate for any other purpose. This is particularly true when data are collected on rare events, such as dependent failures, etc.

2.3. Unreliable information collected

We have to know how reliable and error free is our particular collection system. This can be the most difficult problem of data collection. It can be partially dealt with by having an adequate in-built QA scheme, but this in itself may not be sufficient. What is of primary importance is to ensure that the data are collected by dedicated, well trained and well motivated staff.

3. ANALYSIS AND EVALUATION OF DATA

The problems in this area can be conveniently discussed under the following headings:

(i) insufficient understanding of the failure mechanisms,
(ii) insufficient distinction between the various sources of data,
(iii) statistical shortcomings, and
(iv) bias in evaluation.

3.1. Insufficient understanding of the failure mechanisms

Many problems have been observed in this area. For example, in the case of dependent failures the analysis may be particularly strongly model-driven. The scarcity of the data may then make the phenomenology of the failures appear more important than their mechanism. This can then lead to problems when the model is extrapolated to different situations.

Another example is the difference between time-dependent and demand-dependent failures. If this difference is not appreciated and understood, the incorrect assumption of time-dependency may lead to unrealistic expectations of increased availability of stand-by systems postulated by their more frequent testing.

3.2. Insufficient distinction between various sources of data

There are many sources of reliability data and they all should be considered. However, the limitations and the benefits of the various data sources must be taken into account. The typical sources of reliability data are :

- operational data,
- field trials,
- laboratory testing,
- generic published information,
- expert opinion.

All these sources can provide useful information, but only if used correctly. It is generally accepted that operational data are most appropriate. The advantage of field trials and laboratory testing is that various parameters can be varied, but the limitation is that important operational factors may be missed. This again shows that field trials and laboratory testing must be carefully designed and controlled.

The advantage of the generic published information is that the data are usually conveniently available, and some of them may have been endorsed by virtue of being used by reputable organisations. However, the main disadvantage is that the primary sources of the data are not always given and thus not open to scrutiny. Hence the status of the generic published information may be uncertain, and the use of the information for purposes different than those initially envisaged may be inappropriate and possibly misleading.

The use of expert opinion can be contentious. First, some people find the whole philosophy of the Bayesian approach flawed. However, I do not think that this is the major problem. I believe that the second aspect of this approach causes much greater difficulties - the credibility and the expertise of the experts.

It can be dangerous to use either anonymous experts or well known experts with expertise which is irrelevant to our problem. We must always ensure that we know who the experts are, what are their credentials and what are the bases for their opinion. We must not forget that experts frequently over-estimate their knowledge and that, because of their background and contacts, they may not be giving independent advice. Expert opinion should be tested as any other source of reliability data, and not accepted uncritically.

3.3. Statistical shortcomings

One of the most common examples of this problem is the use of the median of a distribution instead of the mean. Since for certain distributions the numerical difference between the two can be considerable, the confusion can lead to difficulties.

3.4. Bias in evaluation

The whole process of data collection and analysis is rather time-consuming. In order to save on time and effort we may be more prepared to accept without further analysis those results which appear reasonable, and only analyze further those results which do not conform with our experience. Thus, we are quite ready to accept without too many questions what we consider normal, without perhaps appreciating that our assumption of normality may not be justified.

4. RETRIEVAL AND USE OF DATA

After collection and analysis the reliability data become available for retrieval and use. Since the data are commonly used by groups different from those responsible for collection and analysis, the design of the appropriate retrieval system also deserves careful considerations. It is not sufficient to give just the reliability data; the range of validity, the operating conditions and the limitation of the data must be given too.

It was suggested in Section 2 that the objective of the data collection is to collect the relevant information (as opposed to the maximum amount of information). However, the design of the retrieval system must follow a different philosophy: all the information collected must be available for retrieval. Hence, computerization of the retrieval system is required.

If the collection system is designed to give detailed descriptions of the causes and the mechanisms of component and system failures, the data can be used purely qualitatively. Such data can be used most effectively in design development. Thus, for example, we can use the data to re-design a particular component or system to ensure that particular failures are eliminated or at least made acceptable. It must be stressed that to be suitable for this purpose the database must be carefully designed. For example, it is not sufficient to give numerical values of reliability; good qualitative description of the failures must be also given.

As suggested in Section 1, the reliability data are mainly used quantitatively - either in the design stage or the evaluation stage. There are some important differences between the two applications, but they are in many respects similar and inter-related.

The most important common problem is to decide which reliability data should be used. Do we use the data from historical databases, or do we take into account technological progress. There are good arguments for both approaches.

For example, the use of the historical databases will indicate how well we have used proven technology. This will give us a direct comparison with other designs based on the same or similar technological developments.

On the other hand technology has advanced and we do learn from our past mistakes. Thus some improvement in availability and reliability of various components and systems is to be expected. It is then argued that designers should be made aware of this, so that they are reminded to do better than in the past.

I believe that the latter approach is appropriate and legitimate and should be used, provided that:

- there is evidence that over the years the reliability and availability of components and systems have been improving,

- this observed and documented improvement, rather than a postulated (or a hoped for) improvement, is cautiously taken into account,
- the targets for improved performance, reliability and availability are challenging, but realistic.

It cannot be over-emphasized how important the above conditions are. If they are not followed and if the targets for various improvements become divorced from the reality, the credibility of the whole approach is lost.

This implies that, once again, only appropriate reliability data should be used. As in the collection and the evaluation of data, the use of the reliability data must also be planned. Using the data in isolation and in an ad-hoc manner wastes much of the considerable effort put in their collection and evaluation and can lead to distorted and uneconomic designs.

5. CONCLUSIONS

The collection of reliability data must be carefully planned and undertaken by dedicated, well-trained and well-motivated staff.

The reliability data must be analyzed, tested and used as carefully and cautiously, and under the same disciline, as other engineering parameters.

Acknowledgement

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PACS PROGRAMME — PROJECT FOR ANALYSIS OF COMPONENTS AND SYSTEMS OF NUCLEAR PLANTS

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Abstract

Objective of the PACS Programme is to evaluate data of abnormal events on Italian NPPs in operation, for backfitting and design improvements purposes.

Low threshold events will be analyzed with statistical methodology, aimed to evaluate reliability parameters of components and systems

Results of analysis will be stored in a computerized data base, and updated time by time, as new operational data will be available

Some results of PSA studies, already performed, will be revised utilizing plant specific data, drawn out by PACS programme

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1. Objective and Scope

Objective of the PACS programme is to evaluate, in a systematic way, the operating experience (o.e.) of Italian plants in operation, for backfitting purposes and design improvements of new plants.

First of all, data of Caorso BWR plant will be analyzed, taken into account the following:

- this is the most recent Italian plant in operation;
- most of its o.e. is expected will be applicable to Montalto BWR plant, now in construction;
- a quality assurance (Q.A.) programme was implemented on Caorso plant from the beginning, so that all operational data were collected in a systematic way.

In a second stage data of the other plants in operation will be analyzed.

Methodology and software package are expected could be applied to plants other than nuclear ones.

Furthermore technical results could give some indication of behaviour of components used in conventional plants as well, at least for some selected failure modes. Obviously different design and operational criteria should be taken into account

Methodology of analysis will be of both qualitative and quantitative type, so that first by predominant failures and failure modes will be searched, in a second stage data will be analyzed with statistical techniques, aimed to find out reliability parameters of the analyzed parts.

As a result of the programme, a software package for reliability analysis will be available and a data bank containing reliability parameters of components and systems

The software package will be developed on personal computer. Guided mennes will be provided, with a provision to update, time by time, data entry as new data will become available. In such a way "living" reliability data will be available, and indication of degraded failure rate will be drawn out, if any, on the basis of trend analysis.

Some connections will be established with other ENEA/DISP activities, such as probabilistic safety studies (PSA), already performed for Caorso and Montalto plants. Some results of PSA studies will be revised, utilizing plant specific data drawn out from PACS programme.

Mainly the job will include the following steps:

- definition of components and systems to be analyzed, on the basis of the master part list of Caorso plant and the classifications already used for PSA studies;
- definition of failure modes to be analyzed, on the basis of design and real plant characteristics (i.e. only drift towards technical specificiation limit violations, instead of all events of drift, will be considered);
- definition of models (i.e. components in operation, in stand by, etc.);

- development of the software package, data entry and output format, technical analysis of the selected components and systems and storage of results;
- plant backfitting and design modifications implementation, if any

2. Pilot Study and Ongoing Programme

A pilot study was performed manually on diesel generators of all Italian plants in operation, with the aim to orientate the entire programme, to improve plant safety on the analyzed system and to define the software resources needed.

We analyzed this system, as an example of two different conditions, a system in stand-by and in operation. In such a way we had a complete modelling of possible failure modes

Furthermore DGs are one of the most important safety related systems

In fact, many PSA studies underline the importance of DGs performance in contributing to core damage frequency.

Data were analyzed with the aim to evaluate the following parameters:

- unavailability on demand,
- reliability in operation.

For the first parameter, we considered all startups of the diesels and the failures to startup were considered.

For the second one, the time of real operation of these engines and the pertinent failures to operate were evaluated.

Reliability parameters of single DG and total ones were found out. Also homogeneity analysis of results for each plant and in general were performed, so that the influence of different desig characteristic, manufacturing and operation could be evaluated.

As results of the analysis the following aspects were focalized.

- surveillance programme of Caorso plant is adequate;
- a better performance of the diesels could be reached with the implementation of a preheater system of the lubrication oil;
- dishomogeneity was found among the 4 diesels of Caorso plant, mainly for DG3;
- critical subsystems were found out for each diesel and for each plant.

Further studies are now in progress on.

- drift of instrumentation of emergency safety systems of Caorso plant;
- malfunctions of valves of safety related systems of Caorso and Trino plant

Average reliability parameters of all pressure instrumentation has been evaluated and compared with the target in IIIE500. Now specific evaluation is in progress for each system, instrument type and manufacturing.

Analysis of valves malfunctions was recently started up. At the moment classification of valves is in progress, aimed to group valves according to system, type, size, manufacturing and environment condition.

IAEA'S EXPERIENCE IN COMPILING A 'GENERIC COMPONENT RELIABILITY DATA BASE'

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Abstract

Reliability data are an essential part of a probabilistic safety assessment. The quality of data can determine the quality of the study as a whole. Among all the data which are needed for performing a PSA study, component failure data are the ones most frequently mentioned.

It is obvious that component failure data originated from the plant being analyzed would be most appropriate. However, in few cases complete reliance on plant experience is possible, mainly because of the rather limited operating experience and usually limited number of failures for meaningful statistics. Nuclear plants, although of different design, often use rather similar components, so some of the experience could be combined and transferred from one plant to another. In addition information about component failures is available also from experts with knowledge on component design, manufacturing and operation.

That bring us to the importance of assessing generic data. (Generic is meant to be everything that is not plant specific regarding, the plant being analyzed). The generic data available in the open literature, can be divided in three broad categories. The first one includes data base used in previous analysis. These can be plant specific or updated from generic with plant specific information (latter case deserve special attention). The second one is based on compilation of plants' operating experience usually based on some kind of event reporting system. The third category includes data sources based on expert opinions (single or aggregate) or combination of expert opinions and other nuclear and non-nuclear experience. If one is to use generic data sources either directly or as a prior for updating, much information is required about the different aspects of generic data sources.

This paper reflects insights gained compiling data from generic data sources and highlights advantages and pitfalls of using generic component reliability data in PSAs.

Considering current IAEA efforts to prepare a computer code package for event tree and fault tree analysis in personal computers (PSAPACK) and the associated need for a reliability data base, a compilation of published component reliability data was undertaken at the IAEA. Some of the features of the data base, like the coding system, are, therefore directly governed by the package.

As of today the generic data base contains about 1000 different records, including practically all the components which are accounted for in PSA studies of Nuclear Power Plants.

Having in mind the goal of compiling data from many data sources, 20 sources have been included so far. The amount of information contained in the various sources is, however, substantially different. Some of the sources provide up to 180 different records, while one source was cited in only two of the records.

With many different sources providing different types of information, it was necessary to define a unique record form which would enable inclusion of information in a systematic and consistent manner and also user friendly for information overview and retrieval.

The record form was defined as having 21 lines, characterizing 10 categories of information.

The IAEA Generic Data Base was created using the IBM-PC software dBASE III, so it can be stored in data base or in the plain text format. Therefore its use is not limited to PASPACK.

During the development of the IAEA Generic Data Base insights were gained in how different data bases address different problem areas, namely: component boundary definition, failure mode definition, operating mode definition, operating environment definition. These insights and possible ways of avoiding or solving such problems are addressed in the paper.

The IAEA effort to compile a generic component reliability data base aimed at identifying strenghts and limitations of generic data usage and at highlighting pitfalls which deserve special consideration. It was also intended to complement the PSAPACK package and to facilitate its use.

Moreover,it should be noted, that the IAEA has recently initiated a Coordinated Research Program in Reliability Data Collection, Retrieval and Analysis. In this framework it is expected that the issues identified as most affecting the quality of existing data bases would be addressed and alternative solutions proposed.

1. INTRODUCTION

Reliability data are an essential part of a probabilistic safety assessment. The quality of data can determine the quality of the study as a whole. Among all the data which are needed for performing a PSA study, component failure data are the ones most frequently mentioned.

It is obvious that component failure data originated from the plant being analyzed would be most appropriate. However, in few cases complete reliance on plant experience is possible, mainly because of the rather limited operating experience and usually limited number of failures for meaningful statistics. Nuclear plants, although of different design, often use rather similar components, so some of the experience could be combined and transfered from one plant to another. In addition information about component failures is available also from experts with knowledge on component design, manufacturing and operation.

That bring us to the importance of assessing generic data. (Generic is meant to be everything that is not plant specific regarding, the plant being analyzed). The generic data available in the open literature, can be divided in three broad categories. The first one includes data base used in previous analysis. These can be plant specific or updated from generic with plant specific information (latter case deserve special attention). The second one is based on compilation of plants' operating experience usually based on some kind of event reporting system. The third category includes data sources based on expert opinions (single or aggregate) or combination of expert opinions and other nuclear and non-nuclear experience. If one is to use generic data sources either directly or as a prior for updating, much information is required about the different aspects of generic data sources.

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Considering current IAEA efforts to prepare a computer code package for event tree and fault tree analysis in personal computers (**PSAPACK**) and the associated need for a reliability data base, a compilation of published component reliability data was undertaken at the IAEA. Some of the features of the data base, like the coding system, are, therefore directly governed by the package.

2.IAEA's GENERIC DATA BASE

As mentioned earlier a primary reason for developing the data base was to have readily available reliability data for use in conjunction with the PSAPACK. As the package is planned to be used for analysis of different plants, one important aspect was to draw data from a wide variety of sources.

Usually, different sources present information in a different manner, therefore a common input form had to be defined. To enable data retrieval and direct use with the PSAPACK, a unique coding system had to be developed. Having chosen the data sources, the selection of the data for inclusion had to be made. At last, the data input and quality control to create the generic data base was a very time consuming task. Each of the above points define steps in a generic data base compilation, and are elaborated in more detail next.

As of today the generic data base contains about 1000 different records, including practically all the components which are accounted for in PSA studies of Nuclear Power Plants.

2.1. Data sources

Having in mind the goal of compiling data from many data sources, 20 sources have been included so far. The amount of information contained in the various sources is, however, substantially different. Some of the sources provide up to 180 different records, while one source was cited in only two of the records.

To highlight some of the basic characteristics of the sources included, the best way is to divide them in the three basic categories (mentioned earlier) each of which show some unique characteristics.

Some of the sources belong to more than one category. A typical example is the "German Risk Study" where some data are NPP operational experience, some are combinations including NPP experience, while rest is a combination of several different data sources, not including NPP experience. Therefore the ultimate data source is not always unique.

2.1.1. Plant specific data

Two basic subgroups exist inside this category. The first is plant specific data drawn directly from sources available at the plant (logbooks, maintenance records, work orders etc.), and the second one is when generic data are updated with plant specific information.

The first subgroup is normally considered the best source of data for the analyzed plant, but that is not necessarily the case when one uses these data at another plant. Generally, this is a source rarely found. The only source in the IAEA Data Base fully in that category is NUREG 4550 (Vol.3.Surry NPP) and it provides only 10 records. It should be noted that some other sources also provide the single plant operating experience data but as most of the data provided there belongs to other catetgory, they are quoted later.

The second subgroup considers generic data updated with single plant operating experience. This procedure is usually applied when either limited plant specific data are available, or available data could tend to over-or-underestimate component reliability. In fact in most of the recently completed PSA studies (which are not using generic data base) component reliability data are derived in this manner. IAEA Data Base include several sources of that kind(e.g. Oconee NPP PRA, Zion NPP PRA and a source identified as "Old W PWR".

In addition to the problems encountered in defining component boundaries and failure modes which are going to be addressed later, the means of acquiring raw data at the plant have the greatest impact on the quality of reliability data derived.

There are basically two sources to derive raw data at the plant. These are logbooks and maintenance work orders. Both of them have advantages and drawbacks.

Usually deriving raw data from the maintenance work orders is easier and less time consuming (especially when work orders are computerized). Because every work order adresses, in principle, an abnormal occurrence, events related to each single component could be easily compiled together. The quality of information found in the work orders is generally not very good, because work order forms are filled by the personnel actually performing work. Examples like work orders open for months or years and work done on one component identified as done on another are common. Logbooks, especially ones filled by control room personnel are more accurate, but deriving raw data from there is extremely time consuming.

Even if the raw data are drawn from the logbooks or maintenance records, one is still not sure that all the failures of a certain component have been reported. if both sources are searched, the probability of failures not reported is lower. However it still exist, and can result in an overestimate of component reliability.

It is understandable that the quality of component failure data is directly proportional to the quality of plant's records. If the plant has a dedicated reliability data collection system in place, that would be obviously the best possible source of raw data.

The problem is even worse for demand related failures, when the actual number of demands is not readily available and have to be assessed on the basis of average time on power or calendar time. If a component is started for testing purpose, it is usually not known weather it started immediately or after a number of trials.

Operating experience for the standby systems involving failure to run given start is usually limited to running time of about 1 hour. However it is usually used (in analysis) as the long term failure rate, without any evidence that the long term failure rate is equal or comparable to the short term one.

2.1.2. Data extracted from reporting systems

A widely known NPP event reporting system is the Licensee Event Report System used in the USA. Safety significant events occurring at the NPP-s have to be reported, so it is possible to identify component failures related to those events. Identification of component failures is not always straightforward, and other means of discovering components involved have to be utilized.

The IAEA Data Base includes 4 sources of that kind, namely LER based rates for valves, pumps, I & C and control rods.

In one way similar to the LER based rates are the failure rates published in the Swedish "Reliability Data Book", which provides the reliability parameters derived from Swedish LER-s, ATV system (The Swedish

Thermal Power Reliability Data system) and information provided by the plant staff.

Advantages of reliability parameters derived in that manner is that the actual component population covered is very large, what guarantee more reliable statistics. On the other hand, LER systems are event oriented and not component oriented, so actual component failure could get misinterpreted or overlooked. In addition, some of the component failures are never reported in the system because their failures either did not cause any safety significant event or they were not required to report. Furthermore, a small percentage of events is not reported because of plant personnel general attitude towards reporting system. All this factors may lead to possible overestimates in component reliability.

Another problem area is the operating time and number of demands of the component. Operating time is usually estimated on reactor operating time, and number of demands is estimated as an average also based on operating time. This can drive the predicted reliability parameter in either direction.

Compilation of that kind tend to diminish differences in component design and operational practice and environment, which is sometimes a very important information and can greatly influence the component reliability.

To conclude, reliability parameters found in this type of source should be used with care in PSA studies.

2.1.3. Data based on expert opinion, nuclear and non-nuclear experience

Categories in this group include single expert opinion, aggregate expert opinion, aggregation of several non-nuclear sources, aggregation of expert opinion and other sources and aggregation of operating experience of several NPP-s. Usually, even a single data source includes several of these categories. It is obvious that any aggregation of data (if properly performed) provides more reliable data than single expert opinion or single source.

The most widely known representative of this category is The IEEE Standard 500. Its 1977 version mostly includes expert opinion, while the 1984 version also includes nuclear and non-nuclear experience. Other examples of data sources which are included in the IAEA Data Base are: NUREG 2728-IREP (Interim Reliability Evaluation Program, which adopted data base from EGG-EA-5887), NUREG 2815, PSA procedures guide (data from expert opinion combined with IREP data base), Sizewell B assessment (operating experience including nuclear and other industrial sources).

The WASH-1400 (combination of expert opinion, non-nuclear, nuclear sources) also belongs to this category and it is important to mention that it still is a widely used source. Some of the sources included in the IAEA Data Base like NUREG 2886 and NUREG 3831 draw data for parameter estimation from a limited group of plants. Other example is the Shoreham NPP PRA- GE data, which draw data only from GE operating plants.

The quality and reliability of data in this category can vary substantially, depending on the final source. It is important mentioning that expert opinion was several times proven to be in very good agreement with actual operating experience data.

2.2. Record form

With many different sources providing different types of information, it was necessary to define a unique record form which would enable inclusion of information in a systematic and consistent manner and also user friendly for information overview and retrieval.

The record form was defined as having 21 lines, characterizing 10 categories of information (table 1).

Table 1: Record categories

1. code	1 line
2. component type	4 lines
3. operating mode	1 line
4. operating environment	1 line
5. failure mode	2 lines
6. failure rate	5 lines
7. repair time	1 line
8. source	2 lines
9. component boundary	1 line
10.comments	3 lines

1. Every record has a code which is a unique combination of 5 alphanumeric characters. A detail description of the coding system is presented later.

2. Component type is described in 4 lines, namely: type, subtype, detail type 1 and 2. Type characterizes basic component type (e.g. "pump", "valve"). Subtype characterizes more specifically the component category (e.g. motor driven pump, solenoid operated valve, pressure sensor, AC motor). Some of the components do not have information at this level (battery charger). Detail type 1 contains information about the system where the component is located or other characteristics as voltage, or pipe diameter etc. Valve types (e.g. gate, butterfly, etc.) are also included in this line. "General" means that further characterization is not possible. Detail type 2 is the last entry of the component description. Usually a detailed division of component categories which should belong to this entry is not available. For most of the valves, pumps and some transformers information about size or the system which the component belongs is found in this line.

3. Operating mode is the next category. Operating mode is a particularly important characteristic for pumps (standby, alternating or running). For other components this information is of less importance. Precise, information of that type is seldomly included in data bases. When the component operating mode is obvious, like a safety injection pump which is a standby pump, this information is included. In other cases "all" operating modes was the default value chosen.

4. Operating environment is the next entry which, similar to the previous one, is seldom found in data sources. It is obvious that harass environment should influence the component failure rate, but very few sources address that. For example IEEE 500 provides a multiplication factor for most of components listed for environments like high radiation, temperature, humidity or vibrations. WASH 1400 provides different failure rates for pumps and motors in extreme, post accident environment. Failure rates are, particularly in the cases where the operating experience is the basis for determining the failure rate, usually based on normal operating environment. A default value "normal" was chosen for all cases where no other environmental condition was indicated. Some of the sources addressing components operating environment define "normal NPP environment" as the usual one.

5. The failure mode category is presented in two entries, one describing "generic" failure mode and the other presenting failure mode as found in the original source. Details about the failure modes are going to be described later. Briefly, a generic failure mode was assigned because the coding system was not able to cope with the number and differences in failure modes found in the sources. The original failure mode was, however left in the record for users' clarification.

6. The failure rate is presented in 5 entries. The first entry is failure rate description, giving information about the failure rate (mean or median), upper and lower bounds (percentiles of the distribution, low and high or max. and min.values), and defining the failure rate as per hour or per demand. The failure rate entry provides the actual numerical value for the mean, median, or best estimate value. Upper and lower bound entries provides the actual numerical values, respectively. If the error factor is available it is given in the fifth entry. Upper and lower bounds and error factors are not always available, therefore n/a (meaning not available) is used instead.

7. Repair time is the next category. It indicates the average repair time associated with a component failure. It is also very seldomly found in generic sources. Some sources provide duration based on recorded repair times, in others repair times are a mean value of several maintenance durations on a particular component. For the real case generic information of this kind is not of much use.

8. Source is presented in two entries, one indicating the exact source (name of publication, page #., table #.) if available. The second entry gives information about the ultimate source of data (e.g. expert opinion, operating experience).

9. Component boundary is one of the problem areas to be addressed later. Very few sources provide adequate information about component boundary. The best information is found in the Swedish Reliability Data Book, where a sketch is made for each component. Whenever this information was not available,"detail not available" was written.

10. Comment entries are the last category of a record. Usually all the information found in the sources and considered relevant is written here. That includes the prior source and/or mean, if the data source is an updated generic, the operating experience (total population covered, number of demands or operational time, number of failures), additional failure rates relevant to the component (with or without command failures) etc. Practically all information which could by any means clarify failure rate, failure mode or component description are written here. The comment category is an integral part of each record and very important when choosing any record for further calculation or comparison.

Table 2. presents the complete record form.

Table 2: Complete record form

CODE	10 spaces
TYPE	65 "
SUBTYPE	65 "
DETAILTY	65 "
DETILTY1	65 "
OPMODE	30 "
OPENVIRO	65 "
GENFAILMOD	50 "
FAILMODE	50 "
FRATEDESCP	30 "
FAILRATE	10 "
UPBOUND	10 "
LOWBOUND	10 "
ERRORFCTOR	10 "
REPAIRTM	10 "
SOURCE	30 "
ULTSOURCE	65 "
COMMENTS	65 "
COMMENTS1	65 "
COMMENTS2	65 "

2.3. Coding system

Coding system is the area where the PSAPACK code mostly influenced the IAEA Data Base structure. In accordance to the PSAPACK requirements, each record code could have 4 alphanumeric characters. The fifth character is the character describing the source but the PSAPACK does not use this information (table 3).

Table 3: Coding system example

COMPONENT CODE	DESCRIPTION
K T A I W	fuse, general, fail to open, WASH 1400
P M D R T	pump, motor driven, centrifugal horisontal, flow rate 130-200 kg/s, fail to run, Swedish Reliability data book
R R A C E	relay, protective, all types, fail to close,IEEE 500
V C B O F	valve, self operated, check, less than 2 inches diameter, fail to open, CANDU assesment.

Because each record must have its own unique code and there are component categories which should have the same code, longer (more characters) code would have been preferable. However, some of the fault tree analysis codes included in the PSAPACK limit identification of basic event to eight alphanumeric characters. Moreover, it was felt that at least 4 characters are needed for further identification of components (including its physical position for eventual common cause or dependency analysis). Therefore only 4 characters were used for basic component identification and failure mode description.

Originally, there were more than the 100 different failure modes. That number required 2 characters for coding, leaving only two characters for the component type. As the generic failure modes were designed, one character was sufficient to describe the component failure mode.

Three alphanumeric characters were then used for the component caracterisation. For the components types with many subdivisions (for example valves), the first character is unique for the component type, the second is unique for subtype (for example **'v'** is valve and **'vm'** is motor operated valve). The last position characterizes the detail types 1 and 2. No firm rule exist for the last position. Usually when the detail type is 'general' or no further division exists, character **'a'** is in the third position.

For the components with few subdivisions, the first two positions characterize the type (e.g. **'lt'** is transmitter), while the last one characterizes the subtype or detail type, if any.

For the components which are 'one of a kind' all three positions characterize single component type (e.g. **'xmc'** stands for manual control device).

In the IAEA Data Base there are about 450 different components listed by component identification code. They are divided in 76 types. As there is no space in the coding system for operating mode or environment, sometimes the same component is coded differently because of operating conditions (e.g. motor driven pump without further subdivision is coded **'pmb'** when in alternating operating mode, and **'pmr'** when in running mode).

2.4. Data selection

After choosing the data sources and defined the record format and coding system, the actual data input was performed. Some of the sources (like 'WASH 1400' or 'IREP') were included completely, while in others (like 'IEEE 500' or 'Old W PWR') data for inclusion were carefully chosen.

From some of the data bases, data for components which are plant specific and are not comparable with any other (e.g. emergency AC source-hydro unit) were excluded. Also, the data directly adopted from any other generic data base (which is also in the IAEA Data Base) were not included.

The IEEE Standard 500 provide single source values and aggregate values. In the IAEA Data Base, depending on each particular case, single or composite values or even both (when particularly illustrative) were included. If 'per demand' and 'per hour' failure rates are provided, both were included, along with the source where the data are coming from.

Sources like NUREGs providing failure rates for pumps and valves usually present two values, namely: with command failures and without command failures. Values given in the failure rate entry are usually without command failures (clearly stated in the comment entry). The "with command failure" rate is also cited in the comment entry.

These sources often divide data in categories in accordance to NSSS vendor. In the IAEA Data Base usually the overall value is given, what is than described in comment entry.

2.5. Data extraction from the Data base

The IAEA Generic Data Base was created using the IBM-PC software dBASE III, so it can be stored in data base or in the plain text format. Therefore its use is not limited to PASPACK.

The PSAPACK package provides its users with the options for browsing through and retrieval of data from the IAEA Data Base. Retrieval of data can be accomplished by knowing the exact code of the record of interest. Then the complete record or the entries of interest can be retrieved The second way of data retrieval is to view the Data base record by record, and than retrieve information entering the particular record number. The PSAPACK code also includes a small data base editor which allows the user not only to retrieve data but also to change, modify, add or delete any information. By retrieving data and combining them with data which were added or modified (if any) the PSAPACK form its own small data base which is then used for solving a particular problem.

Using the dBase III code it is even easier to 'search' for or to 'locate' a certain code, type, subtype or any other relevant information. Browsing through the chosen fields, after locating a particular set of components of interest, one can easily compare any of the values included in the IAEA Data Base.

3. PROBLEM AREAS CONNECTED WITH GENERIC DATA BASES

When using a generic data base one has to be aware of possible problem areas. Considering the areas where misinterpretation can occur, the following 4 areas have been identified in the folowing order of importance:

-component boundary definition
-failure mode definition
-operating mode definition
-operating environment definition

Even when deriving failure rates from raw data from the plant being analyzed, these are issues which can lead to substantial errors.

During the development of the IAEA Generic Data Base insights were gained in how different data bases address each one of these issues. These insights and possible ways of avoiding or solving such problems are addressed next .

3.1. Component boundary

It is obvious that a main source of misinterpretation is the component boundary definition. Some of the experts agree that variations in component boundaries are the primary reason for failure rate fluctuation between sources. Although that statement seems to be too rigid, component boundaries could, depending on the particular component, change failure rates substantially.
It is therefore interesting to see how different sources address this issue.

Probably the best defined component boundaries are in the Swedish Reliability Data Book, because practically each component category has a sketch exactly indicating the component boundary and points of interface with other systems or components. Usually, in the component boundary, local control and protection (if any) are included.

Some of the 'NUREG' documents also have adequately defined component boundaries, with precise definition of interface points.

Other sources are defining a component as being an "off-the-shelf" item. This is an interesting and remarkable definition, but it assumes that "off-the-shelf" items have the same meaning everywhere, what is not necessarily the case for all the components.

Data bases which are part of PSAs, usually do not provide detailed definition of the component boundary. This is understandable, because these sources were compiled for specific use. When performing data updating, component boundary gain importance because of the need for matching the prior with the plant specific operating experience.

The sources which base their failure rate upon the combination of nuclear and non-nuclear experience (or even expert opinion) do not provide detailed boundary description. The level of similarity of different sources combined is not known, but it can be expected that certain differences would exist.

For the sources mostly based on expert opinion, the question of strictly defined boundary becomes a more academic one. However, cases like lube oil being part of diesel or breaker included or not in a pump boundary must be addressed to avoid significant (orders of magnitude) variations in the failure rates.

One way of avoiding serious problems with component boundary definitions is to define 'generic' component boundaries. That, of course, does not help in already existing data sources, but could save considerable trouble in the future. However, this is mainly applicable to data collection efforts undertaken during the performance of a PSA. In that case component boundaries should reflect two, sometimes opposite, requirements: the level of detail needed (or wanted) by the system model and the level of detail of plant records where raw data are retrieved from.

There are generally three major interfaces to be defined in connection with the component boundary definition, namely:

- **mechanical interface (incl.cooling system,lubricating system, etc. where appropriate)**
- **power supply interface**
- **control system interface**

3.2. Failure mode

Component failure mode is another problem area, although of a different character than the boundary definition. Failure modes found in various sources show significant difference even when describing basically the same failure. For example, in the sources which were included in the IAEA Generic Data Base over 100 different failure modes were found. Difference between some of this failure modes is basically in wording (e.g. fail to run vs. failure to run) and it is therefore easy to understand that they describe the same failure. In other cases it is sometimes difficult to understand the exact failure mode and compare it among sources

To compare failure modes and also to enhance the IAEA Generic Data Base coding system, considerable effort was undertaken to define generic failure modes.

In addition to the component design and function in the system, there are three basic component operating modes which affect the failure mode. These are:

-standby
-alternating
-continously operating.

There are also two distinct failure rate definitions. One is time related (further divided in standby and operating hourly rate) and the other is demand related. These were also taken into account while determining, for each single component (or group of components) possible ways (modes) of failure. All that served as the bases for defining generic failure modes.
Finally the original failure mode was included under one of the generic categories.

The disadvantage of the approach described is that it opens the way for inconsistencies in the grouping. For example generic definitions "failure to function' and 'failure to operate' describe basically the same failure mode, but while first is defined as per hour, the second is per demand. Because some sources define 'failure to operate' as per hour value, while others define it as a per demand value, the same (in words) failure mode is listed in different generic categories. Another possible complication comes from sources like NUREG 2815, Baseline Data, where all failure rates are defined per hour, while some of them are actually demand related. Therefore a generic failure mode, such as 'fail to start' (which is defined per demand) include per hour failure rate comming from that source.

The failure mode 'all modes' deserves special attention because it is usually a composite failure mode, actually containing several 'single' failure modes. The problem here is that each component usually have different failure modes. Whenever possible failure modes contained in 'all modes' are listed in the comment entry of the IAEA Generic Data Base record form.

Generic failure modes as proposed in the IAEA Generic Data Base are one of the possible ways of defining them. It is, however, not unique and it would be indeed possible to define them in several other ways. Altogether 28 failure modes were defined. The 19 of them considered most important, are shown in Table 4. Nine others cover minor number of peculiar failures like 'overheated' or heat exchanger 'tube' or 'shell' leak.

Table 4: Generic failure modes

FAILURE MODE	FAILURE MODE CODE
ALL MODES	A
DEGRADED	B
FAIL TO CHANGE POSITION	C
FAIL TO REMAIN IN POSITION	D
FAIL TO CLOSE	E
FAIL TO FUNCTION	F
SHORT TO GROUND	G
SHORT CIRCUT	H
OPEN CIRCUT	I
PLUG/RUPTURE	J
SPURIOUS FUNCTION	K
FAIL TO OPERATE	L
FAIL TO OPEN	O
PLUG	Q
FAIL TO RUN	R
FAIL TO START	S
RUPTURE	T
OTHER CRITICAL FAULS	X
LEAKAGE/EXTERNAL LEAK	Y

3.3. Operating mode

Component operating mode is of importance for active components, while generally have much less meaning for passive components. Even for active components there are cases where the operating mode has more or less importance, depending primarily on the way and mechanism of how the failure occurs.

Obviously operating mode is of great importance for pumps and other components which perform their function by continuously moving. These components have operating modes defined in three cathegories:

standby,
alternating and
running (operating).

For components which perform their function changing between discrete states, (e.g. valves), operating mode as defined above is actually status of the system they belong to. Operating mode pertinent to the component itself should be normally open or normaly closed position.

The majority of the sources do not define the component operating mode. The only sources which define operating mode are some of the NUREG LER sources.

PSA studies used as the data sources usually define the system where the component is located. For most of the systems it is possible to determine the operating mode, what could be used for defining active components operating mode.

Although not directly connected with the operating mode, one very important characteristic which sometimes is overlooked is the duration of the operation. For standby components, if the failure rate is determined based on operating experience, it is based on recorded operation during test performance, what is usually one or several hours. In the real case, particular components are required to operate for times which substantially differ from the one which was the base for the failure rate determination. Most of the sources do not address that problem.

When modeling standby components, failures during standby must be accounted for. Failures occurring during standby are not revealed until a test or an actual component demand, therefore are usually included in the model as a demand related failure. In this cases the demand related failure should comprise those failures whose mechanism is purely related to the demand (e.g. high current to motor windings during start) and also failures related to the time which the component spent in a standby condition.

However,if data base provides only demand related failure rate without indication how long is the componet in standby between two demands, this overlooks the fact that component failure during standby is time related and could vary substantially with variation in time between tests or actual demands.

Some of the sources recognized this fact and provide hourly failure rate for standby condition. On the other hand that approach is a possible source of error, because it is normally impossible to distinguish between time and demand related failures.

3.4. Operating environment

As mentioned earlier, the component operating environment is rather poorly defined in most of the sources. Most of the sources do not address it at all, while some of them are defining environment as the normal power plant environment. This definition could basically hold for normal operation or accidents which do not change environmentally affected parameters. However, when performing a PSA one is interested to predict the outcome of accident in environments, that could in certain cases change component failure rates substantially.

WASH 1400 is a source which provides separate failure rate for post accident situation for pumps and motors. The IEEE Standard 500 lists the environment multipliers for most of the components included, for environmental effects like high radiation, humidity, temperature and pressure.

Environmental effects could obviously affect component failure rate in different manners, therefore careful consideration should be given to this issue. Data from plant operating experience assume a normal environment, because operating experience data are normally either from normal operation or from test data, both of which are quite different from accident conditions.

On the other hand, the number and types of components affected by post accident conditions are usually rather limited. The extent of that is greatly dependent on plant design and type of accident.

Other type of extreme environment condition which can occur in NPP-s are high temperature condition occurring after the failure of room cooling systems. For most of electronic components or systems it is relatively easy and accurate to predict the effects of extreme environment and experimental data is available. For mechanical components like pumps, high temperature condition and consequently accelerated failure rates are relatively more complicated to predict.

4. CONCLUSION

Generic component reliability data is indispensable in any probabilistic safety analysis. It is not realistic to imagine that all possible component failures and failure modes modeled in a PSA would be available from the operating experience of a specific plant in a statistically meaningful way.

The degree that generic data is used in PSAs varies from case to case. Some studies are totally based on generic data while others use generic data as prior information to be specialized by plant specific data. Most studies, however, end up in a combination where data for certain components come from purely generic data and for others from Bayesian updating.

The IAEA effort to compile a generic component reliability data base aimed at identifying strenghts and limitations of generic data usage and at highlighting pitfalls which deserve special consideration. It was also intended to complement the PSAPACK package and to facilitate its use.

Moreover,it should be noted, that the IAEA has recently initiated a Coordinated Research Program in Reliability Data Collection, Retrieval and Analysis. In this framework it is expected that the issues identified as most affecting the quality of existing data bases would be addressed and alternative solutions proposed. In particular the following areas are being adressed:

-**Component Failure Data Collection System**(PR China)

-**Development of Data Collection System of Reliability Data of NPP Systems, Components and Events Important to Plant Safety**(FR Germany)

-**Bayesian Analysis Under Population Variability** (Greece)

-**Develpment of Methods and Procedures for Collection and Analysis of Data for Probabilistic Safety Assesment**(Hungary)

-**Inteligent Interface for Database Interrogation by Making use of Fuzzy Sets and Possibility Theory**(EC JRC-Ispra)

-**Reliability Improvements and Experimental Reliability Determination of Nuclear Reactor Instrumentation**(Romania)

Reliability Data Collection and Analysis(GB)

Contributions in Field of Data Collection and Analysis for Probabilistic Safety Analysis Application(USA)

Finally, it should be mentioned that the work described in this paper is being complemented by an extensive quality assurance, where all the records are being reviewed. A comprehensive IAEA Data Base user's guide is also under preparation. Related to the PSAPACK package, the small data base for benchmark calculations or test problems is going to be extracted from the complete IAEA Data Base. This small data base should to contain about 60 records, including all the components and the failure modes which are tipically considered in the PSA studies.

DEVELOPMENT OF RELIABILITY DATABASES AND THE PARTICULAR REQUIREMENTS OF PROBABILISTIC RISK ANALYSES

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Abstract

Nuclear utilities have an increasing need to develop reliability databases for their operating experience. The purposes of these databases are often multiple, including both equipment maintenance aspects and probabilistic risk analyses.

EDF has therefore been developing experience feedback databases, including the Reliability Data Recording System (SRDF) and the Event File, as well as the history of numerous operating documents.

Furthermore, since the end of 1985, EDF has been preparing a probabilistic safety analysis applied to one 1,300 MWe unit, for which a large amount of data of French origin is necessary.

This data concerns both component reliability parameters and initiating event frequencies. The study has thus been an opportunity for trying out the performance databases for a specific application, as well as in-depth audits of a number of nuclear sites to make it possible to validate numerous results. Computer aided data collection is also on trial in a number of plants.

After describing the EDF operating experience feedback files, we discuss the particular requirements of probabilistic risk analyses, and the resources implemented by EDF to satisfy them.

1. INTRODUCTION

Since the end of 1985, EDF has been working on a probabilistic safety analysis of one of the 1,300 MWe units of its PWR facilities (ref. 1).

To complete this project, a large amount of data is necessary, to obtain which EDF is systematically drawing upon its operating experience.

Particular use has been made of the vast databases represented by the Event File and the Reliability Data Recording System (SRDF).

Probabilistic risk analyses are nevertheless very demanding in terms of quality (nature and content of raw data) and in terms of quantity (size of sample and availability of general and specific data concerning the site studied). This being the case, the raw data must be as complete as possible and its recording must be supplemented by numerous in-plant audits.

2. OBJECTIVES

Concerning reliability data for use in probabilistic studies, there are, for EDF, two principal objectives.

The first objective is that of obtaining, from operating experience with EDF PWR units, a range of validated reliability parameters relating to the different items of equipment used in the studies. This involves:

- Establishing a list of the equipment allowed for in the reliability studies,
- Monitoring of this equipment in the units,
- Validation of the data obtained.

For monitoring of equipment, use is made essentially of SRDF, but also of the EDF-SPT Event File and of various operating experience feedback documents such as incident and defect reports.

Validation is carried out by analyzing samples of failures and by in-plant audits.

The second objective is that of guaranteeing the durability of the data. The requirement is for monitoring and periodically revising the information, involving a complex data processing setup.

3. MEANS

3.1 Nature of data

The information is of the following two types:

- Raw data

  An item of raw data is essentially a description of a phenomenon: failure, damage, repair, circumstance etc. This description is often made in a structured manner using a set of questions and answers, in which the answers may be pre-established. A free description frequently accompanies this coded description.

- Processing results

  The raw data is subjected to mathematical processing to obtain reliability parameters:

  - failure rates (in operation, on demand etc.),
  - the durations of repair, failure, unavailability etc.

Depending on the nature of the objectives of the studies and the way in which systems are modeled, the interpretation of raw data for subsequent processing may be of primary importance.

3.2 Sources of raw data

The principal source of raw equipment reliability data is SRDF.

The raw SRDF data is captured directly at the nuclear plants in the form of:

- descriptive reports of failures,
- equipment operating tables.

The principles of the collection system are:

- maximum decentralization,
- data capture based on criteria as simple and reliable as possible.

A list of the equipment monitored is drawn up, and clear definitions are given.

The records cover the greater part of the electrical and electromechanical equipment. A start was made in 1978 on the Fessenheim and Bugey sites, in 1984 scope included the CP1 and CP2 standardized 900 MWe units and in 1986 it included the 1,300 MWe units. Some 600 components are monitored, a large number of which are parts of safety-related equipment.

The data flow is of approximately 100 to 150 failure records per unit per year, involving the work of two or three technicians at each site.

At the end of 1986, SRDF contained some 110 reactor-years of observation and 11,300 records of which:

- 2,600 concern pumps and motors,
- 5,000 concern valves.

In addition, there is a system designated SRDF-A for the collection of electronic equipment failures ("SPIN and Controbloc", I and C systems, Plant computers etc.) from 1,300 MWe units.

This system was put into service at the same time as the first EDF 1,300 MWe units, i.e. 1984. The failures are entered by maintenance technicians at the sites. Each record is a subject of analysis with the manufacturers and checking at power plant level.

At the end of 1987, the observation period ran to approximately 20 reactor-years, the file containing about 2,500 records of failures in electronic equipment of all types.

Finally, the EDF-SPT Event File makes it possible to find a large number of failures which have affected safety-related equipment or caused unit unavailability.

In short, the Event File represented, at the end of 1987, 185 reactor-years for the 900 MWe units and 19,500 records, including 1,200 scrams, 1,000 scheduled shutdowns and 500 turbine trips.

3.3. On-site audits

The collection and processing of data from national archives must be supplemented by in-plant audits in the following cases:

- equipment not monitored,
- initial sample insufficient,
- results not offering a sufficient guarantee.

The audits involve the analysis of the operating and maintenance department archives.

3.4 Scanning of magnetic tapes of unit computers

To supplement and validate the conventional approach for data collection, software for scanning unit computer magnetic tapes are under development. The objectives of these programs are:

- carrying out automatic calculations of the number of actuations and the operating times of the components covered in SRDF,
- obtaining all the standard reactor states, hour-by-hour,
- finding protection system actuations,
- analyzing and monitoring the automatic sequences (for example the safety injection sequence).

Systematic processing of the computer tapes has been carried out since June 1987 at the Saint-Laurent B site.

In the 1,300 MWe units, this processing shall be linked to scanning the records of the tagging assistance computers. This makes it possible to determine, with a high degree of accuracy, outages of all items of equipment, particularly those which are safety-related.

4. REQUIREMENTS OF PROBABILISTIC STUDIES AND PROCESSING METHODS

To carry out the 1,300 MWe unit probabilistic safety study, it is necessary to procure a great amount of reliability data (failure rates, unavailability etc.) from EDF operating experience feedback.

In this field, the use of raw SRDF data is systematic. Nevertheless, the direct use of reliability parameters given by this system is not possible as, on the one hand, the failure definitions used for the probabilistic safety study are more restrictive and, on the other hand, the samples must be carefully checked (size and representativeness). In practice, the processing of qualified data therefore involves re-examination of all raw data.

The collection system is necessarily decentralized, which entails complications.

Even with precise definitions and unambiguous criteria, the description of a failure frequently involves interpretation, which can result in errors in terms of the objectives of the studies and lack of homogeneity in the records.

There are been cases of definitions resulting from maintenance or operating practice being substituted for theoretical definitions. Thus, when equipment to which technical operating specifications apply is tagged out for repair, the plants generally assume that the equipment has completely failed (such equipment is not made unavailable without a good reason), whereas in certain cases it could perform its function.

In consequence, in a reliability study, it is necessary to check the coherence and homogeneity of the data on the basis of all the descriptive material concerning the failure.

In addition, a certain number of items of equipment may have insufficient numbers of operating hours or numbers of actuations, and it is necessary to establish groups on the basis of study of the samples concerned.

The analysis of failure records before the calculation of parameters in this manner makes it possible to obtain homogeneous and realistic data, due to low interpretation dispersion resulting from the judgment of the engineers, and to rigorous selection of complete and pertinent failures, as concerns probabilistic safety studies.

The analyses necessitate the study of raw records and extensive examination of plant operating archives. Nevertheless, at the present time, the use of computer records of the units (maintenance, management of works and tagging out) makes it possible to rapidly obtain a large amount of information. It would appear that in the future, it will thus be possible to make considerable progress in the field of equipment reliability data.

5. RESULTS

As part of the 1,300 MWe unit probabilistic safety analysis, a list has been drawn up of about 140 electromechanical components specific to EDF PWR power plants for which reliability parameters of French origin are sought using all the EDF operating experience feedback resources.

This work has provided nearly 470 items of reliability data concerning failure rate parameters (in operation and on actuation) corresponding to different modes and repair durations. The common-cause failures have been quantified for 30 generic components. Finally, the unavailability rates for preventive or curative maintenance have been established for about 30 components or safety functions.

The following examples illustrate different types of results obtained in the 1,300 MWe unit probabilistic safety analysis.

In the field of reliability parameters (failure rates in service, on demand or repair time), a comparison is made with the results obtained in other studies. In the case of the auxiliary feedwater pumps and the standby diesel generators, the following table is obtained.

	EDF 900 MW	PRA Oconee	PRA Connecticut Yankee	WASH 1400
Feedwater pumps				
λ (per hour)	2.4×10^{-4}	2×10^{-5}	2×10^{-5}	2.4×10^{-4}
γ (per demand)	2.5×10^{-3}	5×10^{-4}	4×10^{-3}	10^{-3}
τ (in hours)	20 h	—	40 h	—
Standby diesel generators				
λ (per hour)	3×10^{-3}	10^{-3}	1.3×10^{-3}	8×10^{-3}
γ (per demand)	1.5×10^{-3}	4.6×10^{-3}	5×10^{-3}	3×10^{-2}
τ (in hours)	9 h	—	—	—

In this table, a comparison is made between the results obtained by EDF from the Event File and by INPO using LER and SER, in the field of analogue measurement systems.

λ/h	EDF (ref. 3)	INPO (ref. 4)
Nuclear flux measurements	6×10^{-6}/h	8×10^{-6}/h
Pressure measurements	2.8×10^{-6}/h	3×10^{-6}/h

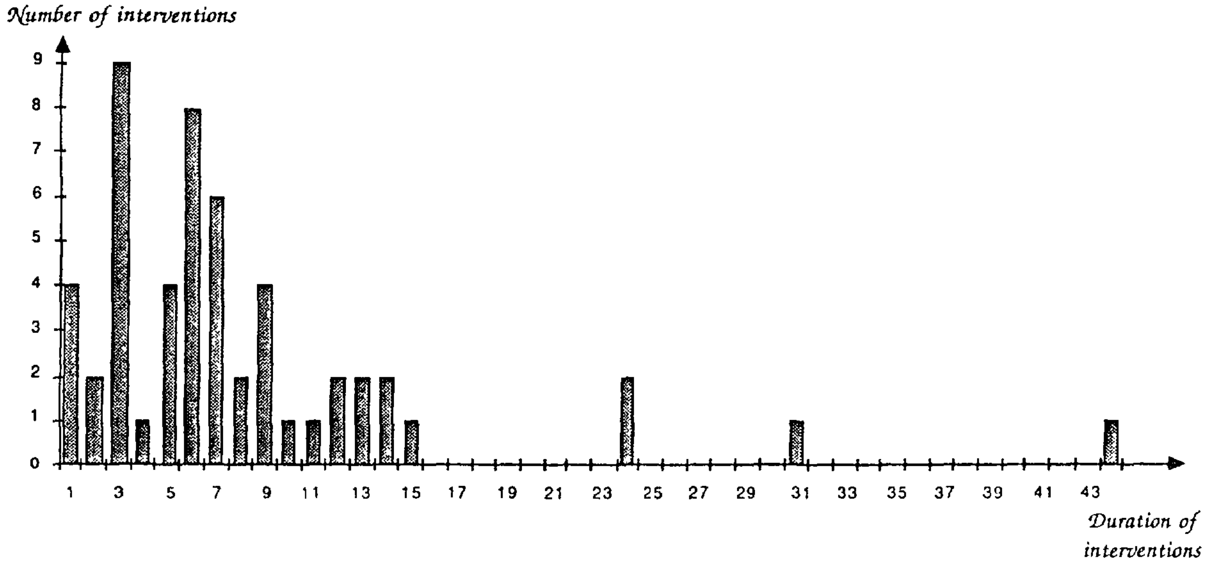
The results match extremely well.

In the same way, a similar comparison can be made of unavailability rates of safety-related equipment:

	EDF 900 MWe unit experience feedback	Audit on 1,300 MWe unit site	US data (source INPO ref. 5)
Motor-driven auxiliary feedwater pumps	5×10^{-3}	5×10^{-3}	7×10^{-3}
Standby diesel generators	3.3×10^{-3}	7.8×10^{-3}	2×10^{-2}

Finally, an example of an audit of equipment maintenance is given for the duration of maintenance of standby diesels on a 1,300 MWe unit site.

These examples demonstrate that the different sources of data show considerable coherence, but care must nevertheless be taken not to underestimate the difficulties in obtaining and validating data.



Number of interventions outside unit shutdown on diesel engines as a function of their duration (average duration 8.4 hours)

6. CONCLUSION

It is clear that EDF PWR unit operating experience is extensive, and is sufficient for the successful collection and processing of reliability data to be envisaged.

EDF has provided itself with large-scale resources for attaining this objective by the implementation of operating experience feedback management tools (SRDF, Event File, unit computer magnetic tapes etc.).

Nevertheless, the requirements of probabilistic risk analyses are considerable, specifically:

- Guarantee of comprehensiveness:

  It must be made sure that all failures pertinent to the study have effectively been recorded at the sites.

- Validity of the operating tables:

  The calculation of quality reliability parameters involves obtaining certain numbers of hours of operation and of activations of validated equipment.

- Interpretation of raw data:

  A certain number of failures involve difficulties in evaluating potential consequences with regard to the safety functions of the equipment.

  In such cases, expert opinion is primordial, and can affect the results.

- Adequacy for system modelling:

  The nature and the modes of failure must be compatible with the choices made in the system studies.

It would appear that in the future, it will be possible to make considerable progress by using the site computer systems. This can take place from the level of detection of failures and unavailabilities for maintenance, and can contribute to the quality of the parameters by accurate counting of the operating times and the actuations.

It is this that EDF is currently working on, and from now on EDF will only use its own operating experience feedback data in its probabilistic studies.

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CHARACTERISTICS AND USE OF THE COMPONENT EVENT DATA BANK

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Abstract

The Component Event Data Bank (CEDB) is a centralized bank collecting, at the European level, data describing the operational behaviour of components of Nuclear Power Plants (NPP's) operating in various European countries. It is one of the three event-data banks of the European Reliability Data Systems (ERDS), developed and managed by the JRC-Ispra (the other two banks being the Abnormal Occurrence Reporting System and the Operating Unit Status Report) . The CEDB stores information on the operational history (operational times and/or number of demands of intervention in a year, failure-events reports) of components of NPP's well identified by their engineering and operation characteristics.

As of December 1987, the CEDB contains data from about 5200 mechanical and electromechanical components, pertaining to 21 component-families and 51 engineering systems, monitored for an averaged time of 5 years in 10 LWR Units (and in the steam-water cycle of other reactor type Units or conventional Units) located in 7 European countries; the failure-events recorded are about 4200.

The CEDB receives data either from national banks, such as the EDF-SRDF and the VATTENFALL-ATV, or directly from the NPP's.

The CEDB (as well as the whole of the ERDS) has as main objective the promotion of safety. In particular it has been conceived as a support to the analyst in his safety assessment for the design of a new NPP or the backfitting of an old one. By putting together the operational experience of European NPP's, it was the intention to create a database of raw data, to be conveniently processed, in order to:

- improve the credibility of existing estimated reliability parameters by exploiting the necessary feed-back from plant operation;
- provide a solution to one of the major problems of the reliability analyst; namely the wide spread in reliability parameters existing in the current literature, especially for mechanical and electromechanical components;
- allow comparison between the performance of components of plants of different countries.

This paper shortly describes:

- the main features of the bank classification scheme, its data retrieval capabilities and on-line statistical processing programmes;
- some improvements under study to better meet PSA needs.

Some examples of on-line data treatment are given and commented. The structure of an organized output (a CEDB-Reliability Data Book), which is being implemented, is shortly illustrated.

The results of a study on linked multiple failure-events and some analyses based on the application of multivariate analysis techniques are summarized and the interest to continue to investigate along these lines is shown.

1. Data for Probabilistic Safety Assessment; aims and structure of the European Reliability Data System

The JRC started in 1978 the "European Reliability Data System" (ERDS) project, aimed at organizing in a series of data bases the operational experience of Nuclear Power Plants (NPP's) /1/. The ERDS, most of which has been operational since 1984, collects, organizes and disseminates at the European level, information on operation of NPP's, its main objective being the promotion of safety. In particular the supply to the analyst of data on component failures and on safety-significant abnormal events derived from operating experience and necessary for its safety assessment.

The ERDS has been structured into four data banks, three of which store raw event data, the fourth one being dedicated to the organization of reliability parameters. The three raw event data banks are:

a) the Component Event Data Bank (CEDB), which stores information on the operational history of safety-significant (or important for plant availability) components of some NPP's (component operating time and/or number of demands of intervention in the year, failure-event reports) /2,3/;
b) the Abnormal Occurrences Reporting System (AORS), which stores information on safety-related events;
c) the Operating Unit Status Report (OUSR), which stores information on Unit productivity and availability and on events which lead to a loss of generating capacity (hystogram of Unit power during the year, plant event description in free text and coded format). This bank is now based mainly on the PRIS-IAEA system.

The fourth bank, of a different nature, the structure of which is still in a study phase, is:

d) the Reliability Parameter Data Bank (RPDB), the purpose of which is the storage of reliability parameters of homogeneous classes of NPP components, derived from operational experience (CEDB) and literature sources (e.g. IEEE St 500).

2. CEDB structure and data processing

2.1 Introduction

The CEDB is a centralized bank collecting, at the European level, operational data characterizing the behaviour of components of Nuclear Power Plants (NPP's) operating in various European countries.

Working groups of experts have operated since 1978 to set up reference classifications and a bank structure well-suited to the general requirements of the ERDS project and capable of ensuring compatibility, when transcoding, with the following national data banks:

- SRDF/EdF (France)
- GRS/RWE (Germany)
- ENEL Data Collection System (Italy)
- SRS/UKAEA (Great Britain)

Reference has also been made to:

- NPRDS/INPO (USA)
- ATV (Sweden)
- LER/NRC (USA)
- GADS/NERC (USA)
- UNID/TVA (USA).

The CEDB, as well as the whole ERDS, has been developed using the DBMS ADABAS of Software A.G. on an AMDAHL 470/V8, running under OS/MVS, connected to the internal JRC TP network and to the external telecommunication network.

As of December 1987, the CEDB contains data from about 5200 mechanical and electromechanical components, pertaining to 21 component-families and 51 engineering systems, monitored for an average time of about 5 years in 10 LWR Units (and partly in the conventional part of other reactor type Units or in the steam-water cycle of fossil-fueled power plants) located in 7 European countries; the failure-events recorded are about 4200.

The CEDB stores data coming from the following three national data banks:

- the French EDF/SRDF;
- the Swedish VATTENFALL/ATV;
- the German GRS.

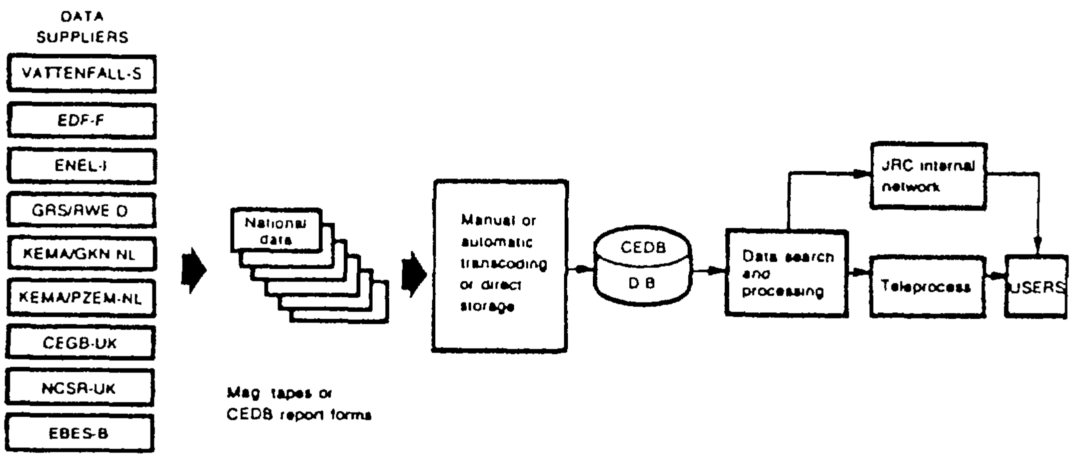
Original information coming from these reporting systems need to be transferred into the CEDB structure, i.e. homogenized in the same reporting scheme and language (English). A semi-automatic transcoding programme is being implemented for the transfer from SRDF to CEDB; a computer-aided transcoding programme already exists for the transfer from ATV to CEDB.

The following Organizations supply data to CEDB in the CEDB structure: ENEL (I), KEMA (NL), EBES (B), CEGB (UK), NCSR/UKAEA (UK). All Spanish Utilities have decided to adopt the CEDB scheme, to collect data in all their operating NPP's and to supply them to CEDB.

JRC is exchanging information with the National Nuclear Safety Authority of the People's Republic of China, which has chosen to implement a national reliability data system similar to the ERDS (CEDB and AORS) for data collection in their future NPP's.

By putting together the operational experience of European NPP's, it is intended to create a database of raw data, to be conveniently processed, in order to:

- improve the credibility of existing estimated reliability parameters by exploiting the necessary feed-back from plant operation;
- provide a solution to one of the major problems of the reliability analyst: namely the wide spread in reliability parameters existing in the current literature, especially for mechanical and electromechanical components;
- allow comparisons between the performance of components of plants of different countries.



Source: S. BALESTRERI, "The European Component Data Bank and its Uses", Reliability Data Bases, A. Amendola and A.Z. Keller (Eds.), Reidel Publisher, Dordrecht, 1987. PER 1388/87

Fig. 1 - Flow of information from the data suppliers, through the CEDB, to the users.

2.2 CEDB data collection scheme and report forms

2.2.1 Classification scheme

A comprehensive classification scheme for failure-event reporting /2,3/ results from the combination of the two following schemes:

- a Plant Classification Scheme, for the characterization of the context in which the event occurs (plant type, Reference Engineering System Classification, Component-family Classification);
- a Failure Classification Scheme, for the general characterization of the failure event, the following repair action and of its consequences at plant and system levels.

2.2.2 Plant Classification Scheme and Component Report Forms

The Plant Classification Scheme is based on the establishment of the following hierarchical levels for structuring the plant: component, system, system grouping.

a) component: it is a structure, or equipment, considered as an aggregate of mechanical and/or electrical parts, constituting a well identified element inside the plant. It has defined performance characteristics and can be removed and replaced within the plant. The CEDB classification covers about 40 component families (TABLE I). A piece-part reference list for each component family has been set up to enable the failure reporter to single out the part(s) of the component involved in a failure and to better characterize the ensuing repair action (TABLE II). The identification of a component in the bank is performed by specifying the following data: functional position in the plant, system to which it pertains, engineering (e.g. design) characteristics, operating characteristics (operating values of the "numerical" engineering characteristics), environmental conditions, mode of operation, maintenance type and schedule. Data collection forms are adopted by the CEDB (Figs. 2,3,4,5): the first two forms, the Component Report Form and the Operation Report Form, are filled in once, to characterize the component for which failure data will be collected.

Text continued on p. 226.

TABLE I : Component family reference classification list of the CEDB

ACTE	Electromechanical Actuator
ACTH	Hydraulic Actuator
ACTP	Pneumatic Actuator
AMPL	Amplifier
ANCT	Annunciator Modul/Alarm
BATC	Battery Charger
BATT	Battery
BLOW	Blower/Fan/Ventilator
BOIL	Boiler
CKBR	Circuit Breaker
CLTW	Cooling Tower
CLUT	Clutch
COMP	Compressor
CPCT	Capacitor
DLGE	Diesel-Generator set
DRYE	Air, Gas Dryer/Dehumidifier
ELHE	Electrical Heater
ELMO	Electric Motor
ENGI	Internal Combustion Engine
EXCH	Heat Exchanger
FILT	Filter
FUSE	Fuse
GENE	Electrical Generator
INCO	Instrumentation-Controllers
INST	Instrumentation-Field
INSU	Insulator
MOPU	Motor-Pump Unit
PIPE	Piping/Fitting
PUMP	Pump
RECT	Rectifier
RSTR	Resistor
SAVA	Safety/Relief Valve
STGE	Steam Generator
SUPP	Pipe-Support
SWIT	Switchgear
SWTC	Switch
TANK	Accumulator/Tank
TRAN	Transformer
TRSD	Transducer
TUGE	Turbine-Generator Set
TURB	Turbine
VALV	Valve (except safety valve)
WIRE	Electrical Conductor, Wire, Cable

Source: CEDB Handbook

TABLE II : Component family reference classification - component family VALV: boundary definition and piece part list

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Component family = VALV

Boundary definition:
Component boundary is identified by its interface with plant equipment, that is:
- body ends, connecting the component to the installation of which the valve is part.

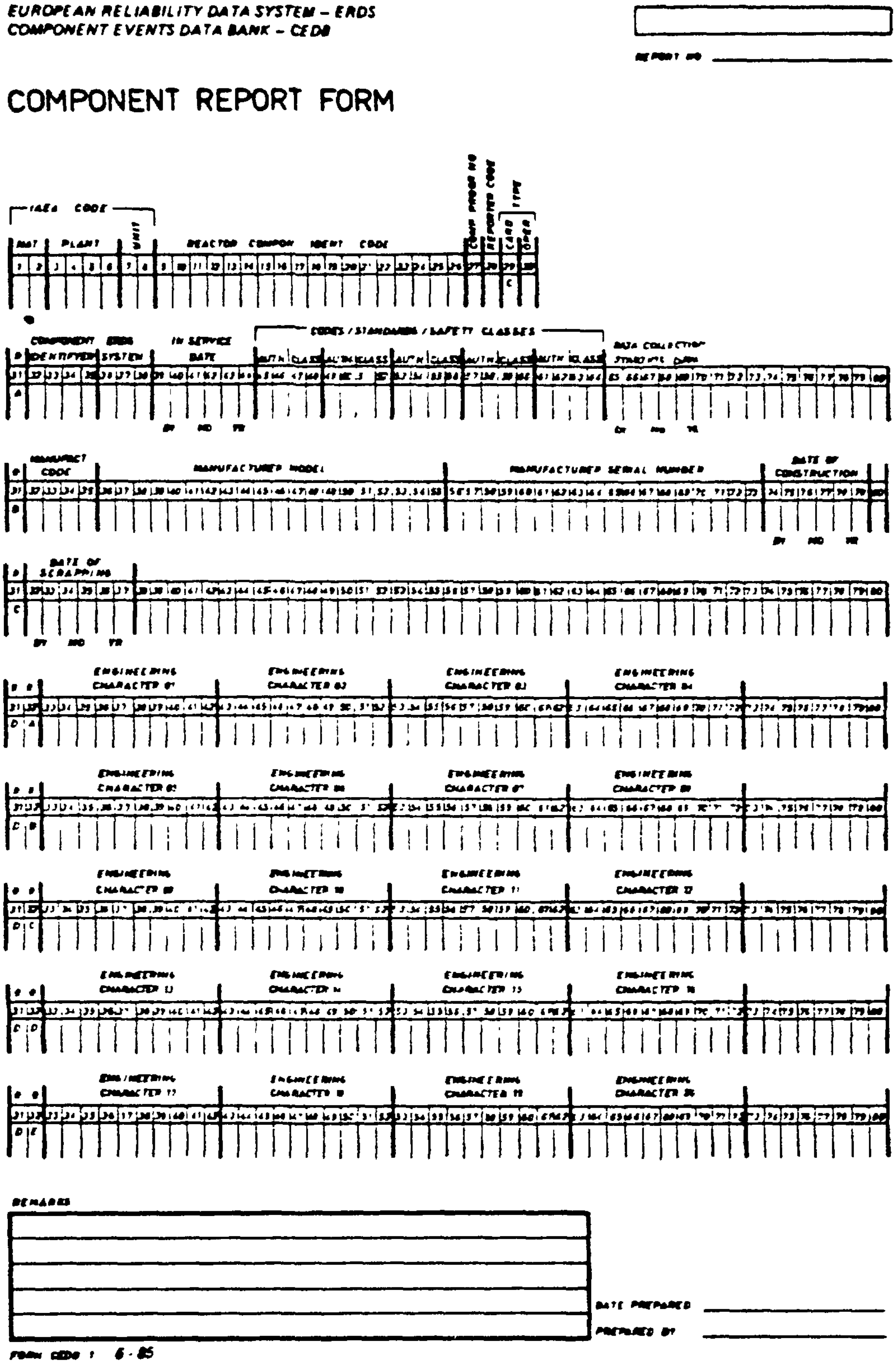
Within the component boundary are included:
- auxiliary devices, as far as they are dedicated to the'unit;
- protection and trip devices;
- instrumentation for the monitoring of the status of the component.

Outside the component boundary are:
- actuator;
- support/control devices.

Piece-part list

1	Body
11	Diaphragme
13	Trunnion bearings (ball valves)/shaft bearings (butterfly)
130	Regulating/setting devices
14	Positioner
15	Spring
16	Instrumentation/monitors/recorders
17	Damper/shock absorber (check valve)
2	Bonnet
28	Body-bonnet connection
29	Body seat
3	Stuffing box
30	Disc/ball/plug/wedge
32	Pipe connection
33	Supports
4	Body trunnion (ball valves)/hinge pin (check valves)/shaft (butterfly valves)
8	Stem
78	Limit switch
81X	Cooling system components (general)
82X	Protection devices (general)
82A	Protection devices: sensing elements
90X	Sealing (general)
90A	Sealing: packing gland/stem
90B	Sealing: body-bonnet
90C	Sealing: pipe connection flange
90D	Sealing: bellows/stem
999	Other

Source: CEDB Handbook


EUROPEAN RELIABILITY DATA SYSTEM - ERDS
COMPONENT EVENTS DATA BANK - CEDB

REPORT NO

COMPONENT REPORT FORM

IAEA CODE
NAT
PLANT
UNIT
REACTOR COMPON IDENT CODE
COMP. PROGR NO
REPORTED CODE
CARD
TYPE

COMPONENT IDENTIFIER
ERDS SYSTEM
IN SERVICE DATE
CODES / STANDARDS / SAFETY CLASSES
AUTH CLASS
AUTH CLASS
AUTH CLASS
AUTH CLASS
AUTH CLASS
DATA COLLECTION STARTING DATE
DY MO YR
DY MO YR

MANUFACT CODE
MANUFACTURER MODEL
MANUFACTURER SERIAL NUMBER
DATE OF CONSTRUCTION
DY MO YR

DATE OF SCRAPPING
DY MO YR

ENGINEERING CHARACTER 01
ENGINEERING CHARACTER 02
ENGINEERING CHARACTER 03
ENGINEERING CHARACTER 04

ENGINEERING CHARACTER 05
ENGINEERING CHARACTER 06
ENGINEERING CHARACTER 07
ENGINEERING CHARACTER 08

ENGINEERING CHARACTER 09
ENGINEERING CHARACTER 10
ENGINEERING CHARACTER 11
ENGINEERING CHARACTER 12

ENGINEERING CHARACTER 13
ENGINEERING CHARACTER 14
ENGINEERING CHARACTER 15
ENGINEERING CHARACTER 16

ENGINEERING CHARACTER 17
ENGINEERING CHARACTER 18
ENGINEERING CHARACTER 19
ENGINEERING CHARACTER 20

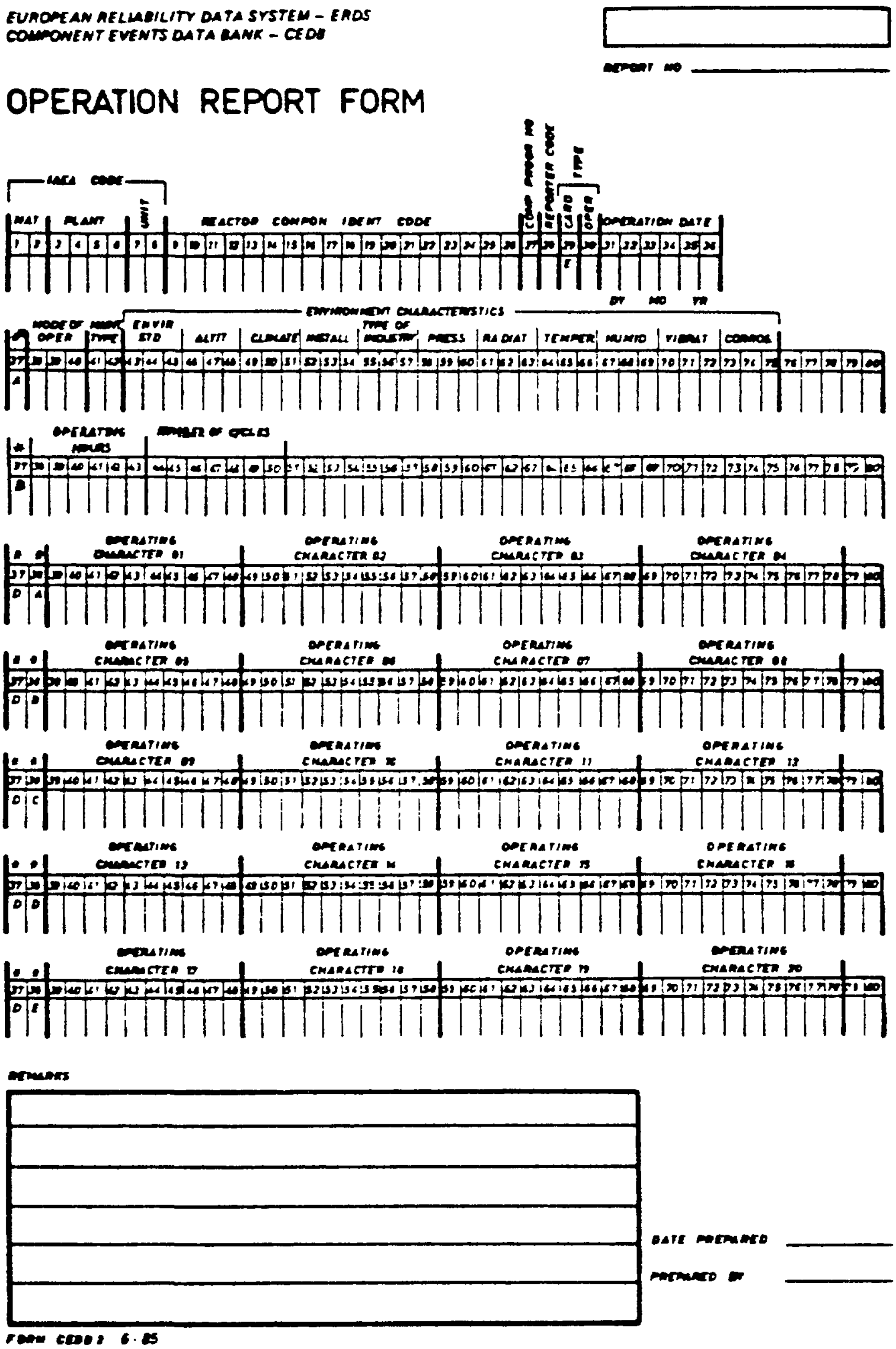
REMARKS

DATE PREPARED
PREPARED BY

FORM CEDB 1 6-85

Source: CEDB Handbook

Fig. 2 : Component Report Form


EUROPEAN RELIABILITY DATA SYSTEM - ERDS
COMPONENT EVENTS DATA BANK - CEDB

REPORT NO

OPERATION REPORT FORM

IAEA CODE
NAT
PLANT
UNIT
REACTOR COMPON IDENT CODE
COMP PROGR NO
REPORTER CODE
CARD
TYPE
OPER
OPERATION DATE
DY MO YR

ENVIRONMENT CHARACTERISTICS
MODE OF OPER
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CLIMATE
INSTALL
TYPE OF INDUSTRY
PRESS
RADIAT
TEMPER
HUMID
VIBRAT
CORROS

OPERATING HOURS
NUMBER OF CYCLES

OPERATING CHARACTER 01
OPERATING CHARACTER 02
OPERATING CHARACTER 03
OPERATING CHARACTER 04
OPERATING CHARACTER 05
OPERATING CHARACTER 06
OPERATING CHARACTER 07
OPERATING CHARACTER 08
OPERATING CHARACTER 09
OPERATING CHARACTER 10
OPERATING CHARACTER 11
OPERATING CHARACTER 12
OPERATING CHARACTER 13
OPERATING CHARACTER 14
OPERATING CHARACTER 15
OPERATING CHARACTER 16
OPERATING CHARACTER 17
OPERATING CHARACTER 18
OPERATING CHARACTER 19
OPERATING CHARACTER 20

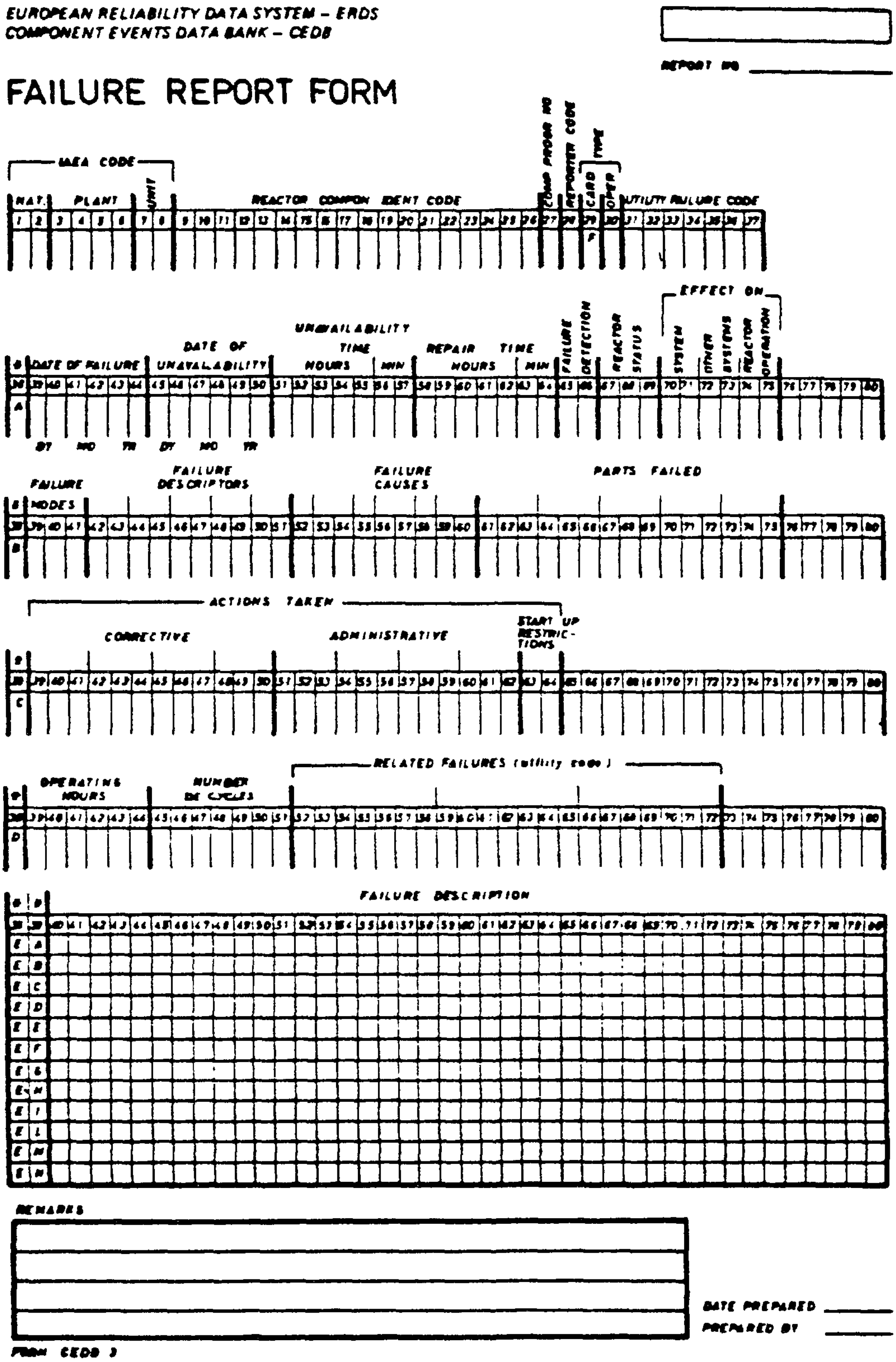
REMARKS

DATE PREPARED
PREPARED BY

FORM CEDB 2 6-85

Source: CEDB Handbook

Fig. 3 : Operation Report Form


EUROPEAN RELIABILITY DATA SYSTEM – ERDS
COMPONENT EVENTS DATA BANK – CEDB

REPORT NO

FAILURE REPORT FORM

IAEA CODE
NAT. PLANT UNIT
REACTOR COMPON. IDENT CODE
COMP PROGR NO
REPORTER CODE
CARD TYPE
OPER.
UTILITY FAILURE CODE

DATE OF FAILURE
DATE OF UNAVAILABILITY
UNAVAILABILITY TIME HOURS MIN
REPAIR TIME HOURS MIN
FAILURE DETECTION
REACTOR STATUS
EFFECT ON SYSTEM OTHER SYSTEMS REACTOR OPERATION
DY MO YR DY MO YR

FAILURE MODES
FAILURE DESCRIPTORS
FAILURE CAUSES
PARTS FAILED

ACTIONS TAKEN
CORRECTIVE
ADMINISTRATIVE
START UP RESTRIC-TIONS

OPERATING HOURS
NUMBER OF CYCLES
RELATED FAILURES (utility code)

FAILURE DESCRIPTION

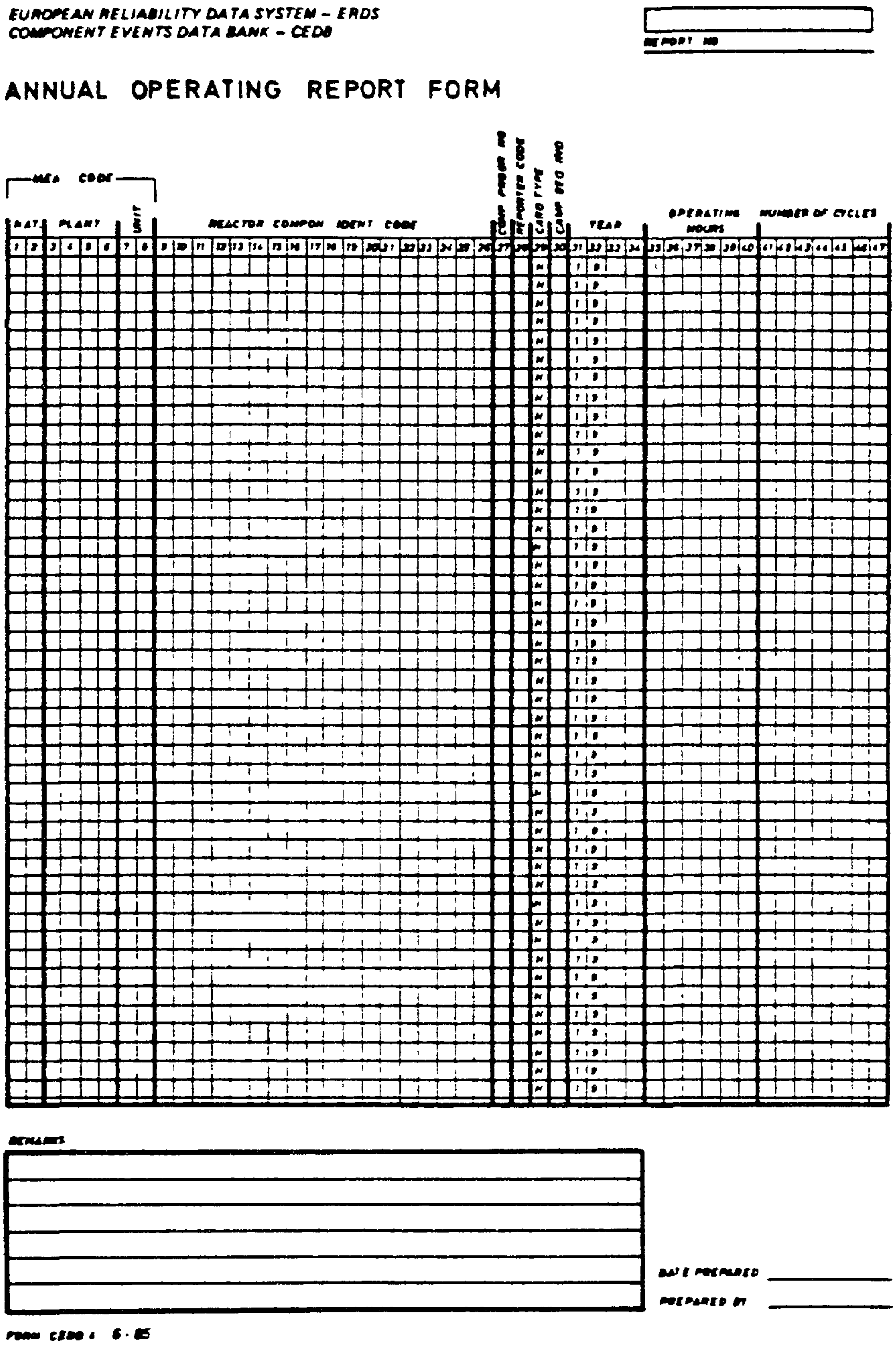
REMARKS

DATE PREPARED
PREPARED BY

FORM CEDB 1

Source: CEDB Handbook

Fig. 4 : Failure Report Form


EUROPEAN RELIABILITY DATA SYSTEM - ERDS
COMPONENT EVENTS DATA BANK - CEDB

REPORT NO

ANNUAL OPERATING REPORT FORM

AREA CODE: NAT. PLANT UNIT

REACTOR COMPON IDENT CODE

COMP PROGR NO

REPORTER CODE

CARD TYPE

CARD SEQ NO

YEAR

OPERATING HOURS

NUMBER OF CYCLES

REMARKS

DATE PREPARED

PREPARED BY

FORM CEDB 4 6-85

Source: CEDB Handbook

Fig. 5 : Annual Operating Report Form

Comment: Experience shows that differences in defining the boundary of the component (i.e. of the material which can contribute to its failure) are a major cause of variation in reliability parameters given by the various sources.

The CEDB makes in general a distinction between motive equipment (valve-actuator, turbine, electric motor, etc.) and driven equipment (valve-body, pumps, electrical generator, etc.), i.e. it classifies each of the above mentioned equipment items as components. In addition it classifies a few composite items (motor-pump unit, turbine-generator set, diesel-generator set).

Other banks monitor the aggregate (e.g. the GRS bank considers aggregates such as "valve + reduction gear + electric motor" and "booster pump + electric motor + over-geam + main pump").

b) system and system-grouping: a system is a set of mechanical, electric, electronic components univocally identified by whether:
   - it accomplishes a clearly defined function inside the plant,
   - it accomplishes more than one function, but in different plant operating modes (i.e. cold-shutdown, emergency, normal operation, etc.).

   A system grouping is a set of systems which are characterized by common properties, such as:
   - they can be framed in a more general logic function (e.g. the engineering safety features),
   - their functions are related to the accomplishment of plant-operating services (e.g. reactor auxiliary systems).

A Reference System Classification for LWR Units has been set up /4/; more than 180 systems, grouped into 13 system-groupings, have been singled out. General guidelines to set up a reference plant classification scheme are illustrated by /5/. The LWR classification has been extended to cover also GCR/AGR and PHWR.

2.2.3 Failure classification scheme and report form

The Failure classification scheme adopts the following basic definitions:

a) Component failure: is defined as the termination or the degradation of the ability of a component to perform a required function; it is a component malfunction which requires some repair. A "functional" unavailability (i.e. a lack of performance because of the absence of a proper input or support function) or a shut-down of the component due to conditions external to the process are not considered as a failure.

b) Failure mode: it is the effect by which a component failure is observed. Failure mode types are correlated with the component operation mode. The failure modes are subdivided into two general classes (TABLE IV):
   - "(not demanded) change of operating conditions (or state)" for components asked to accomplish a function during a certain time;
   - "demanded change of state not achieved, or not correctly achieved" for components which are called to operate on demand.

A set of reliability parameters (failure rate or failure-on-demand probability, repair rate) corresponds to each relevant component failure mode. The classification adopted by CEDB is presented in TABLE III. Among the codes related to failure mode on demand, codes A, B, C and F apply to components such as valves, breakers, actuators. Codes D, E and F apply to rotating components such as pumps, electric motors, diesel generators, etc. As far as the second class of failure modes is conerned, "change of operating conditions (or state) not required", two categories have been singled out, namely:
- degree of suddenness;
- degree of seriousness.

The first category describes whether the unavailability of the component is contemporary to the detection of the failure or of the abnormality or whether the unavailability of the component could be deferred. The second category refers mainly to the mode of change of the component function, i.e. to its level of degradation (no output, outside specification) or to its peculiarity (operation without request, erratic output).

TABLE III: Failure modes classified by CEDB

FAILURE MODE	
<u>On demand</u>	
A	Wont'open
B	Wont'close
C	Neither opens nor closes/does not switch
D	Fails to start
E	Fails to stop
F	Fails to reach design specification
<u>On operation</u>	
. suddenness	
A	Sudden failure
B	Incipient failure
C	Not defined
. degree of seriousness	
A	No output
B	Outside specifications
C	Operation without request
D	Erratic output (false, oscillat. instability, drift, etc.)

A failure-event is characterized by the failure mode, failure cause, failure descriptors, failure detection, parts failed, plant status at the time of failure, effect of failure on the system to which the component pertains, on other systems/components, on plant operation, corrective and administrative actions.

The repair action is mainly characterized by the repair time, the unavailability time, the parts involved in the failure, the corrective action taken (component replacement, overhaul, modification, etc.). All the above-mentioned information items on a failure-event are recorded by the data collector in a "Failure Report Form" (Fig. 4).

2.2.4 Annual Operating Report Form

The component yearly operating time and number of cycles/demands are recorded by the data collector in the Annual Operating Report Form (Fig. 5) once a year.

2.2.5 Some improvements under study to meet PSA needs

In order to make the CEDB as suitable as possible for PSA, the following areas are being investigated:

- enlarging the CEDB component classification, to make the CEDB able to cover all the equipment important for PSA (e.g. to monitor also the electric/electronic control equipment);
- better characterizing the actual function that some components, important for the safety, perform in the plant, by revising some engineering/operating attributes (such as the component application);
- better characterizing the component unavailability due to maintenance, by recording also the reactor status during the repair (as an additional failure attribute), the average test frequency adopted by the operator and the observed average duration of the component unavailability, if this is so, during the test with plant in operating conditions (as two additional component operating attributes).

  It could also be convenient to collect data on the preventive maintenance (by the introduction of a "maintenance report form"), to answer the requests of some Utilities interested in maintenance policy analysis and optimization. This would be less important for PSA, due to the fact that preventive maintenance causing unavailability is mostly performed in shut-down conditions.

2.3 Data retrieval and processing

2.3.1 The CEDB, as all large computerized component event data banks, allows, through its enquiry procedure:

- the selection of a set of specified components and of the relative set of failures of a specified type;
- the statistical treatment of this set of failures (and of the associated repairs);
- the retrieval and the display of all information items stored in the bank, such as the characteristics of a component and the report of each failure it suffered.

2.3.2 TABLES IV, V and VI give lists of the component, operation, failure attributes, with the indication for each attribute if it is questionable or not, i.e. if a selection can be made or not on the basis of this attribute. Through stepwise refinements made by the "SELECTION" command, the user can select, for example, a set of valves, installed in PWR's, pertaining to the "condensate and feedwater system", globe-valve type, of a specified diameter range, operating at a certain temperature range; of all the failures which occurred to this set of valves, the user can now select those corresponding to the failure mode in operation-degree of suddenness "incipient". To this last set of failures selected, the user can apply all the statistical application programmes inserted in the on-line enquiry procedure.

TABLE IV : Attributes of the entity COMPONENT

ATTRIBUTE	EXTENDED ATTR	QUEST	TAB	CLASS	TYPE	INDEX
CLA	CODES/STANDARD/SAFETY CLASS	Y	Y		A	0
COMP-CODE	REACTOR COMPONENT CODE	Y			A	0
CON-DA	DATE OF CONTRUCTION	Y			D	0
CYCLES	ANNUAL PROGR. NUMBER OF CYCLES	N			N	0
EN-CH-COD	ENGIN. CHARACTERISTIC	Y		ENG	A	0
EN-CH-NU	ENGIN. NUMERICAL PARAMETER	Y		ENG	F	2
FIRST-CYCLES	FIRST PROGR. CYCLES	N			N	0
FIRST-HOURS	FIRST PROGR. NUM. OPER. HOURS	N			N	0
HOURS	ANNUAL PROGR. OPERATING HOURS	N			N	0
IA-COD	IAEA REACTOR CODE	Y			A	0
ID	IDENTIFIER	Y		ENG	A	0
MAN	MANUFACTURER CODE	Y	Y		A	0
MODEL	MANUFACTURER MODEL	Y			A	0
NA	NATION	Y	Y		A	0
REA-PO	REACTOR POWER RANGE	Y	Y		A	0
REA-TY	REACTOR TYPE	Y			A	0
SC-DA	DATE OF SCRAPPING	Y			D	0
SER-NUM	MANUFACTURER SERIAL NUMBER	Y			A	0
SE-DA	IN SERVICE DATE	Y			D	0
SY	ERDS SYSTEM	Y	Y		A	0
YEAR-CYCLES	YEAR OF CYCLE PROGR.	N			N	0
YEAR-HOURS	YEAR OF OPERATING HOURS	N			N	0
Y-STAR	STARTING YEAR	Y			N	0

TABLE V : Attributes of the entity OPERATION

ATTRIBUTE	EXTENDED ATTR	QUEST	TAB	CLASS	TYPE	INDEX
AL	ALTITUDE	Y	Y		A	0
CLI	CLIMATE	Y	Y		A	0
CORRO	CORROSION	Y	Y		A	0
H	HUMIDITY	Y	Y		A	0
IND	TYPE OF INDUSTRY	Y	Y		A	0
INS	INSTALLATION	Y	Y		A	0
MAI	MAINTENANCE TYPE	Y	Y		A	0
MODE	MODE OF OPERATION	Y	Y		A	0
OP-CH-COD	OPER. CHARACTERISTIC	Y		ENG	A	0
OP-CH-NU	OPER. NUMERICAL PARAMETER	Y		ENG	F	2
OP-DA	OPERATION DATE	Y			D	0
PR	PRESSURE	Y	Y		A	0
RA	RADIATION	Y	Y		A	0
TE	TEMPERATURE	Y	Y		A	0
V	VIBRATION	Y	Y		A	0

TABLE VI : Attributes of the entity FAILURE

ATTRIBUTE	EXTENDED ATTR	QUEST	TAB	CLASS	TYPE	INDEX
AC-AD	ADMINISTRATIVE ACTION TAKEN	Y	Y		A	0
AC-CORRE	CORRECTIVE ACTION TAKEN	Y	Y		A	0
CA	CAUSE	Y	Y		A	0
CYCLES	CYCLES AT FAILURE OCCURRENCE	N			N	0
DES	DESCRIPTOR	Y	Y		A	0
DET	DETECTION	Y	Y		A	0
EF-OT	EFFECT ON OTHER SYSTEMS	Y	Y		A	0
EF-REA	EFFECT ON REACTOR OPERATION	Y	Y		A	0
EF-SY	EFFECT ON SYSTEM	Y	Y		A	0
FA-DA	FAILURE DATE	Y			D	0
HOURS	HOURS AT FAILURE OCCURRENCE	N			N	0
MODE-DEM	MODE ON DEMAND	Y	Y		A	0
MODE-OP-DEG	MODE OPER. DEGREE	Y	Y		A	0
MODE-OP-SU	MODE OPER. SUD.	Y	Y		A	0
PA	PART-FAILED	Y	Y		A	0
REA-STAT	REACTOR STATUS	Y	Y		A	0
REL	RELATED FAILURES	Y			A	0
REMARKS	REMARKS	N			A	0
REPAIR-TIME	REPAIR TIME	Y			N	0
RES	START-UP RESTRICTIONS	Y	Y		A	0
U-DA	DATE OF UNAVAILABILITY	Y			D	0
UNAV-TIME	UNAVAILABILITY TIME	Y			N	0

2.3.3 CEDB data processing methods and computer programmes

Through the CEDB interactive enquiry procedure, the analyst can estimate reliability parameters for a specific component category in which he is interested. The on-line statistical processing includes at present:

a) Point and interval estimation (for complete and censored samples) of:

a.1) constant reliability parameters (time-independent failure rate in operation, constant unavailability on demand, time-independent repair rate), by /6,7,8,9/:

- Bayesian parametric approach (with priors: beta, uniform, log-uniform, log-normal, histogram);
- classical approach (maximum likelihood, confidence interval).

a.2) non-constant reliability parameters (time-dependent failure rate in operation, time-dependent repair rate) by the Bayesian non parametric approach (with prior identified by a sample of times to failure or by a failure-time analytical distribution) /10,11/.

b) Test of hypothesis on the law of failure and repair time distribution:

- exponential (for complete and censored samples);
- Weibull, log-normal and gamma distribution, increasing failure rate, decreasing failure rate (only for complete samples).

Effective graphical tools can give on-line the representation of an observed time-dependent failure rate; of the prior and the posterior distributions (Bayesian parametric approach); of the cumulative failure distribution function F of the observed, the prior and the posterior sample (Bayesian non-parametric approach), etc.

In refining a selected sample of failures for a statistical analysis, the analyst can retrieve and review each event to identify and delete from the sample those failures which appear not to be independent.

As an example of off-line statistical processing tool, we mention the recently implemented interface which allows the application of the SAS statistical package, available at the JRC /12/, to the full set of data related to a sample of components.

3. CEDB use and data analysis

3.1 On-line inquiry for inference of reliability parameters

Some examples of data treatment by using a Bayesian parametric approach (with the necessary assumption of constant failure rate or constant unavailability on demand) and the Bayesian non-parametric approach (in the case that the test of exponentiality of the failure time distribution is rejected and, as a consequence, failure rate cannot be assumed constant) are given in /13/. The CEDB, with the full package of data processing programmes inserted in its on-line enquiry procedure, is an effective and versatile tool for the safety analyst. Nevertheless, on the basis of the experience gained in data treatment, we have verified that the number of reported failures, characterized by a sudden and complete loss of the function, is often very small, even when the size of the sample observed is remarkable (TABLE VII). This is due, of course, to the high reliability features of most of the categories of components monitored. It is to be noted that, when the number of failures is too small, the implemented test of exponentiality cannot be made and, as a consequence, the Bayesian parametric approach (with the assumption of constant reliability parameter) is the obligatory way to exploit the stored information.

The application of the test of exponential distribution (when it can be made) shows that a constant parameter can be assumed in a few cases only /14/; nevertheless, we noted that when the sample refers to complete failures of components performing the same function in similar plants (i.e. components having similar design-related and application-related attributes), the test of exponentiality is accepted with a certain frequency.

3.2 CEDB - Reliability Data Book

Generic reliability data for pre-defined classes of components, derived by a statistical processing of the contents of the CEDB, are being collected in a Reliability Data Book (RDB) /14/. This book is produced as an organized output of the bank and will be periodically updated. It gives, for each item considered:

TABLE VII Examples of CEDB on-line data treatment for the pumps of the Auxiliary Feedwater System of PWR's (constant failure rate)

Component/ /function	System/ /failure mode	Sample observed: No comp.s / Operating time/demands / No failures/repairs	Prior (fail./h or fail./d) Median	Prior 95th percentile	Source	Posterior (fail./h or fail./d) Median	Posterior 95th percentile
Electric motors/ /drivers	B10/ /fails to start	32comp.s 9527 d 0 failures	$2.47\ 10^{-5}$	$6.85\ 10^{-5}$	1	$2.25\ 10^{-5}$	$5.89\ 10^{-5}$
Steam turbines/ /drivers	B10/ /fails to start	10 comp.s 966 d 0 failures	$1.6\ 10^{-3}$	$1.8\ 10^{-2}$	2	$4.78\ 10^{-4}$	$1.95\ 10^{-3}$
Steam turbines/ /drivers	B10/ /fails to run once started	10 comp.s 10475 h 1 failure	$4.7\ 10^{-7}$	$4.3\ 10^{-5}$	2	$2.56\ 10^{-5}$	$1.68\ 10^{-4}$
Emergency feed pumps (without drivers)	B10/ /fails to start	31 comp.s 5079 d 3 failures	$2\ 10^{-4}$	$1.9\ 10^{-3}$	2	$4.53\ 10^{-4}$	$1.06\ 10^{-3}$
Emergency feed pumps (without drivers)	B10/ /fails to run once started	31 comp.s 26145 h 8 failures	$4.7\ 10^{-7}$	$4.3\ 10^{-5}$	2	$2.62\ 10^{-4}$	$4.59\ 10^{-4}$
Emergency feed pumps (without drivers)	B10/ /all fail. modes	18 comp.s 509 h (tot. rep. time) MTTR = 12 h STD = 13.6 h 42 repairs (failures detected with reactor in operation)				min rep. time 1 h max rep. time 50 h	

NOTES

B10 Auxiliary feed water system

Source 1: IEEE ST 500, 1977 pag. 204-205: prior assumed as a lognormal function, "maximum" and "recommended" values are used as the 95th and the 50% percentiles, respectively, of the distribution.

Source 2: OCONEE PRA, NSAG 60, June 1984, Vol. 4 pag. B.13: prior assumed as a lognormal function.

- supporting qualitative and quantitative information on the item's technical and operational characteristics and on the sample monitored (sample size, total time of observation/number of demands, total number of failures, etc.);
- for each of the relevant operation modes/failure modes, the mean time between failure, the mean (number of) demands between failure, the mean repair time and the relative standard deviations. In the case that a constant failure rate model can be assumed (test of exponentiality accepted), the following additional information is provided: an estimate of the failure rate in operation or on demand, its standard deviation and the 90% symmetrical confidence interval.

Be merely consulting this book, the analyst, without accessing the bank, can quickly obtain generic reliability data for the item he is interested in, evaluate their credibility on the basis of the sample size from which they have been derived, use them as priors for Bayesian updating on the basis of plant-specific data, etc.

3.3 Analysis of linked multiple failure events (MFE's)

A study on linked MFE's (such as the CCF's), based on the establishment of a comprehensive classification scheme of these event-types and on a search method performed through an extensive screening of the data base, was carried out between 1983 and 1985 /15/. The proposed general classification scheme is based on the combination of these four possible linking factors: failure-cause, failure-mode, temporal distribution of the linked failure-events, failed component(s) involved. Through some practical application of the above-mentioned search method, the authors have shown how component failure databases can effectively be exploited for identifying potential hazardous dependent failures and gain knowledge for improving design and reducing operation errors and malpractices. This way of investigation appears to be very promising and will be continued.

3.4 Application of multivariate analysis techniques

3.4.1 The results of a first application of covariate analysis to a data set of 543 gate valves, with the associated 156 failures, are reported in /16/. All the failures are considered in the study, irrespective of their failure mode (145 failures are classified as "incipient", 11 as "sudden"). According to the results obtained, the failure rate related to this sample of valves, which had (nearly all) incipient failures (i.e. failures with degraded performance of the function) is a constant for the valves which operated in a given engineering system inside a given plant unit; it changes its value from system to system (inside the same plant unit) and from unit to unit. The plant unit dependency could be due more to the presumable differences existing between the criteria followed by the various data collectors for recording partial failures (differences between the criteria adopted by the data collection schemes, subjectivity in their interpretation) than to differences in component performance. Nevertheless, ascertaining the importance of the factor "manufacturer" and "maintenance policy" in determining the component real performance remains the most crucial problem. These results, demonstrative of the effectiveness of the methodologies used, ask for further investigation to verify all their significance from the engineering point of view.

3.4.2 Multiple correspondence analysis is being performed on a data set of 417 pumps, taking into account only their "sudden" (i.e. complete) failures. This technique is applied for the first time for CEDB data analysis and conclusive results are not yet available. We note that some interesting discriminations are obtained in the graphical representations of failure data coming from plants of different type (PWR, BWR) or from plants pertaining to different countries. This line of investigation appears to be very promising.

4. Conclusions

Experience has shown that the CEDB structure is well suitable to operate as a centralized bank (i.e. to receive data from national banks through a computer-aided transcoding process) or to a direct collection of data, in the CEDB format, in the NPP's. Due to the package of data treatment programmes inserted in its enquiry procedure, it is an effective tool for the user to infer good reliability parameters. The line of investigation based on the application of multivariate analysis techniques appears to be very promising for failure trend analysis, failure root cause analysis, failure dependency on the component-attributes, etc.

A progressive increase in the future of the CEDB data base, to be guaranteed by the joint effort of the European data collectors, will enable the user to more and more exploit its capabilities.

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EVALUATION OF RELIABILITY DATA SOURCES IN CHINA

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Abstract

The paper first reviews the development of probabilistic safety assessment in the last few years in China.We have done the following work:

(1) Training and improving the personnel's capabilities of performing PSA.

(2) Organizing activities forputting emphasis on the solving PSA for Qinshan and Guangdoing NPPs.

(3) Working out the project for setting up the Reliability Data Bank in Research Institute of Nuclear Power Operation.

Secondly, the paper introduces a plan on the development of PSA in Research Institute of Nuclear Power Operation in the near future. The plan includes:

(1) Investigating and translating the information from abroad so as to train personnel's capabilities continuously.

(2) Importing part of the applied analysis procedure and putting into computer M-240D.

(3) Sending some people to Qinshan NPP to attend the commissioning there for collecting operating data.

(4) Setting up the Reliability Data Bank.

(5) Developing the analysis model and multifunction computer code and improving PSA methodology.

The paper finally expresses our hope to co-operation and exchange experiences with other countries in the field of PSA technology.

A. Review

Since TMI incident of NPP in 1979 in U.S.A,nuclear safety measures have been widely enhanced in the countries all over the world.The organizations which are specially engaged in nuclear safety research have been set up one after another and a lot of effective work has been done.

As is widelyknown,China is a developing country.Nuclear power development in China has just started.A 300 megawatt -electric (MWe)Nuclear Power Plant at Qinshan in Zhejiang province and two 900MWe PWR units at Daya Bay in Guangdong province are being constructed.In order to meet the requirements of national economic development our government also contemplates to construct two 600-MWe PWR units at the same site in Qinshan after the first phase of construction.This project will be completed and put into commercial operation soon after 1993.China has devoted much attention to the nuclear safety.In 1986,The Ministry of Nuclear Industry decided to found a Research Institute of Nuclear Power Operation at Wuhan city in Hubei province as a direct technical supporting unit of NPPs.The Institute sets up 6 divisions as follows:

The division of operation management of NPPs;

The center of operating personnel training of NPPs;

The center of quanlity assurance of NPPs;

The center of in-service inspection;

The division of NPP's components research and design;

The division of information material.

There are 300 members of nuclear engineering experts covering wide fields of expertise in this Institute. About 150 members are specialists possessing nuclear engineering experiences over ten years.The first engineering stage of the Institute's elementary construction was completed in 1985. Now,It is proceeding to the second engineering stage.After the second phase is completed,the Institute will be a technical supporting unit with wide technical service.

Because the city of Wuhan is located in the center of our country,it has developed water,land and air transportation. This is advantageous to makeing the Institute become a nationwide center of technical service and information communication.

The probabilistic safety analysis (PSA) is now increasingly applied to design,licensing and operation of NPPs.It is an important part of NPP's operation research.In order to elaborate using PSA for finding out where the weak points lie and searching for their possible improvement in design and operation as early as possible, from 1983 Ministry of Nuclear Industry set about organizing some people to translate and publish U.S.NRC publications NUREG-2300,and through these activities to train personnel's capabilities.National Nuclear Safety Administration and Ministry of Nuclear Industry organized together two expert teams to solve PSA issues for Qinshan NPP and Guangdong NPP.These activities have been supported by international experts.The agreement between MNI and GRS, Federal Republic of Germany,for PSA of Qinshan Plant will be completed in two years.Now PSA level one is near completion, and the PSA of Guangdong Plant has participated in the IAEA's inter-regional project of PSA,having got the great concern from the division of Nuclear Safety Reliability and Risk Assessment,IAEA.

Nowadays,In our country Some units like Ministry of Electronic Industry and Ministry of Aviation and Spaceflight Industry have established the research center of complete system reliability.All important universities also have estabilished PSA research courses.All these will give Nuclear Industry an advantageous technical support.

MNI Authority has made a project which plans to Set up a big reliability data bank of NPPs at Research Institute of Nuclear Power Operation in Wuhan in order to develop PSA more effectively,and it will be gradually expanded to be a center of nationwide information commumication for Nuclear Power.

B. <u>Plans in the near future</u>

The division of NPP's operations management which is a subordinate unit of RINPO (Research Institute of Nuclear Power Operation) undertakes the PSA research tasks.Now,there are 37 members in this division.21 members are engineers with experiences over ten years in the fields of reactor physics, thermalhydraulics,loops, electrotechnics,control,radiation protection,computer software and hardware etc.Among them there are ten senior engineers.Now,we are putting apropriate persons into the field of PSA research and also the team will be expanded increacingly.

For creating reliability data bank,the division organized people to make overall investigations in the hardware area and get prepared to construct.For example,To select computer Model and interface component for data bank and communication network;and large engineering simulator for data analysis.A consultant meeting of internal experts was held in 1987.Chinese government will allocate funds for purchasing computer and manufacturing components.The data bank is required to start collecting and storing operating experience data before the first chinese Nuclear Power Plant is put into operation.

In the software area, The main task is training personnel capability for data collecting, analysis and retrieval in next two years. The scope of research will be expanded and deepened step by step.

The plan is as follows:

1. Training personnel capability

By means of holding seminars inviting specialists to come to give lectures, investigating and translating information materials from abroad, the division makes researchers recognize and know well the status of NPP system reliability research, its developing trend, research direction and method.

Two terms of seminar were organized in 1987. The lecturers were Professors of Jiao Tong University. Meanwhile, according to the investigation, a part of information materials about National Centre of systems reliability in England were translated into chinese.

2. Creating co-operative relationship

A long term co-operative relationship between the division and Jiao-Tong University, Shanghai has been established. The University send teachers to help the division to introduce computer programs about fault tree analysis and put it into computer M-240D which is located in the computer center nearby.

First of all we completed 3 programs which are program PREP for qualitative evaluations, program FTAP for quantitative evaluations and program KITT for maintainable system reliability analysis. Then, both sides will join together to prepare a sythetic fault tree analysis program which will possess functions of "Sample" and "Importance" and also include functions of the 3 programs above.

3. going to in-situ

The division will organize its members to go to in-situ and to attend commissioning of Qinshan NPP. It will make its members grasp the regularity of collecting engineering operation data and solve to found specific field data collection compaigns. Based on this, construction of communication network will be completed.

4. creating reliability data bank

The plans to establish engineering data bank will cover about 30 systems, 4000 to 5000 components. It must start collecting, storing and retrieving data before the first chinese NPP is put into operation.

5. setting up and developing analysis model.

On the basis of the PRA level one we trend to develop the PRA level 2 or 3,

6. setting up the specialist network.

to found the specialist network for evalvation of reliability in PWR design,operation and maintenance.

7. developing new computer programs.

to develop multifunction computer analysis program and new calculation method programs,the latter including those due to uncertainty effect assessment,human reliability factors and common cause failure effect.

C. Conclusion

Before the first Chinese Nuclear Power Plant is put into operation,there is a precious term.During this time we must learn and apply as many experiences (both positive and negative) as possible from advanced countries to avoid unnecessary repetition and never follow roundabout way.So,We hope to establish the co-operation relationship and exchange experiences with other countries all over the world in the PSA technical research field.

The openions are as follows:

1) We sincerely hope foreign experts would come to China,to give lectures,exchange experiences and train personnel capability.We also hope IAEA would give us more help in this field.

2) We would like to take part in international co-operation to tackle key problems in PSA methodologies improvement and new calculation program development.

3) We welcome foreign experts to China to consult and direct us in creating reliability data bank including hard-ware and software.

4) We would suggest and support a Co-ordinated Agency's Research Programme on reliability data bank organized by IAEA,and will participate actively.

DATBASE: A COMPUTER CODE FOR THE HANDLING OF RELIABILITY DATA SOURCES IN PSA STUDIES

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Abstract

DATBASE is a computer code to obtain component unavailability from generic or specific failure rate data sources and to generate the proper numerical input data to be used in the following fault-tree analysis computer codes: FTAP (Fault Tree Analysis Program), SETS (Set Equation Transformation System) and TREE - MASTER. The code includes an option to modify the unavailability of basic events in the fault tree of a system which makes it a complementary tool in sensitivity studies. Also it can be applied to the screening of events in common cause failure analysis. The code was developed for an IBM-PC or compatible microcomputer using dBASE III plus.

I. INTRODUCTION

One of the objectives of the PSA group in the Comision Nacional de Seguridad Nuclear y Salvaguardias (CNSNS) - the Mexican Nuclear Regulatory Body - is the development, validation and application of risk analysis methodologies and their corresponding computer codes. At present these technics are used for the Laguna Verde Nuclear Power Station AC Blackout Assessment.

Within the scope of this assessment - and due to the fact that a key element in the estimation of a system s reliability is the availability of appropriate failure rates for the components which make up the system [1] - it was recognized the need to develop a computer program to handle failure data and to compute component unavailability for application in quantitative fault tree analysis.

The "DATBASE" computer code is the result of fulfilling the above need. The main features of the code are the following:

1. DATBASE includes a failure rate data base that incorporates updated information from various sources.

2. The code uses a set of equations that describes the stochastic failure behavior of the various components in a time dependent model.

3. A program is included that uses the above equations to calculate the unavailability for specific components or events in terms of the failure rate in the data base and the particular conditions assumed for the event in question.

4. It prepares the proper numerical input data for the FTAP [2], SETS [3] and TREE-MASTER [4] fault tree analysis computer codes.

5. Allows the user the screening of events with the same predetermined location in the plant or the screening of those events with unavailability greater or equal to some selected value.

6. DATBASE is programed in dBASE III Plus [5] and it is compiled for its use in an IBM-PC or compatible microcomputer.

In Section II of this work we present the basic equations used in the program and their particular application to different types of components. Section III describes the failure rate data base used in the code, whereas Section IV presents the DATBASE program structure. Finally in Section V we suggest further applications for the DATBASE computer code.

II. BASIC EQUATIONS

We shall assume that a component can be in one and only one of two possible states: available or unavailable – which we will denote by 0 and 1 respectively –, and that such a component can go randomly from one state to the other. Then the equations that describe this random behavior are

$$\frac{dP_0(t)}{dt} + \lambda(t)P_0(t) = \mu(t)P_1(t) \tag{1}$$

and

$$\frac{dP_1(t)}{dt} = \mu(t)P_1(t) = \lambda(t)P_0(t) \tag{2}$$

where

$P_0(t)$ = Probability that the component is in state 0 at time t,

$P_1(t)$ = Probability that the component is in **state** 1 at time t,

$\lambda(t)$ = Probability per unit time that the component goes from state 0 to state 1 at the time instant t,

$\mu(t)$ = Probability per unit time that the component goes from state 1 to state 0 at the time instant t.

Equations (1) and (2) can be solved for $P_0(t)$ and $P_1(t)$, namely

$$P_0(t) = \frac{\lambda}{\lambda+\mu} e^{-(\lambda+\mu)t} + \frac{\mu}{\lambda+\mu} \tag{3}$$

and

$$P_1(t) = \frac{\lambda}{\lambda+\mu} \left[1-e^{-(\lambda+\mu)t} \right] \tag{4}$$

where we have assumed $\lambda(t)$ and $\mu(t)$ constants and $P_0(0)=1$.

For periodically tested standby components, which constitute most of the Emergency Core Cooling Systems (ECCS), there are at least four kinds of contributions to the component unavailability, namely [6]: hardware failures, unavailability due to test, unavailability due to unscheduled repair and unavailability due to scheduled maintenance. In PSA applications it's common practice to include these four contributions as four different events in the system's fault tree [6]. The unavailabilities due to scheduled test (Q_{TEST}) and maintenance (Q_{MAIN}) are not random and are given by [1]

$$Q_{TEST} = \frac{\tau}{T_s} \qquad \text{and} \qquad Q_{MAIN} = f_m T_m$$

where τ is the average test duration, T_s is the time interval between tests, f_m is the scheduled maintenance frequency, and T_m the time of the scheduled maintenance action.

The average unavailability due to hardware failures for a periodically tested standby components (Q_{AV}) is derived from equation (4) and is given by

$$Q_{AV}(T_s) = 1 - \frac{1}{\lambda_s T_s}\left[1 - e^{-\lambda_s T_s}\right] \qquad (5)$$

with λ_s denoting the standby failure rate.

For untested standby components the average unavailability can be obtained from expression (5) by replacing T_s by T_P where the latter stands for the fault exposure time [1].

The average steady state unavailability (Q_{CONT}) for continuously monitored standby components is derived from (4) as

$$Q_{CONT} = \frac{\lambda_s T_R}{1+\lambda_s T_S} \qquad (6)$$

where $T_R = \frac{1}{\mu}$ is the mean time to repair

In the case of nonrepairable online components the unavailability, according to equation (4), is

$$Q_{OP}(T_M) = 1 - e^{-\lambda_0 T_M} \qquad (7)$$

with λ_0 and T_M the operating failure rate and the mission time respectively.

For online repairable components the average steady state unavailability (Q_{OPR}) is obtained from equation (6) by replacing λ_s by λ_0.

III. FAILURE RATE DATA BASE

DATBASE can obtain component unavailability either from specific or generic data bases. However, presently we don t have readly available specific data and therefore we have installed generic or baseline data in the code. Although various data sources were used for the development of the DATBASE failure rate data base, the NUREG/CR-2815 data base [1] was selected as baseline source. When some data was not available in this reference, it was selected from specific PSA studies [7-8], or from references [9], [6] or [10] respectively. Our data base contains the information distributed in nine arrays: CODE, COMPONENT, FAILURE MODE, MIN, MEAN, MAX, DISTRIBUTION, REFERENCE and REMARKS.

IV. DATBASE PROGRAM STRUCTURE

The program structure and the interaction of files are schematically described in figs. 1 and 2 respectively. DATBASE requires a data base (file DATBASE.dbf) which includes the information contained in the fields: CODE, COMPONENT, FAIL-MODE, MIN, MEAN, MAX, DIST, REFERENCE and REMARKS. Also an alphabetic index file (DATBASE.ntx) is generated from the field CODE of DATBASE.dbf. The program also uses a master file (BASEOUT.dbf) which contains the arrays: CODE, LABEL, TYPE, TIME, LOCATION, COMPONENT, FAIL-MODE, MEAN and UNAVAILABILITY.

The program creates another dBASE file (SYSTEMDT.dbf) copying the structure given in BASEOUT.dbf . This new file contains the whole information about the basic events of the system's fault tree to be analyzed, i.e. the event code (field CODE), the label given for the event in the system's fault tree (field LABEL), the affected component and its location (fields COMPONENT, FAIL-MODE and LOCATION), the corresponding failure rate (field MEAN) and the information needed to calculate the unavailability (field Q) from the component type (field TYPE) and the required mission time or time between tests (field TIME).

The name of the system s fault tree to be analyzed, together with the information corresponding to the fields: LABEL, CODE, TIME, TYPE and LOCATION are supplied in an interactive way by the user. The remaining information is obtained from the data base and the program own calculations.

Following the input of all the component data, the program proceeds to calculate the unavailability for each event. DATBASE uses the appropriate equation according to the type of component, i.e. if it is a periodically tested standby component, an untested standby component, a continuously monitored standby component, a nonrepairable online component or a repairable online component. Also it is possible to handle the failure pprobability on demand as basic information instead of the failure rate.

The suitable failure rate is obtained from the data base for each event. The component name and its failure mode are also obtained from the data base.

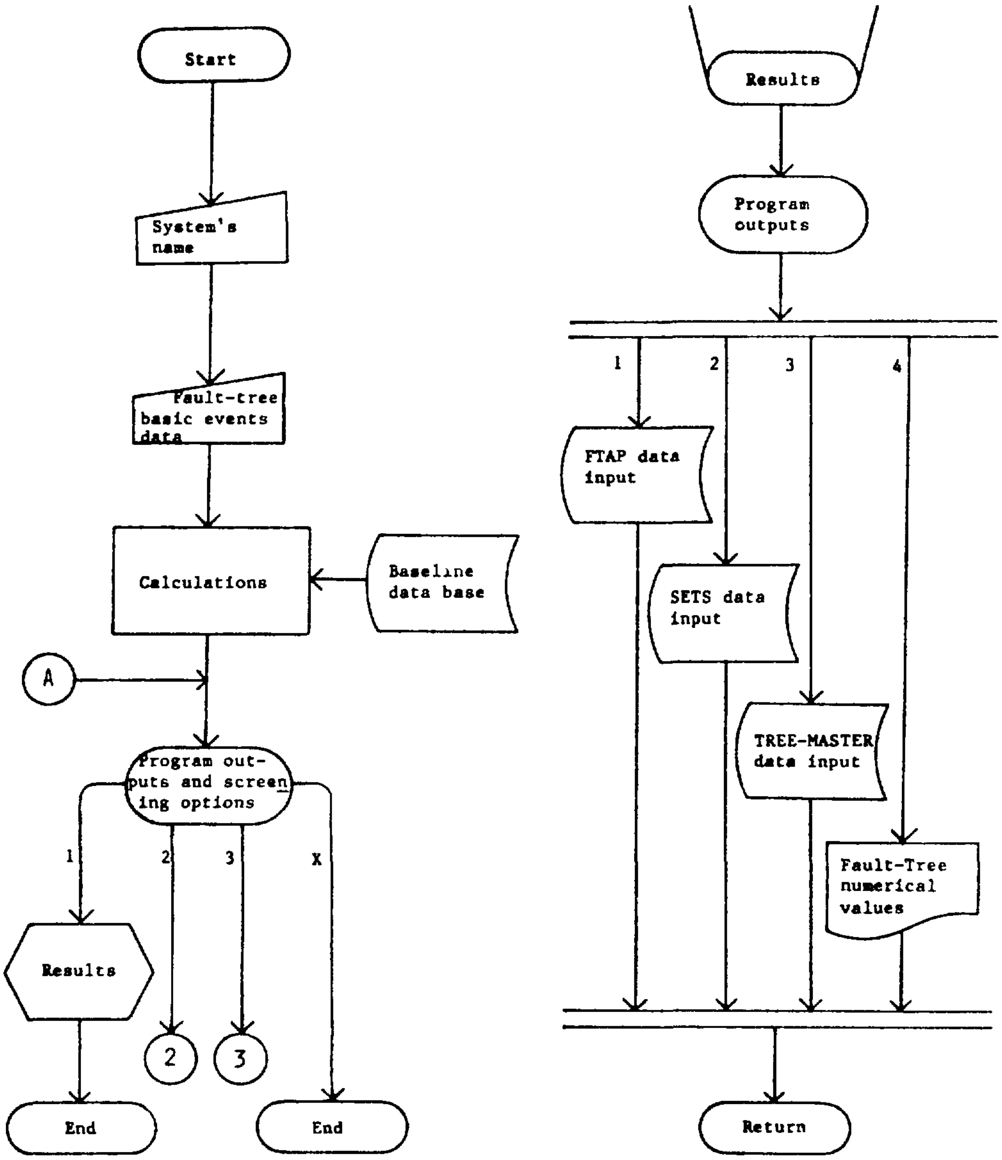

FIG.1. DATBASE program structure

DATBASE gives four different format outputs for the computed results. The first output option prints all the numerical values generated by the program for the basic events of the system's fault tree. The remaining three output options correspond to the proper data input files for the FTAP, SETS and TREE-MASTER fault tree analysis computer codes.

V. DATBASE FURTHER APPLICATIONS

DATBASE was designed to provide support in screening components or events both for sensitivity and common cause failure preliminary analysis. This type of analyses can be done by searching all the events that have some feature or characteristic in common. In particular, our present interest is in components within the same location at the nuclear power station, with unavailabilities greater or equal to some specified value and further with the same failure mode.

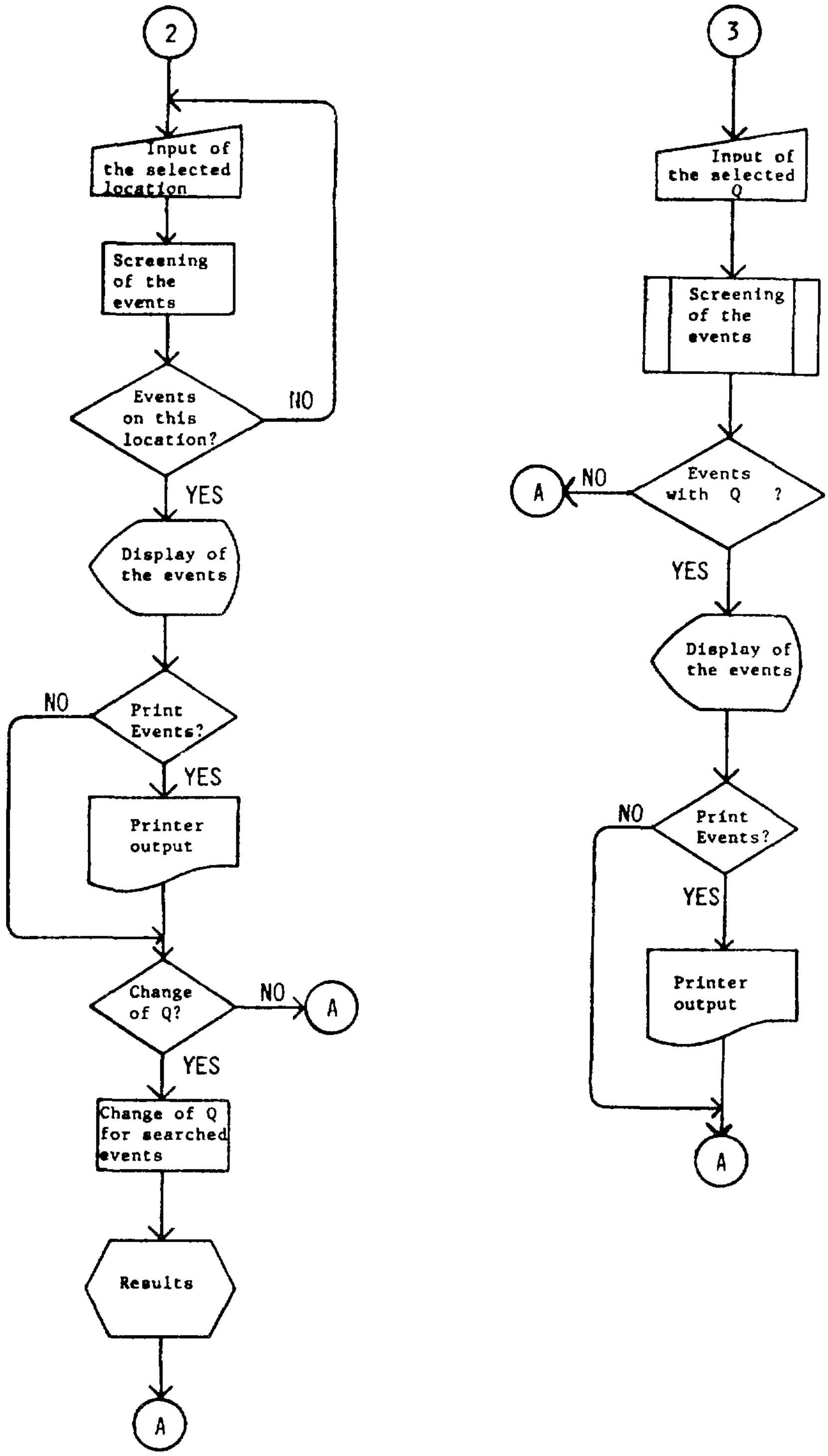

FIG.1. (cont.)

DATBASE also permits the automatic change from the actual unavailability into a predetermined one for those events within the same selected location. This allows to detect events with the same given location and to measure the impact of these events in the system's global unavailablity.

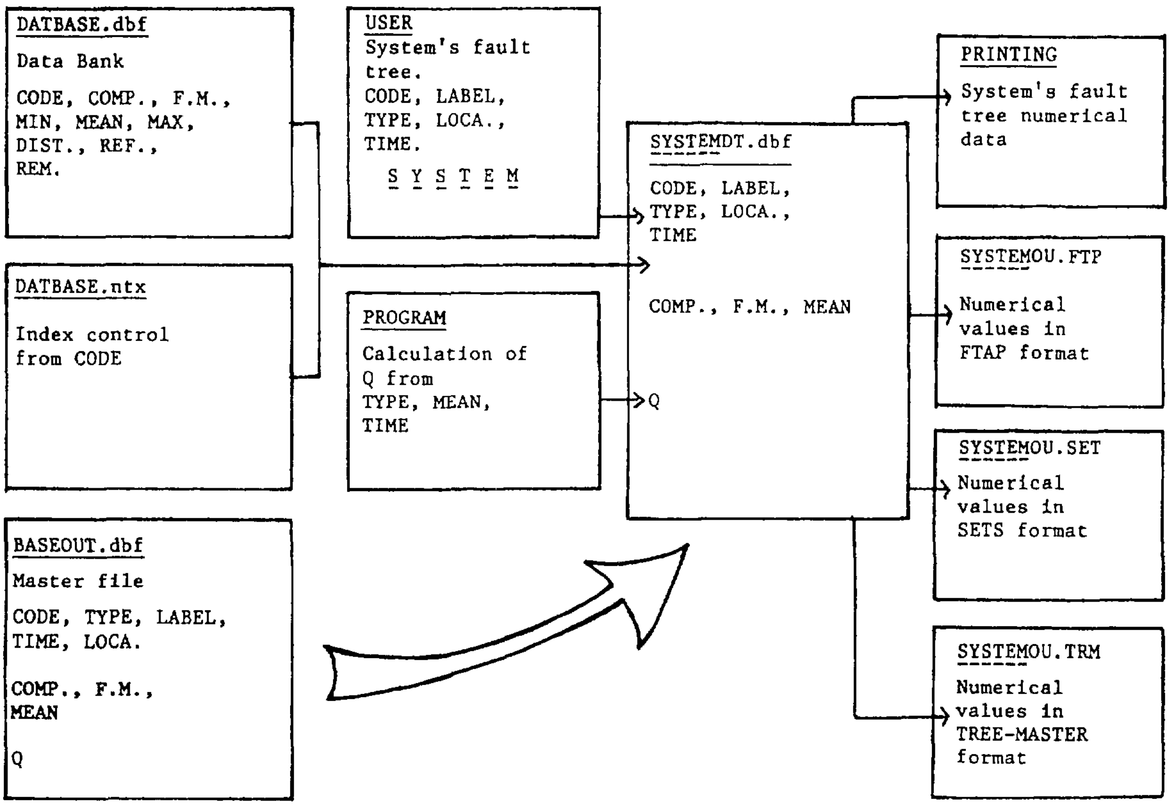

FIG.2 DATBASE interaction of files.

Since specific data is not available at present for the Laguna Verde Nuclear Power Plant, it was decided to incorporate generic data as a failure rate data base for the program. In addition, in the comming future the program will be adpated to update the generic failure rate data with site specific data. To achieve this objective the theorem of Bayes will be used to derive plant specific distributions for the failure rate of components out of the generic data bank distributions used as prior distributions. Then the resulting posterior distributions can be used as updated inputs in plant specific reliability or risk analysis [11].

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METHODS FOR COMBINING PLANT SPECIFIC OPERATING EXPERIENCE DATA WITH DATA FROM OTHER NUCLEAR POWER PLANTS IN PRA/PSA STUDIES

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Abstract

Based on the operational achievements measured by the availability performances, scram frequencies and other performance indicators the nuclear power plants can be regarded to form quite an inhomogeneous population also with respect to their performances. The studies have shown that i.e. the availability performances achieved by the nuclear power plants are correlated with several factors: the unit design and vintage, the degree of standardisation, the equipment suppliers, the competence and co-operation of all the nuclear energy parties in the site country, etc.

The heterogenity of nuclear power plants with respect to the plant type and main characteristics, on the one hand, and the differences between the operational achievements, on the other hand, result in estimation problems when we are evaluating for example failure rates and initial event frequencies for plant specific PRA/PSA studies. These estimation problems are usually solved by using empirical or subjective Bayesian methods.

The empirical Bayesian method utilizes the prior distribution based on the operating experiences from a population of some nuclear power plants and the plant specific evidence. In the subjective Bayesian method the operating experiences from other plants are weighted in some subjective way and after that combined with the plant specific evidence.

In this paper both the empirical and subjective Bayesian methods are described and compared. In the subjective Bayesian method the rules for selecting weights are discussed. The weighting criteria can be based on some characteristics of the nuclear power plants or on some factors contributed to the plant performances.

The use of subjective weights makes it possible to utilize the largest evidence, but it also has an impact on the uncertainty of the estimates. These uncertainties are also considered in the paper.

Finally the characteristics of the different methods are illustrated with some examples of practical nature.

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1. INTRODUCTION

In PRA/PSA studies, when we are estimating component failure rates and initiating event frequencies, the lack of relevant and sufficient data is a problem which is often encountered. Earlier, in particular, the problem must have been solved by generating purely subjective estimates or subjective probability distributions, based e.g. on "engineering judgements", for unknown parameters.

Nowadays there are several reliability data sources available. The power plant utilities, suppliers and the international organizations collect operating experiences from the nuclear and conventional power plants.

The Bayesian method is a statistical estimation method used frequently in PRA/PSA studies when there is no or very little plant specific operating experience available. If the quantity of interest is a failure rate λ the Bayesian theorem can be written

$$f(\lambda/E) = \frac{f(\lambda)L(E/\lambda)}{\int_0^\infty f(\lambda)L(E/\lambda)d\lambda}, \qquad (1)$$

where λ = failure rate

E = evidence data

$f(\lambda/E)$ = the probability density function of λ given evidence E (posterior distribution)

$f(\lambda)$ = the probability density function "prior" to having evidence E (prior distribution)

$L(E/\lambda)$ = the likelihood function or the probability distribution of the evidence E for a given value of λ.

A plant specific estimate for the component failure rate to be used in the PRA/PSA study is obtained from the posterior distribution which thus is dependent both on the "prior information" and the "evidence" according to the equation (1).

Although the Bayesian approach is well-established in reliability and PRA/PSA studies, it seems that no unique procedure exists for how to choose, interprete and treat the "prior information", the "evidence" and the plant specific data for obtaining of a proper estimate of a component failure rate, for example. It has been stated in the PRA procedures guide /1/: "The prior information reflects the analyst´s degree of belief about the parameter before the evidence; the posterior represents the degree of belief after incorporating the evidence." We have seen two ways to use the available plant specific data: one analyst does, another does not include the available plant specific data to the prior information with the other corresponding data obtained from the other plants. In addition and in particular, because some other data than/with the potential plant specific data are used for the plant specific estimation of the component failure rate, for example, some analysts are sceptically inclined towards the Bayesian approach and consider it too a non-objective statistical principle.

In this paper we shall not concentrate on any problems relating to the validities of the "objective" and "subjective" statistical estimation principles. Nevertheless, we shall discuss two methods for combination of component performance data from other nuclear power plants, including the plant under the PRA/PSA study. Both the methods considered are based on the Bayesian approach.

2. PROBLEM

The heterogeneity of the nuclear power plants with respect to the main characteristics of the plants, on the one hand, and also to the differences between the operational performances, on the other hand, results in an estimation problem: how relevant or useable are the available reliability data from different sources or the operating performance data from other plants when we are evaluating, for example, plant specific failure rates of components for PRA/PSA studies. Although the components and the plants are quite identical with respect to their engineering characteristics, there can be some plant operation and maintenance related factors, which correlate with the performances of the plants and their components. (Of course, it can not be expected the performance data of different valves, for example, or of plants operated in different "environments" do not differ from each other statistically.)

In this paper we shall present two techniques how to apply the Bayesian theorem (1). In the first case a sample of the performance data from other nuclear power plants are used for the estimation of the prior distribution of the component failure rates while in the second case those data are used as the evidence data in the likelihood function, and the relevance of the used evidence data is evaluated.

3. METHODS

3.1 Empirical Bayesian method

3.1.1 Likelihood function

The most common forms of likelihood functions encountered in the Bayesian reliability estimation problems are the Poisson and binomial distributions. When the evidence data are given by the number of failures k over an operating time T, e.g. to estimate the failure rate, the likelihood function is given by a Poisson distribution

$$L(E/\lambda) = \frac{(\lambda T)^k}{k!} e^{-\lambda T} . \qquad (2)$$

3.1.2 Gamma prior distribution

In Bayesian method applications, the so-called "conjugate priors" have often been used as a distribution representing the prior information. Probably one reason for the choice of conjugate priors is that the posterior distribution can be obtained analytically with such priors. In particular, when the available empirical data base is minor, as it is in many cases of practice, any approximative discrete function, instead of a continuous prior distribution, is not feasible to be formed, but a continuous prior distribution is to be chosen. In this case there is no preferred prior distribution, and the choice of the prior distribution type is subjective and can be based on a computational convenience. Moreover, according to the PRA guide /1/, "since noninformative priors contain no generic information, it may be preferable to avoid their use when even minimal generic prior data are available." ("Noninformative" prior distributions are a class of priors that loosely minimize the relative importance of the prior, compared with the data, in generating a posterior estimate; the reliability data given by the reliability data banks, handbooks, etc., are usually regarded as "generic" data.)

We assume that we have obtained from several nuclear power plants the operating histories of N components deemed to correspond to that being analysed. The operating histories are given as the failure number k_i and operating time T_i of each component in the form:

$$E[1,2,\ldots,N] = \{(k_1, T_1), (k_2, T_2),\ldots,(k_N, T_N)\}. \tag{3}$$

The operating history of the specific plant being analysed is given respectively by $E[0] = \{(k_0, T_0)\}$.

The classical maximum likelihood estimator for the failure rate λ_i is $\lambda_i = k_i/T_i$, $i = 0, \ldots, N$. We choose the gamma prior distribution which is the natural conjugate distribution to the Poisson likelihood (2). Then, if the failure rate λ is assumed to have a gamma distribution for its prior,

$$g(\lambda;\alpha,\beta) = \frac{\beta^{\alpha}}{\Gamma(\alpha)} \lambda^{\alpha-1} e^{-\beta\lambda}, \quad \lambda>0, \tag{4}$$

and if the likelihood function is the Poisson distribution (2), the posterior distribution can be expressed in closed form also as a gamma distribution: $g(\lambda;\alpha + k, \beta + T)$. The positive shape parameter α can be interpreted as the prior number of failures in β prior total operating time.

The task is to estimate the parameters α and β of the gamma prior distribution (4) from the available sample E(1,2,...,N). There are several approaches to this estimation task. A "two-stage" Bayesian procedure is represented in the PRA procedures guide.

Two basic statistical estimation methods, among the others, can be used to estimate α and β: the method of matching moments and the marginal maximum likelihood method, which are described by Martz and Waller /2/, for example, and programmed in the Technical Research Centre of Finland /3/.

After the parameter estimation the posterior mean, median, variance and $100(1 - \gamma)\%$ symmetric probability interval are given by the formulas /1/

Posterior mean: $(\alpha + k_0)/(\beta + T_0)$

Posterior median: $\chi^2_{.50}(2\alpha + 2k_0)/(2\beta + 2T_0)$

Posterior variance: $(\alpha + k_0)/(\beta + T_0)^2$

Posterior $100(1 - \gamma)\%$ symmetric probability interval:

$$[\chi^2_{\gamma/2}(2\alpha + 2k_0)/(2\beta + 2T_0);\ \chi^2_{1-\gamma/2}(2\alpha + 2k_0)/(2\beta + 2T_0)]$$

(The respective characteristics of the gamma prior distribution (4) are obtained from the above formulas by setting $k_0 = 0$ and $T_0 = 0$.)

3.2 Subjective methods for combining data sources

The phrase "subjective method" is here used to emphasize that the estimates or the distributions obtained by using such a method are not totally based on statistical observations. The observations are often interpreted in various ways which are always effected by some analyst dependent factors. These factors originate from the experience which the analyst has gathered; they may be based on statistical or empirical observations but they cannot be formulated for example as likelihood functions or as other similar concepts.

As opposite to "subjective methods" it is possible to speak about "objective methods" which include the empirical Bayesian statistics and classical statistical techniques. The objectivity of these methods can, however, be questioned because of the enormous number of possible interpretations of the raw data material. The objectivity of the "objective methods" cannot be increased by forgetting these uncertainties.

3.2.1 A " mixture method"

In the PRA-procedures guide /1/ a method referred to as the "mixture method" is discussed. It involves fitting a suitable prior to each generic source and then combining the individual prior distributions $f_i(\lambda)$ by forming a mixture,

$$f(\lambda) = \sum_{i=1}^{N} \omega_i f_i(\lambda), \; 0 < \omega_i < 1, \; \sum_{i=1}^{N} \omega_i = 1, \qquad (5)$$

where N is the number of sources.

If the mixture (5) is used as a prior distribution, the corresponding posterior distribution will also be a mixture of the individual posterior distributions, namely

$$f(\lambda/E) = \sum_{i=1}^{N} \omega_i' f_i(\lambda/E), \qquad (6)$$

where the new (updated) weights are

$$\omega_i' = \frac{\omega_i \int_0^\infty f_i(\lambda) L(E/\lambda) d\lambda}{\sum_{i=1}^{N} \omega_i \int_0^\infty f_i(\lambda) L(E/\lambda) d\lambda}, \; i = 1,\ldots,N. \qquad (7)$$

According to the PRA-procedures guide /1/ several methods are suggested tor determining the weights ω_i; no method is, however, presented in the PRA-procedures guide /1/.

3.2.2 A subjective Bayesian method for combining evidences

Another method for combining several evidences has been proposed by Pulkkinen et al /4/. Basicly the same ideas have been applied be Mosleh and Siu in estimation of common cause failure probabilities /5/. Both approaches utilize the possibility to define a subjective probability distribution in the space of all relevant evidences.

Again, it is assumed that there are N independent data sources or evidences the i^{th} evidence being of the form E_i = (k_i failures in time T_i). Further, it is assumed that the relevance (or useability) of the i^{th} evidence to be used as evidence for the case being analysed can be interpreted in a probabilistic way:

$$P(E_i \text{ is relevant}) = \omega_i,\ 0 < \omega_i < 1,\ i = 1,\dots,N. \tag{8}$$

The evidence E_0 is from the specific case under analysis and it is perfectly relevant

$$P(E_0 \text{ is relevant}) = \omega_0 = 1. \tag{9}$$

The probability that the only relevant evidence is E_0 is

$$\varepsilon_1 = \prod_{i=1}^{N} (1-\omega_i), \tag{10}$$

And the probability that all evidences are relevant is

$$\varepsilon_{2^N} = \prod_{i=1}^{N} \omega_i. \tag{11}$$

Between these extreme cases there are numerous combinations of evidences for each of which the probabilities ε_j, $j = 2,\dots,2^{N-1}$, can easily be evaluated. If the evidences E_i, $i\varepsilon C_j$, where C_j is some subset of the set $\{1,2,\dots,N\}$, are relevant we have the evidence E_j^*,

$$E_j^*: \qquad k_j^* = k_0 + \sum_{i\varepsilon C_j} k_i\ ,\ T_j^* = T_0 + \sum_{i\varepsilon C_j} T_i. \tag{12}$$

The probability of having this evidence is

$$\varepsilon_j = \prod_{i\varepsilon C_j} \omega_i \cdot \prod_{i\not\varepsilon C_j} (1-\omega_i). \tag{13}$$

If the prior distribution for λ is $f(\lambda)$ then the posterior corresponding the evidence E_j^* is $f(\lambda/E_j^*)$ and we have this posterior with probability ε_j. The "final posterior" is the mixture of the posteriors $f(\lambda/E_j^*)$

$$f_{post}(\lambda) = \sum_{j=1}^{2^N} \varepsilon_j f(\lambda/E_j^*). \tag{14}$$

and the posterior mean and second moment are

$$E_{post}(\lambda) = \sum_{j=1}^{2^N} \varepsilon_j E(\lambda/E_j^*) \tag{15}$$

and

$$M^2_{post}(\lambda) = \sum_{j=1}^{2^N} \varepsilon_j M^2(\lambda/E_j^*), \tag{16}$$

where $E(\cdot)$ and $M^2(\cdot)$ are notations for the mean and the second moment, respectively. The posterior variance is expressed in terms of the mean and the second moment

$$D^2_{post}(\lambda) = M^2_{post}(\lambda) - [E_{post}(\lambda)]^2. \tag{17}$$

If we use the approximately noninformative gamma prior, i.e.

$$f(\lambda) \sim \lambda^{-1/2}, \tag{18}$$

we obtain the posteriors

$$f(\lambda/E_j^*) = \frac{(T_j^*)^{k_j^*+1/2}}{\Gamma(k_j^*+1/2)} \lambda^{k_j^*-1/2} e^{-\lambda T_j^*}. \tag{19}$$

The "final posterior" is obtained by using equation (14).

The most essential feature of the above method is the interpretation of the evidence weights ω_i as subjective probabilities. This makes it possible to combine the distinct evidences E_i, $i=1,\ldots,N$, to form the "extended evidences" E_j^*, $j=1,\ldots,2^N$. This interpretation also makes it possible to obtain the probabilities of the extended evidences.

The probabilistic interpretation of the weights ω_i is not the only possibility. However, if ω_is are subjective probabilities there is no difficulty of conceptual or technical nature to include the obtained posteriors to the whole of the PRA/PSA. When other interpreations (for instance fuzzy sets or membership functions) are preferred many difficulties are immediately encountered.

The first problem is connected with the combination of evidences and the determination of the weights of the extended evidences. Other problems include the difficulties to include and interprete nonprobabilistic concepts in the PRA/PSA which is based on the notion of probability.

The ω_is describe the probability of the evidence E_i being usable or relevant to be used as data e.g. for the failure rate estimation of the plant under analysis. As subjective probabilities ω_is can be, in principle, determined rather arbitrarily to describe the analysts degree of belief about the relevance of the evidence E_i.

The relevance of the i^{th} evidence can be decomposed to some factors. Assume that one is interested, for instance, to use the occurences of some initiating transients at some nuclear power plants as the background data for the estimation the initiating event frequency for ones own plant. The weights ω_i describe, in a sense, the similarity between the plants and own plant. This similarity is partly the similarity between the plant designs and partly the similarity between the plant operational characteristics or principles. The design similarities can further decomposed to similarities between the plant sizes, plant vintages and plant suppliers etc. The operational characteristics may be decomposed to factors depending on the utility using the plant, the infrastructure of the site country and the authority in the site country. The figure 1 describes the above hierarchy of similarities.

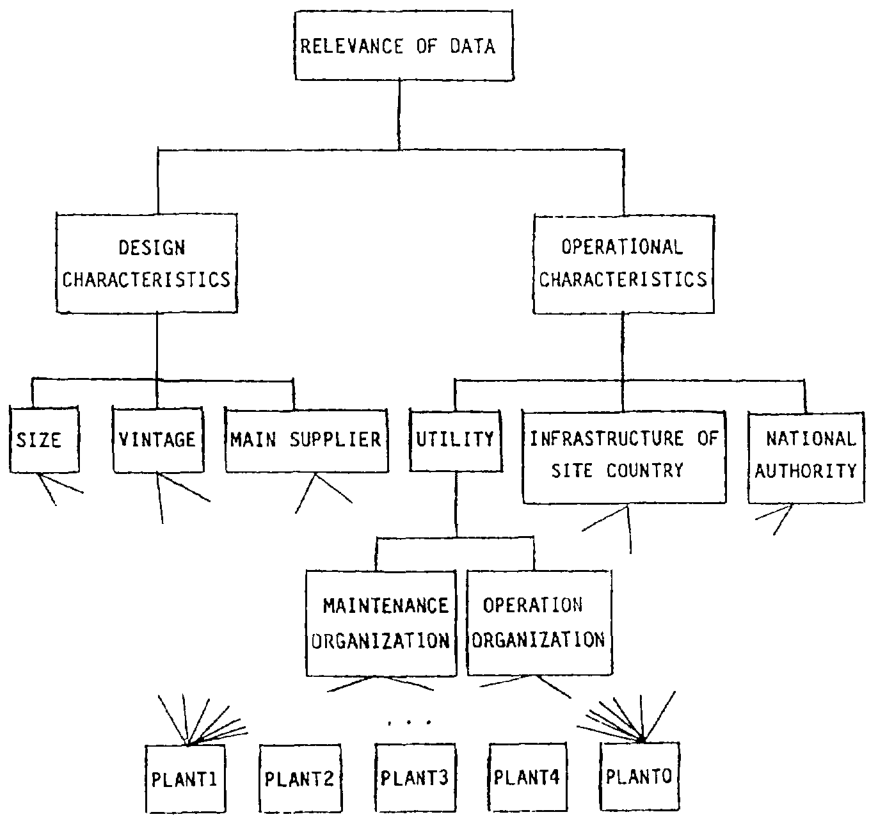

Figure 1. The hierarchy of similarities.

The index of similarity of the plants (i.e. ω_i) can be obtained, for example, by using the analytic hierarchy process (AHP) /6/. In AHP the factors in some hierarchy level are compared in pairwise way with respect to the goal at the next higher level. The comparisons are continued until the lowest level and then the global weights of the factors at the lowest level (i.e. the evidences E_i) with respect to the highest goal (i.e. the relevance of the evidence) are evaluated. These weights can be scaled to be used as probabilities ω_i.

The AHP is originally intented for comparison of alternative decisions in the situations there the decision criteria are conflictive. It can also be used to compare the probabilities of some events /7/. There are several computer codes for performing AHP-analyses, in this context we have used the code EC /8/.

Another possibility to obtain the weights is the use of the multiattribute utility decomposition /9/ which is equivalent to the AHP if some suitable weights are used.

4 APPLICATION OF METHODS

In the following both the empirical Bayesian methods, represented in the chapter 3.1, and the subjective Bayesian method, represented in the chapter 3.2, are demonstrated with the aid of a data set of practical nature. the operating histories of the components corresponding to that under study have been collected from four nuclear power plants, Plant1, Plant2, Plant3 and Plant4 for the PSA study of the specific plant, Plant0. The task is to estimate the failure rate of this component by using the above methods.

The number of the failures, k_i, during the accumulated operating times, T_i, of the components at each plant until today are presented in the table 1. In addition, some general information about each plant is given in the table 1. this information is used for the evaluation of the relevance of the evidence data from the nuclear power plants.

The plants are of the same basic type. However, the plant vintages as well as the capacities vary from one plant to another. Furthermore, the suppliers of the reactors and other components are different and the plants are not situated in the same country.

The availability performance studies of nuclear power plants /10/ have indicated that some factors related to design characteristics as well as to the competence and co-operation of the nuclear energy parties (the power utility with its

Table 1. The operating histories of the components with some general plant information and the relevance probabilities

Plant	Size (MW)	Comm. oper.	Main supplier	Site country	Power utility	Evidence		Relevance probability ω
						k	T	
Plant1	500	10/71	A	a	I	22	99400	0.407
Plant2	520	1/74	A	a	I	4	91700	0.412
Plant3	1300	7/84	B	b	II	2	31800	0.332
Plant4	750	7/80	C	c	III	4	53400	0.505
Plant0	750	5/85	A	d	IV	0	14000	1.000

operational and maintenance staff, the national nuclear authority, the research institutes, the component suppliers, etc.) in the site country contribute to the availability performances of the nuclear power plants.

We have taken these results into account in assessment of the weights for the factors which we have assumed to characterize the relevance of the evidence data from the plants. We have collected the factors effecting on the evidence relevance into the hierarchy given in the figure 1. Then we have done subjective pairwise comparisons between the factors at each hierarchy level with respect to the goal at the next higher level and obtained the global evidence weights of the plants 0-4 using the AHP procedure with the computer code EC /8/. The obtained weights were then scaled to be used as the relevance probabilities ω_i ($\omega_0 = 1$ was fixed, see the equation (9)). The probabilities ω_i are given in the table 1. These probabilities are used in the application of the subjective Bayesian method.

The application of the empirical Bayesian method is straightforward and well known and thus it is not considered in more detail here.

The posteriors obtained with the methods are presented in the figure 2. The central characteristics of the distributions (mean, median) obtained with both methods are in good agreement. The posterior variance obtained with the empirical Bayesian method is significantly smaller than that obtained with the subjective method. This is due to the fact that the empirical method does not take the uncertainty caused by the irrelevance of the data sources into account. However, the posterior variance obtained by using the subjective method is strongly effected by the weights ω_i. The effect of weighting could have been analysed by various sensitivity studies. In this small example we did not perform any detailed investigations.

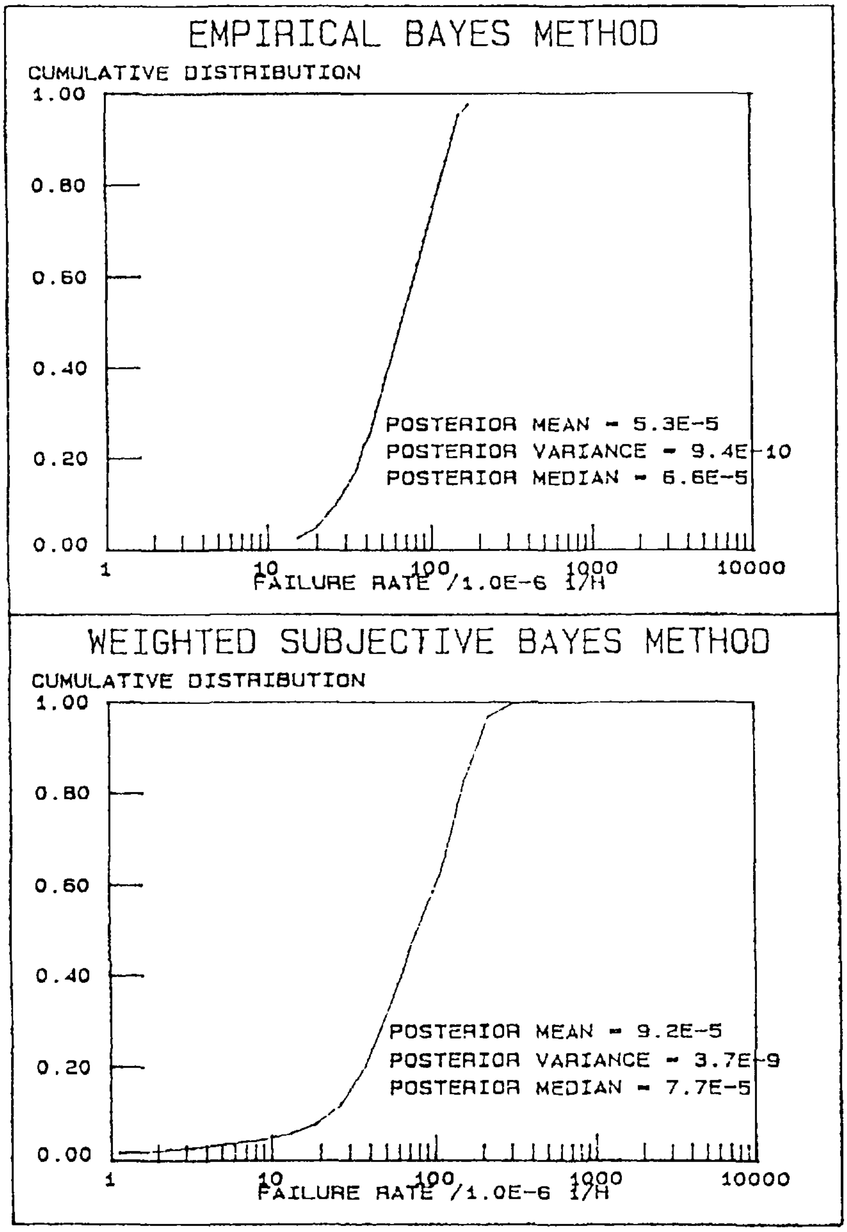

Figure 2. The posterior distributions.

5 CONCLUSIONS

We have considered two different Bayesian estimation methods in the cases where a few evidence data from both the plant under analysis and other plants are available. The methods were both of empirical and subjective nature.

In the empirical method the sample of operating histories N components, given in the form $E[1,\ldots,N] = \{(k_1,T_1),\ldots,(k_N,T_N)\}$, ($k_i$ = the number of failures in time T_i for component i) obeying the Poisson distribution, has been interpreted as an usable prior information and the plant specific operating experience $E_0=(k_0,T_0)$ has been used as evidence data. Because of the computational convenience the gamma prior was chosen to represent the prior information $E[1,\ldots,N]$. The parameters of this prior have been estimated both with marginal maximum likelihood method and with method of matching moments after which the gamma posteriors have been directly determined.

In the subjective Bayesian method the possible incompatibility of the evidences from components 1,...,N was taken into account. The incompatibility was described by introducing the evidence relevance probabilities ω_i for each of the components i = 1,...,N. The evidence relevance probability represents an index of the similarity between the specific plant and the plants from which the data have been gathered. In this way the uncertainty caused by partly non-relevant data could be analysed.

The evaluation of the evidence relevance probabilities has key role in the suggested subjective Bayesian method. Here we have proposed the use of the analytic hierarchy process (AHP) which is a systematic procedure for representing elements of any problem, in particular, the the decision problem hierarchycly.

In our application the hierarchy of similarities was constructed based on the availability performance on one hand, and on purely subjective intuition on another hand. In the light of our application the use of AHP for for evaluating probabilities was encouraging. Other possibilities, such as multiattribute utility decomposition, lead probably rather similar results.

The differencies between the empirical and the subjective approaches are connected with the uncertainty of the estimates. The central measures (mean, median) do not differ significantly. The advantage gained by using the subjective method is in the possibility to use the largest possible data base and to evaluate the uncertainty caused by judgements about the relevance of data. The probabilistic interpretation of the data relevance is not the only possibility, but it suits perfectly to the probabilistic risk assessments.

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DATA SPECIALIZATION FOR RELIABILITY STUDIES — AN APPLICATION TO DIESEL UNITS

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Abstract

The failure rates or failure probabilities of components used in PSA for specific plants are generally determined through the application of the Baye's theorem. Starting from information available in the literature, it is derived the generic distributions a priori which later on are specialized on the basis of statistics related to the behaviour of such components in the specific plant.

Consistent with the interpretation that those generic distributions express the plant-to-plant variability of the failure rates or failure probabilities of similar components, it has been suggested the convenience of considering the elements of the same type -in the same plant- as nominally identical and consequently, of grouping their individual statistics in one single value. This implies not only the derivation of specialized distributions common to such elements, but also that these should be considered as completed dependent if the intention is not underestimate the uncertainties associated to the failure probabilities of a system constituted by such elements.

This work contains an analysis based on the preceding concepts, made with the purpose of determining the failure probabilities on demand corresponding to two systems, each one constituted by two redundant diesel units. These units are respectively part of the emergency power supply and the emergency water supply systems of the Embalse's Candu-600 station.

In one of those systems, the derivations of the specialized distributions of the failure probabilities on demand has special interest due to the fact that periodic operation tests during three years has shown a zero failure result. It is demostrated that the hypothesis of the grouping of statisticdata related to components of the same type implies, in this case, a displacement of the specialized distribution curve to the left, that is, to optimistic values of the system's failure probability on demand.

On the other hand, it shows the calculation of the failure probability on demand regarding the two systems -starting from the same apriori distribution- but obtaining specialized specific distributions for each diesel unit. In other

words, components of the same type in a specific plant are treated as if they were distinguishables. The results show that, in this case, the full dependence assumption lead to more broadened distributions than those of the former approach.

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1. INTRODUCTION

Bayes' theorem is frequently used for deriving data on rates and probabilities of component failures in reliability studies or in PSA developments for a specific plant.

Apostolakis, Kaplan et al.[1] introduced a calculation method consisting, as a first step, in an assessment of an a-priori probability distribution for a given data, as from generic information available in literature. Then, such distribution is specialized by applying Bayes' theorem and statistics obtained from operational experience in the specific plant. The analytic expression of the specialized distribution or of the a-posteriori probability density function is:

$$P\,(f/E) \propto P\,(f) \cdot L\,(E/f)$$

where:

f: is the random quantity of interest (rate or probability of failure);

P(f): is the a-priori probability density function of random quantity f;

L(E/f): is the likelihood function (probability of obtaining evidence E, given an f value);

P(f/E): is the a-posteriori probability density function of f, given evidence E; and,

E: is the evidence (statistical data corresponding to the specific plant).

It has been understood that the a-priori probability density function, P(f), expresses the behaviour of individual components within a generic population concerning variability in failures. Such variability has been associated to different manufacturing qualities, to different plant-to-plant operation and maintenance conditions, to different designs, etc.

The same authors [2] suggest that it is convenient to consider components of the same type -within a given plant- as nominally identical elements and as members of the generic population represented by function P(f).

Consequently, the individual failure statistics concerning those components are given a single value, this -in turn- implying the derivation of a single function, P(f/E), expressing the degree of knowledge attained on the distribution of the failure probability or rate among those components.

Being consistent with this interpretation, the authors [2] demonstrate that, when the failure probability of systems composed by similar elements is calculated, the failure behaviour being represented by identical P(f/E) functions, these functions must be considered as fully dependent so as not to disregard uncertainties in the values of such failure probability.

This paper introduces the analysis of two systems, each one of them composed by two redundant diesel units, to which the above described methodology is applied in order to calculate the failure-to-start probability of those systems.

However, the treatment of the evidence -that is, of the statistical data corresponding to the operational experience in both systems- is based on two different hypothesis concerning the distinguishability of the units in each system. The first hypothesis considers the diesel units in each system as identical elements and, consequently, the number of failures, r_i, and the number of tests, n_i, are shown as two single values. The set of data:

$$\left\{ \sum_i r_i \ , \ \sum_i n_i \right\} \quad i = 1,2$$

represents a failure statistics that is common to both components in each one of the systems.

The second hypothesis assumes that the diesel units in each system are distinguishable; consequently, the information on the number of failures and on the number of tests corresponding to each one of the units is retained. The set of data:

$$\left\{ (r_1 \ , \ n_1), \ (r_2 \ , \ n_2) \right\}$$

represents, in this case, failure statistcs for each one of the components in each one of the systems.

Section 2 shows the development of calculations on specialization in the probability distribution corresponding to each one of the formulated hypothesis. In section 3, the reliability is calculated for start-up demand in both systems. Considering the first hypothesis, the failure-to-start probability is assessed by assuming a system model with two fully dependent in-parallel components, the calculation being adjusted to what was expressed in [2]. Concerning the second hypothesis, the same model was assumed but the calculation of the failure probability for the two systems was performed by applying the Montercarlo sampling technique with Generalized Knowledge Dependence developed by R. M. Cooke and R. Waij [3].

Finally, section 4 provides a discussion of the results obtained by considering both hypotheses and formulates the corresponding conclusions.

2. DATA SPECIALIZATION

2.1. Description of the systems

The analysis was applied to the diesel units in the Emergency Power Supply (EPS) and Emergency Water Supply (EWS) systems of the CANDU-600 Embalse nuclear power plant. These systems are aimed at supplying essential electric power for the instrumentation and the comand of valves and at feeding water to the steam generators, respectively, in case of occurrence of a severe earthquake (DBE).

The EPS system is constituted by two redundant diesel moto-generators, each one with a generating capacity of 50 KW - 380 V AC. The EWS system is consituted by two redundant 75 HP diesel engines, coupled to their respective pumps. Both systems are absolutely autonomous as far as their auxiliary services are concerned, seismically qualified and housed in separate seismic-resistant buildings.

2.2. Generic data

The failure-to-start probability, f, was taken from the Reactor Safety Study [4]; it was assumed that this random quantity is a-priori log-normal distributed with the following parameters:

$$f05 = 1.0E-2$$

$$f50 = 3.16E-2$$

$$f95 = 1.0E-1$$

where f05 and f95 correspond with the lower and the upper bound, respectively, of the failure-to-start probability, Q_d, shown in Table III.2.3. of the Reactor Safety Study [4]. The f log-normal distribution is discretized in the interval between Z0.5 and Z95 percentiles of the corresponding normal distribution (Table 1). Twenty class intervals were used in order to attain an acceptable accuracy in the results of the a-priori distribution calculation.

TABLE 1. DISCRETIZATION OF THE A PRIORI LOGNORMAL DISTRIBUTION OF f

Z	Center of class	X	f	P (f)
-2.8104				
-2.5759	-2.6932	-5.3397	.0048	.005
-2.3414	-2.4587	-5.1746	.0057	.005
-2.1069	-2.2242	-5.0105	.0067	.008
-1.8724	-1.9897	-4.8464	.0079	.013
-1.6379	-1.7552	-4.6823	.0093	.020
-1.4034	-1.5207	-4.5182	.0109	.030
-1.1689	-1.2862	-4.3541	.0128	.041
-0.9344	-1.0517	-4.1900	.0152	.054
-0.6999	-0.8172	-4.0259	.0178	.067
-0.4654	-0.5827	-3.8618	.0210	.079
-0.2309	-0.3482	-3.6977	.0248	.088
-0.0036	-0.1137	-3.5336	.0292	.090
+0.2381	+0.1208	-3.3695	.0344	.095
+0.4726	+0.3553	-3.2054	.0405	.088
+0.7071	+0.5898	-3.0413	.0478	.078
+0.9416	+0.8243	-2.8772	.0563	.066
+1.1761	+1.0588	-2.7130	.0663	.053
+1.4106	+1.2933	-2.5489	.0782	.041
+1.6451	+1.5278	-2.3848	.0921	.029
+1.8796	+1.7623	-2.2207	.1086	.050
				1.000

$X = \sigma_X \cdot Z + \mu_X = 0.7 \cdot Z - 3.454$

$f = \mathrm{Exp}\ (X)$

2.3. Assessment of the a-posteriori distribution of the failure-to-start probability

The statistics on failures to start -obtained from periodical tests performed regularly in both systems- show the following figures for the first three years in operation [5]:

	Number of tests (n)	Number of failures (r)
EPS System		
Unit G1	101	0
Unit G2	96	0
Total EPS	197	0
EWS System		
Unit P1	122	1
Unit P2	115	2
Total EWS	237	3

FIG.1. Failure statistics.

A binomial distribution was adopted for the assessment of the likelihood function of the evidence, considering that such distribution was compatible with the failure-to-start model. Its analytical expression is:

$$L(E/f) = \binom{n}{r} f^{r} (1-f)^{n-r}$$

where:

L(E/f): is the likelihood function of the evidence;

f: is the failure-to-start probability;

n: is the number of tests performed;

r: is the number of failures occurred during the n tests.

The a-posteriori distribution of f is calculated on the basis of two different hypothesis concerning the distinguishability of the diesel units in each one of the systems. These hypothesis were:

Hypothesis (a): it is assumed that both units are nominally identical. Their failure statistics are thus reduced to a single pair of values (See Figure 1):

EPS System: $r = 0$; $n = 197$
EWS System: $r = 3$; $n = 237$

Tables 2 and 3 show the results of the calculations performed as per this hypothesis, for the a-posteriori distributions of f.

Hypothesis (b): it is assumed that the diesel units in the two systems are different and distinguishable entities. Their failure statistics are (see Figure 1):

EPS System: $r_1 = 0$; $n_1 = 101$
$r_2 = 0$; $n_2 = 96$

EWS System: $r_1 = 1$; $n_1 = 122$
$r_2 = 2$; $n_2 = 115$

Tables 4, 5, 6 and 7 show the results of the calculations performed as per this hypothesis, for the a-posteriori distributions of f.

3. RELIABILITY OF THE SYSTEMS

3.1. Hypothesis (a)

As stated under 2.3., it is assumed that both diesel units in each system are nominally identical and that their behaviour concerning failures to start may be described by a single distribution function P(f/E) and is applicable to both units. Consequently, the calculation of the failure probability of the system, Q, is adjusted to the procedure developed in [2] for a configuration in parallel.

Below are the mean, variance, median and 5 and 95 percentile values of the Q distribution corresponding to the EPS and EWS systems, respectively, calculated on the basis of the above described conditions and of the already known ratios among parameters of the log-normal distribution.

EPS system

Mean: $\alpha_Q = \alpha^2 + \beta^2 = 1.7E-4$

Variance: $\beta^2_Q = 3.4E-8$

Q05 = 2.5E-5

Q50 = 1.1E-4

Q95 = 4.8E-4

EWS system

Mean: $\alpha_Q = \alpha^2 + \beta^2 = 3.8E-4$

Variance: $\beta^2_Q = 1.1E-7$

Q05 = 7.9E-5

Q50 = 2.8E-4

Q95 = 9.9E-4

The calculation is performed by using results shown in Tables 2 and 3.

TABLE 2. CALCULATION OF THE A POSTERIORI DISTRIBUTION OF f FOR THE EPS SYSTEM

f	P(f)	L(E/f)	P(f)L(E/f) (x10-3)	P(f/E)
.0048	.005	.388	1.94	.078
.0057	.005	.327	1.51	.061
.0067	.008	.267	2.13	.085
.0079	.013	.211	2.75	.110
.0093	.020	.160	3.22	.129
.0109	.030	.115	3.40	.136
.0128	.041	.078	3.20	.128
.0152	.054	.049	2.66	.107
.0178	.067	.029	1.93	.077
.0210	.079	.015	1.20	.048
.0248	.088	.007	.63	.025
.0292	.090	.003	.26	.010
.0344	.095	.001	.10	.004
.0405	.086	-	.03	.002
.0478	.073	-	.01	-
.0563	.066	-	-	-
.0663	.053	-	-	-
.0782	.041	-	-	-
.0921	.029	-	-	-
.1086	.050	-	-	-
	1.000		24.97	1.000

$L(E/f) = (1-f)^{197}$

Histogram mean: α = .012

Histogram variance: β^2 = 3.0E-5

Lognormal percentiles: f05 = 5.0E-3
f50 = 1.1E-2
f95 = 2.2E-2

TABLE 3. CALCULATION OF THE A POSTERIORI DISTRIBUTION OF f FOR THE EWS SYSTEM

EWS System.

f	P(f)	L(E/f)	P (f)L(E/f) (x10-3)	P(f/E)
.0048	.005	.079	.39	.005
.0057	.005	.105	.48	.005
.0067	.008	.136	1.08	.014
.0079	.013	.168	2.18	.029
.0093	.020	.197	3.97	.053
.0109	.030	.218	6.45	.086
.0128	.041	.225	9.23	.123
.0152	.054	.214	11.52	.153
.0178	.067	.184	12.33	.164
.0210	.079	.141	11.11	.148
.0248	.088	.094	8.25	.110
.0292	.090	.053	4.77	.063
.0344	.095	.025	2.36	.031
.0405	.086	.009	.60	.011
.0478	.073	.003	.20	.003
.0563	.066	.001	.03	.001
.0663	.053	-	-	-
.0782	.041	-	-	-
.0921	.029	-	-	-
.1036	.050	-	-	-
	1.000		75.15	1.000

$L(E/f) = \binom{237}{3} f^3 (1-f)^{234}$

Histogram mean: α = .018

Histogram variance: β^2 = 5.1E-5

Lognormal percentiles: f05 = 8.9E-3
f50 = 1.7E-2
f95 = 3.1E-2

3.2. Hypothesis (b)

In this case the calculation is made for two distributions with different failure-to-start probabilities, corresponding to each one of the diesel units in the EPS and EWS systems (Tables 4, 5, 6 and 7). The failure-to-start probability of the parallel system is simply the product of the failure probabilities of each one of their components.

Such product was performed, for different degrees of knowledge dependence, between the probability distributions in each one of the components by means of the application of the Montecarlo sampling technique with Generalized Knowledge Dependence, developed in [3].

Below are the mean, variance, median and 5 and 95 percentile values of the Q distribution corresponding to the EPS and EWS systems, respectively, resulting from the Montecarlo sampling for different values of parameter d[3] in the range betwen 0 (total independence) and 1 (total dependence).

TABLE 4. CALCULATION OF THE A POSTERIORI DISTRIBUTION OF f FOR THE EPS SYSTEM — G1 UNIT

f	P(f)	L(E/f)	P(f)L(E/f) ($x10^{-3}$)	P(f/E)
.0048	.005	.615	3.07	.033
.0057	.005	.564	2.60	.028
.0067	.008	.509	4.04	.044
.0079	.013	.463	6.02	.065
.0093	.020	.391	7.87	.085
.0109	.030	.330	9.75	.105
.0128	.041	.271	11.10	.120
.0152	.054	.214	11.52	.125
.0178	.067	[illegible]	10.86	.117
.0210	.079	.117	9.21	.100
.0243	.085	.079	6.90	.075
.0292	.090	.050	4.51	.049
.0344	.095	.029	2.78	.030
.0405	.088	.015	1.34	.015
.0478	.078	.007	.56	.006
.0563	.066	.003	.19	.002
.0663	.053	.001	.05	.001
.0782	.041	-	.01	-
.0921	.029	-	-	-
.1086	.020	-	-	-
	1.000		92.44	1.000

$L(E/f) = (1-f)^{401}$

Histogram mean: α = .016

Histogram variance: β^2 = 7.1E-5

Lognormal percentiles: f05 = 6.3E-3
f50 = 1.4E-2
f95 = 3.2E-2

TABLE 5. CALCULATION OF THE A POSTERIORI DISTRIBUTION OF f FOR THE EPS SYSTEM — G2 UNIT

f	P(f)	L(E/f)	P(f)L(E/f) (x10^{-3})	P(f/E)
.0048	.005	.630	3.15	.031
.0057	.005	.580	2.67	.027
.0067	.008	.526	4.18	.042
.0079	.013	.469	6.10	.061
.0093	.020	.409	8.25	.082
.0109	.030	.349	10.30	.103
.0128	.041	.289	11.84	.118
.0152	.054	.231	12.43	.124
.0178	.067	.177	11.88	.119
.0210	.079	.130	10.25	.102
.0248	.088	.090	7.89	.079
.0292	.090	.058	5.23	.052
.0344	.095	.035	3.31	.033
.0405	.088	.019	1.65	.016
.0479	.078	.009	.71	.007
.0563	.066	.004	.26	.003
.0663	.053	.001	.07	.001
.0782	.041	-	.02	-
.0921	.029	-	-	-
.1086	.050	-	-	-
	1.000		100.19	1.000

$L(E/f) = (1-f)^{96}$

Histogram mean: α = .016

Histogram variance: β^2 = 7.7E-5

Lognormal percentiles: f05 = 6.4E-3
f50 = 1.5E-2
f95 = 3.3E-2

TABLE 6. CALCULATION OF THE A POSTERIORI DISTRIBUTION OF f FOR THE EWS SYSTEM — P1 UNIT

f	P(f)	L(E/f)	P(f)L(E/f) (x10^{-3})	P(f/E)
.0048	.005	.327	1.63	.013
.0057	.005	.347	1.60	.013
.0067	.008	.362	2.88	.024
.0079	.013	.369	4.80	.039
.0093	.020	.367	7.39	.061
.0109	.030	.353	10.42	.086
.0128	.041	.328	13.43	.110
.0152	.054	.291	15.69	.129
.0178	.067	.246	16.50	.135
.0210	.079	.196	15.45	.127
.0248	.088	.145	12.74	.105
.0292	.090	.099	8.87	.073
.0344	.095	.061	5.80	.047
.0405	.088	.033	2.90	.024
.0478	.078	.016	1.22	.010
.0563	.066	.006	.41	.003
.0663	.053	.002	.11	.001
.0782	.041	.001	.02	-
.0921	.029	-	-	-
.1086	.050	-	-	-
	1.000		121.86	1.000

$L(E/f) = \binom{422}{1} f\,(1-f)^{421}$

Histogram mean: α = .019
Histogram variance: β^2 = 8.1E-5
Lognormal percentiles: f05 = 7.9E-3
f50 = 1.7E-2
f95 = 3.6E-2

TABLE 7 CALCULATION OF THE A POSTERIORI DISTRIBUTION OF f FOR THE EWS SYSTEM — P2 UNIT

f	P(f)	L(E/f)	P(f)L(E/f) (x10-3)	P(f/E)
.0048	.005	.088	.44	.003
.0057	.005	.111	.51	.003
.0061	.008	.137	1.09	.007
.0079	.013	.166	2.16	.015
.0093	.020	.196	3.96	.027
.0109	.030	.226	6.67	.045
.0128	.041	.251	10.28	.070
.0152	.054	.268	14.43	.098
.0178	.067	.273	18.27	.124
.0210	.079	.263	20.70	.140
.0248	.086	.236	20.74	.141
.0292	.090	.196	17.65	.120
.0344	.095	.148	14.18	.096
.0405	.088	.100	8.79	.060
.0478	.078	.059	4.65	.032
.0563	.066	.030	1.98	.013
.0663	.053	.012	.66	.005
.0782	.041	.004	.16	.001
.0921	.029	.001	.03	-
.1086	.050	-	.01	-
	1.000		147.36	1.000

$$L(E/f) = \binom{415}{2} f^2 (1-f)^{413}$$

Histogram mean: α = .024

Histogram variance: β^2 = 1.2E-4

Lognormal percentiles: f05 = 1.6E-2
f50 = 2.2E-2
f95 = 4.4E-2

EPS System

	d = 0	d=0.2	d=0.4	d=0.6	d=0.8	d=1.0
α_a	2.6E-4	2.8E-4	2.9E-4	3.1E-4	3.3E-4	3.4E-4
β^2_a	4.3E-8	6.1E-8	8.9E-8	1.3E-7	1.6E-7	1.8E-7
Q05	6.5E-5	5.8E-5	5.2E-5	4.7E-5	4.3E-5	4.2E-5
Q50	2.1E-4	2.1E-4	2.1E-4	2.1E-4	2.1E-4	2.1E-4
Q95	6.5E-4	7.3E-4	8.3E-4	9.2E-4	1.0E-3	1.0E-3

EWS System

	d = 0	d=0.2	d=0.4	d=0.6	d=0.8	d=1.0
α_a	4.4E-4	4.6E-4	4.9E-4	5.1E-4	5.3E-4	5.4E-4
β^2_a	9.6E-8	1.3E-7	1.9E-7	2.5E-7	3.1E-7	3.4E-7
Q05	1.3E-4	1.2E-4	1.0E-4	9.4E-5	8.8E-5	8.6E-5
Q50	3.6E-4	3.6E-4	3.6E-4	3.6E-4	3.6E-4	3.6E-4
Q95	1.0E-3	1.1E-3	1.3E-3	1.4E-3	1.5E-3	1.5E-3

3.3. Comparison of results

The results shown under 3.1. and 3.2. above are shown as a graph in Figure 2, so as to facilitate their comparison. The mean, median and 5 and 95 percentile values of the failure-to-start probability for both systems are represented in a logaritmic scale.

For the case of hypothesis (a), the graph shows the values calculated in section 3.1. and corresponding to total dependence between the failure probability distributions of both components. For the case of hypothesis (b), the values shown correspond with different degrees of dependence that are expressed by the values of parameter d.

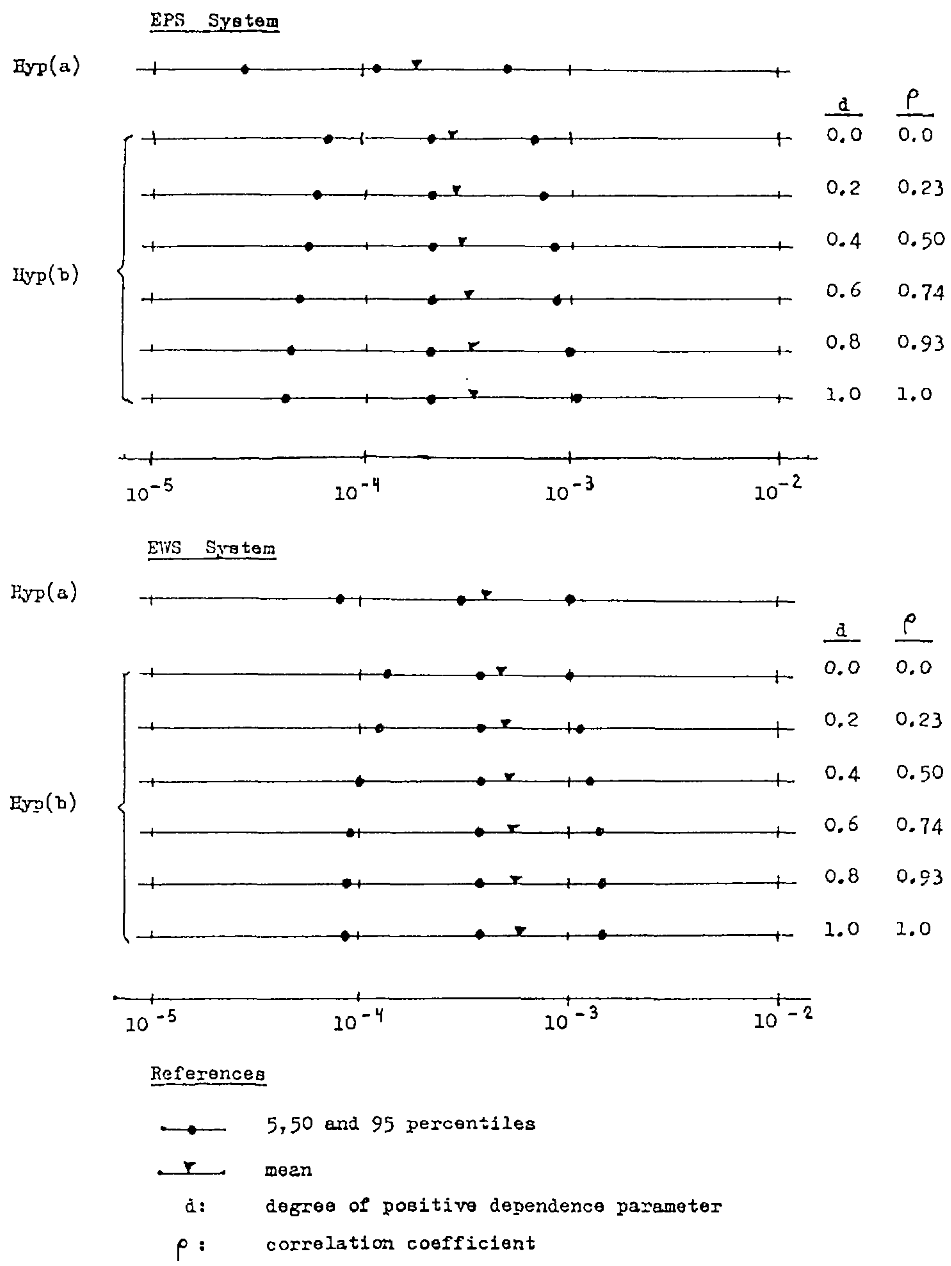

FIG.2. Comparison of results.

4. CONCLUSIONS

The first conclusion arises from the comparison of the values of the parameters in the failure-to-start probability distribution of the EPS system. This system shows a singular condition -which appears frequently in component failure statistics-: it did not fail at all throughout all the periodical tests performed during three years.

It may be observed that hypothesis (a), in which a grouping of the data is considered, leads to a distribution that is sensibly displaced toward the left, that is, to optimistic Q values, if compared with the distribution resulting from considering hypothesis (b), in which the individual information corresponding to each component in the system is maintained.

The information provided by one statistical approach $\{r_1 + r_2 = 0;\ n_1 + n_2 = 197\}$ is essentially different from that provided by the other $\{(r_1 = 0;\ n_1 = 101)\ (r_2 = 0;\ n_2 = 96)\}$, this explaining the displacement observed in the failure probability distributions of the EPS system.

In the case of the EWS system, the displacement between the probability distributions derived on the basis of hypothesis (a) and (b) is practically regardless; in other words, no relevant differences exist between the respective mean and median values.

The second conclusion concerns the dispersion parameters of the failure-to-start probability distribution of systems EPS and EWS and the dependence, or coupling degree, of the probability distributions corresponding to each one of the diesel units involved in those systems.

Considering the distributions derived within the framework of hypothesis (b) as fully coupled does not seem to be adequate because, although they originate from a single distribution -that describing the population-, they are later specialized for two components showing different failure behaviours, thus resulting in different distributions for each one of those components (as may be seen in Tables 4 through 7).

However, the distinguishability hypothesis concerning both diesel units is compatible with reality, since it is improbable that they show an identical failure behaviour during their lifetime. In fact, this hypothesis implies a more complex model and a higher complexity brings along a higher degree of associated uncertainty. Thus, it was considered convenient to assume that the probability distributions developed on the basis of hypothesis (b) be fully coupled. This implied -as may be seen in Figure 2- that the failure-to-start probability distributions for both systems are more broadened than those developed on the basis of hypothesis (a).

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A REVIEW OF INITIATING EVENTS SELECTION AND FREQUENCIES FOR PSA STUDIES

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Abstract

Initiating events (IEs) are the occurrences that initiate accident sequences that potentially lead to core damage in Nuclear Power Plants (NPP). These occurrences can be experienced during the operating life of a plant, recorded and in cases that their frequency appears to be unexpectedly high, may initiate corrective actions.

One of the tasks in performing a Probabilistic Safety Assessment (PSA) calls for initiating events selection and frequency estimation. The review given here discusses the data sources utilized for performing this task. It brings examples from actual PRAs, and some special probabilistic studies that were published.

Two types of sources for IEs identification and frequency estimation are discussed:

a) LER compilations intended for presenting IE frequencies. The history of these data sources for transients, for Loss of Offsite Power (LOSP) and for Loss of Coolant Accident (LOCA) is given.

b) Fault-Tree analyses, or other probabilistic studies, in which a model is constructed and analyzed to compute a certain frequency of a special IE. This method is used mainly in cases of infrequent IEs having a small probability of occurrence.

The discussion of initiating events selection and frequencies is presented in the paper via a breakdown into several subgroups discussed separately:

(1) Transients

(2) Loss of Offsite Power (LOSP)

(3) Loss of Coolant Accident (LOCA) including Interfacing LOCA

(4) Special Initiating Events

(5) Internal Flooding

Relevant methodologies and data for these topics are given in the paper and the treatments performed in several types of PRA are reviewed and compared. Several recommendations are given on the more suitable data sources for PSA and for further data improvement.

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1. INTRODUCTION

Initiating events (IEs) are the occurrences that initiate accident sequences that potentially lead to core damage in Nuclear Power Plants (NPP). These occurrences can be experienced during the operating life of a plant, recorded and in cases that their frequency appears to be unexpectedly high, may initiate corrective actions.

One of the tasks in performing a Probabilistic Risk Assessment (PRA) calls for initiating events selection and frequency estimation. The review given here discusses the data sources utilized for performing this task. It brings examples from actual PRAs, and some special probabilistic studies that were published. There are in general two types of sources for IEs indentification and frequency estimation:

a) LER compilations intended for presenting IE frequencies. The history of these data sources for transients, for Loss of Offsite Power (LOSP) and for Loss of Coolant Accident (LOCA) is given.

b) Fault-Tree analyses, or other probabilistic studies, in which a model is constructed and analyzed to compute a certain frequency of a special IE. This method is used mainly in cases of infrequent IEs having a small probability of occurrence.

The data on failure rates for the above task (b) is derived from LERs as well. It is rather difficult to collect the data from the plant-specific failure and corrective action program. A large number of events may be needed for establishing an adequate data base for failure rate derivation. If sufficient plant specific data is not available, generic failure rates are utilized to evaluate the frequency of the special low frequency IEs.
An IE frequency can be made plant-specific by using a two stage Bayesian update process. In this case the generic compilations of IE frequency data are used as prior, and the collected plant specific occurrences are used to evaluate the posterior. The same technique can be used also for updating failure rates for the use in Fault-Tree analyses for IE frequency estimations.

The discussion of initiating events selection and frequencies is presented in the paper via a breakdown into the following subgroups which are separately discussed in the subsequent sections:

(1) Transients

(2) Loss of Offsite Power (LOSP)

(3) Loss of Coolant Accident (LOCA) including Interfacing LOCA

(4) Special Initiating Events

(5) Internal Flooding

This discussion, however, is limited to US type BWR and PWR plants.

Relevant methodologies and data for these groups of IEs are given in each section and the treatments performed in several types of PRA are reviewed and compared.

2. TRANSIENT INITIATING EVENTS

Two classes of transient IEs are discussed in this section: transients with successful scram and Anticipated Transients Without Scram (ATWS). The data for the frequency estimation of both classes is obtained from the same data source, using different treatments on the basic data.

2.1 Transient Data Base

The initial review of initiating event data was performed by the Reactor Safety Study[1] (RSS). Twenty transient IE categories were selected for BWRs and twenty-three for PWRs. Electric Power Research Institute (EPRI) published in 1978 its first study of anticipated transients[2]. This work which will be referred to as NP-801, included compilation of operational occurrences in 12 BWRs and 30 PWRs. For BWRs it reported 459 events in 37 selected categories of different IEs. For PWRs it reported ~1000 events categorized in 41 different IEs. The data in NP-801 covered NPP experience up to 1979 which was the equivalent of 47 plant years and 131 plant years for BWR and PWR respectively. In 1982 EPRI published an update in report NP-2300[3]. It reported 903 events for BWRs and 2093 events for PWRs under the same IE categories. The number of plants covered increased to 16 BWRs and 36 PWRs and the amount of plant years covered was 101.5 plant years and 213 plant years for BWR and PWR respectively.

Some of the changes made in NP-2230 relative to the NP-801 study are:

a) NP-801 had used an "effective in-service date" as supplied by the utilities. NP-2230 uniformly used the first day of commercial operation as the starting point for reporting plant anticipated transients. Because of this change, 137 events of NP-801 were excluded from NP-2230 update.

b) NP-801 reports 191 events within 37 plant years that occurred in the years subsequent to the first year of plant operation (less than half the total number of events). NP-2230 reports 647 events in 85.5 plant years (70% of the total).

It is clear that NP-801 included very early periods of plant operations, such as from criticality to commercial operation, whereas NP-2230 included events that occurred only after commercial operation was initiated. In general this is about a half year later. Table 1 is a comparison of the evaluation of several initiator frequencies based on these two data sources.(4)

Table 1: SNPS-PRA and BNL Results for Initiator Frequency and Sources of Differences

	SNPS-PRA: EPRI-NP-801 Data[5]				BNL Review: EPRI-NP-2230 Data[6]				BNL Review: Two-Stage Bayesian	
Transient	1st Year	Subseq. Years	All Years Average	SNPS-PRA Weighted Average*	1st Year	Subseq. Years	All Years Average	Weighted Average	Subseq. Years	All Years**
Loss of Condenser Vacuum (2,4,8)	1.6	0.38	0.67	0.41	1.0	0.38	0.47	0.40	0.40	0.50
Turbine Trip	16.9	4.14	7.3	4.46	13.4	6.39	7.39	6.59	6.85	7.89
MSIV Closure (5)	2.2	0.19	0.67	0.24	1.67	0.27	0.47	0.31	0.29	0.57
Loss of FW (22)	0.6	0.16	0.27	0.18	0.27	0.11	0.13	0.12	0.11	0.13
LOOP (31)	0.4	0.11	0.16	0.08+	0.13	0.12	0.12	0.08+	0.12	0.15++
IORV (11)	0.7	0.08	0.20	0.09	0.53	0.15	0.21	0.16	0.19	0.25
CRW (27, 28)	0.1	0.03	0.04	0.03	0.13	0.10	0.11	0.10	0.11	0.12
Total	22.5	5.09	9.3	5.49	17.1	7.52	8.9	7.76	8.07	9.65

+Based on SNPS grid data.
++Based on NSAC-80 report[10].
*Used in the PRA.
**Used in the BNL review.

The first four columns of the table show a BWR-PRA results. The next four columns represent results obtained from applying the same methodology to the more recent data source (NP-2230). The two last columns present results using the updated source and the two stage Bayesian methodology[5, 6]. It can be seen that most of the increase in BNL reviews initiator frequencies derives from the updated experience of BWR related events, rather than from the use of the Bayesian methodology.

The BWR-PRA differentiated between the impact of failures during the first year of plant operation and failures occurring in later years. However, in the review[4] it was argued that the data base of NP-801 used was not sufficiently refined for this purpose. The latter update, given in NP-2230, showed that the impact of ignoring the first year of plant operating experience causes a reduction of about 20% in initiator frequencies (see last two columns of Table 1). In addition, the "weighted average" approach utilized in the BWR-PRA weighted the data from the first year as (1/35) and the data from subsequent years as (34/35). This approach apparently results in small underestimation of initiator frequencies, since we lack experience from aging plants (after 30 to 40 years of operation).

Table 1 depicts that the number of shutdowns due to anticipated transients is higher than experienced in the recent years. This is apparently because the NP-2230 data base extends to 1981 only. An updated review of the experiential data was published in 1985 by INEL[7].

The INEL study updates the data base and covers additional 10 BWR and 14 PWR plants. It updated the IE occurrences until December 1983 so that the data base includes 251 BWR plant years with 1832 events and 423 PWR plant years with 3574 events. Now, most of the events come from years following the first two years of operation.

The INEL study reviewed the data base of NP-2230 and determined that:

a) INEL basically concurs with the choice of initial day of commercial operation as the starting point for event recording.

b) INEL judges that the NP-2230 categories of IEs are adequate*, but in need of better definitions to accurately describe the event data.

*"adequate" - include "fine" categorization which allow the PRA analyst much flexibility in combining events into transients groups according to the specific plant responses.

The new INEL data falls too many times into the more broad categories PWR-37 and BWR-34. More detailed breakdown of these categories can help the PRA analyst.

c) The BWR and PWR IE categories are comparable in most events and the main difference is in the treatment of the condenser originated transients.

d) While accepting the NP-2230 categorization INEL suggests a new scheme of 50 BWR (instead of 37) and 56 PWR (instead of 41) IE categories for study in the future. Full event definitions should be developed for the new scheme to enable their future use.

The above findings were based on a thorough comparison of the NP-2230 data base and the NRC Gray Book(8)* data base for 11 plants which were selected for comparison. For each plant selected the events that occurred during the third and eighth year of operation were carefully compared. It was found that 66 (27%) events were categorized differently based on the event description in each of the two sources compared. However, about one third of the discrepancies were because the Gray Book did have less information than in the NP-2230 description. The final conclusion was that on the whole the NP-2230 data was found to be valid and is indicative of US commercial NPP experience. This is because the deviations were small (and not random) and in general, did not cross "borders" of the broad groups of transients used in the PRA studies. Another important conclusion was that sufficient amount of detail in the event descriptions is crucial for correct categorization.

Many PRAs used the NP-801 data source(2). Only newer PRAs such as the Oconee PRA(9) that were prepared after 1983 used the NP-2230. None of the published PRA today used the INEL data base.

2.2 An Application of NP-2230 in the Oconee PRA

The Oconee is a plant that has three units all of them operating for several years. Therefore, a significant plant-specific experience was accumulated. These plants' specific records were used to modify the

*USNRC's Operating Units Status report (Gray Book) generic data given in NP-2230 by using the two stage Bayesian method. In addition the plant-specific events were also used to modify the list of initiators, i.e., additional IEs were considered beyond the standard 41 IEs of NP-2230 (e.g. "Loss of ICS bus KI" or "Low PZR pressure signal"). Furthermore, based on the original list of the IEs in the RSS(1) (Table I.4-9 there), it was decided to include steam-line and FW-line break initiators in the list of Oconee transients. Thus, the Oconee PRA included 10 IEs collapsed from the 41 PWR IEs and four IEs beyond the standard list of NP-2230 (or INEL).

The Oconee PRA frequency evaluation provides additional insight on the use of NP-2230 data. The PRA analysts reevaluated the records of the plant transient events and came out with a different number of events than in the NP-2230 transient categories for Oconee; the main difference was in the number of cases of the "partial loss of MFW" transient. Many of these events were recategorized as "turbine trip" in Oconee PRA instead of "partial loss of MFW" in NP-2230. In the course of the review(10) of the Oconee PRA additional considerations were included which resulted in some small changes in the frequency determination of the IEs as can be seen in Table 2. The table provides also a comparison with three other probabilistic studies.

Table 2: Comparison of OPRA and BNL Initiator Frequencies With Several Other Studies

Initiator	Arkansas[1] IREP	Midland PRA[3]	B/W Owner Group	OPRA	BNL Review
T_1: {Turbine trip / Reactor trip	7.1	6.1 / 1.9	{4.1	{4.9	{4.9
T_2: Loss of MFW	1.0	0.7	0.9	0.64	0.5
T_3: Partial loss of MFW				0.69	0.69
T_4: Loss of condenser vacuum				0.21	0.21
T_5: LOOP	0.32	0.135	0.14	0.17	0.12
T_6: Loss of air				0.17	0.21
T_7: {Excessive FW flow / MUPS malfunction	-	0.22 / 2.9E-3	0.22	0.092	0.092
T_8: Spurious ESF	-	-	-	0.01	0.01
T_9: SLB and TBV failure	-	6.7E-3	}0.052	3.-3	0.053
T_{10}: FW-line break	-	-	-	9.3E-4	9.3E-4
T_{11}: ICS malfunction	-	0.055	0.048	0.02	0.05
T_{12}: {Loss of SWS	2.6E-3	3.7E-6	-	4.0E-3	4.9E-3
Loss of CCW	-	4.1E-5	-	-	-
T_{13}: Stuck open spray	-	-	-	0.044	0.044
T_{14}: {Loss of ac power bus / Loss of dc power bus	0.035 / 0.036	- / -	- / -	5.4E-3	5.4E-3
R: SG tube rupture	-	0.014	0.017	8.6E-3	8.6E-3
RPV: RPV rupture	-	-	-	1.1E-6	1.1E-6
Aout: Interfacing LOCA	-	7.7E-7	-	1.4E-7	
VS: Very small LOCA	0.020	5.0E-3	8.3E-3*	-	3.0E-3
S: Small LOCA	6.9E-4	3.3E-3	4.E-4	3.0E-3	3.0E-3
M: Medium LOCA	1.6E-4	4.7E-4	-	-	-
A: Large LOCA	8.7E-5	2.0E-4	-	} 9.3E-4	}9.3E-4

*Taken from the ORNL precursor study NUREG/CR-2497. It includes induced LOCAs (about 40%).

2.3 ATWS Selection and Frequencies

The data base of NP-2230 or INEL is suitable for the derivation of ATWS frequencies and for the selection of several types of ATWS which require a different plant response. Table 3 (taken from Ref. 10) provides an example of how the NP-2230 data is used for the evaluation of ATWS in the Oconee PRA.

Table 3: Mean Annual Frequencies of Transient Categories at Oconee (from EPRI-NP-2230)

Transient Category	EPRI-NP-2230 Grouping	All Power Levels		Power Level Greater than 25%	
		All years (19.8 years)	Subsequent years (16.8 years)	All years (19.8 years)	Subsequent years (16.8 years)
Loss of condenser vacuum	25,27,30	0.25	0.12	0.20	0.12
Turbine trip	3,9,12,15,19, 21,23,28,33, 34,36-40	5.10	4.40	4.30	3.70
Loss of main feedwater	16,22,24	0.70	0.48	0.35	0.18
Loss of offsite power	35	0.05	<0.01	0.05	<0.01
Load increase	26,29	0.05	0.06	0.0	<0.01
Loss of RCS flow	1,14	0.25	0.18	0.15	0.12
Control rod withdrawal	2	0.10	0.12	0.10	0.12
RCS depressurization	4,5,7	0.05	0.06	0.05	0.06
Boron dilution	11	<0.01	<0.01	<0.01	<0.01
Excessive cooldown	6,20	<0.01	<0.01	<0.01	<0.01
MSIV closure	17,18	N/A	N/A	N/A	N/A
Inactive RCS loop startup	13	<0.01	<0.01	<0.01	<0.01
Total		6.60	5.50	5.20	4.30

The first column shows the grouping which is performed according to the specific plant response. The frequency of each ATWS category is evaluated for all years including first year of operation (column 2) and for subsequent years after the first year of operation (column 3). Because an ATWS from power level smaller than 25% is considered benign, the frequencies for this case are also evaluated, and they are used in the PRA analysis.

An example of the use of NP-801 is in the Shoreham PRA[11]. The frequency of ATWS challenges to the scram system as evaluated for power level greater than 25% was 3.87 per year. In the review[4] the NP-2230 was used and the ATWS challenge frequency raised to 7.34 per year which is higher by almost 100%. This is because NP-801, has 60% of the data from the first year of plant operation which includes many cases of low power testing. In NP-2230 the time period before commercial operation was removed. Thus, in the latter, only 33% of the data are from the first year of plant operation. This figure is further improved in the INEL data base.

3. LOSS OF OFFSITE POWER (LOSP)

3.1 Data Sources

Loss-of-Offsite-Power (LOSP) experiential data have been reviewed in four studies since 1980:

1. Scholl[12] reviewed the data received from licensees following a June 1980 NRC request to submit licensee experience with LOSP events. This review includes a list of 109 occurrences of LOSP events.
2. A SAI study was summarized in EPRI-NP-2301[13] which used data collected from 47 nuclear power plant sites. The report presents frequency and duration of LOSP from 45 occurrences through April 1981, representing 375 plant years of experience.
3. A NSAC/ORNL study was reported in NSAC/80[14] which covered 52 nuclear power plant sites, for the period prior to December 1983. It summarizes 47 LOOP events in 530 plant years.
4. An USNRC study for the resolution of the "Station Blackout" issue was reported in NUREG-1032. The study covered 52 NPPs (all the US NPP sites of December 1983 excluding three with one offsite power connection). It summarizes 55 events in 533 plant years.

The review of the data sources has resulted in several findings:

a) The Scholl data base is rather conservative and needs additional evaluations prior to its utilization in PRAs.

b) The NP-2301 data base is more realistic. A few events are apparently missing from this source. Its recovery probability information is relatively conservative for use in PRAs.

c) The NSAC/80 data base appears to be suitable for realistic PRA analyses. It recommends exclusion of several total LOOP occurrences during shutdown which it judges to be "impossible" during operation. Our judgment is that these are inadvertent human errors that should be included for completeness.

The above three studies reported the LOSP events by plant and per geographical regions which have similar weather conditions and an interconnection agreement with respect to keeping a reliable electric supply in that region.

d) The NUREG-1032 data[15] are based on almost the same data base as the NSAC/80 however, it includes the "shutdown" events as well. The main improvement of this study is that it provides a breakdown of all the LOSP events into well defined causes which allows tailoring of the LOSP frequency of a new plant according to its design and also allows for evaluating the improvement that may be expected by a design change in an older plant.

3.2 Some Applications

Ref. 16 presents a review of the results of applying the first three data sources to a BWR. The application was based on using the LOSP events in the geographical region of the new BWR plant for generating a prior to serve as a first estimate to the plant LOSP frequency (One-Stage Bayesian). The NUREG-1032 study[15] presents several comparisons between PRAs results using plant or regional LOSP experience, and the results of the general methodology suggested taking into account plant specific features such as plant protection against weather conditions in the region, grid related design and plant specific design and procedures. It concludes that the generic models can usually provide good "ball park" results for generic applications and perspectives. It would be less suitable for plant-specific LOSP frequency determination at least until better data would be accumulated on the design features associated with each LOSP occurrence.

It may be concluded that while NUREG-1032 may be more suitable for the newer plants, NSAC/80 would be more suitable for two stage Bayesian updating of data for older plants.

4. LOCA FREQUENCIES

The Reactor Safety Study (RSS) evaluated the Loss of Coolant Accident (LOCA) frequencies by inference from generic data on pipe break in the non nuclear industries. This is the basis for the values shown in Table 4 for the RSS. The Reactor Pressure Vessel (RPV) rupture probability was also based on non nuclear vessel experience. While the frequency of the latter did not change much by newer PRAs (most of them still use the RSS value), the LOCA frequencies were reevaluated in the newer PRAs. Midland[17b],

Table 4: A Comparison of LOCA Frequencies in Various PRAs

LOCA Type Initiator(*)	Very Small LOCA	Small LOCA	Medium LOCA	Large LOCA	RPV Rupture	Interfacing LOCA	SG Tube Rupture
PRA	VS	S	M	A	RPV	Aout	R
ARKANSAS IREP	0.020	6.9E-4	1.6E-4	8.7E-5	--	--	--
MIDLAND PRA	5.0E-3	3.3E-3	4.7E-4	2.0E.4	--	7.7E-7	0.014
B/W Owner Group	8.3E-3	4.0E-4	--	--	--	--	0.017
OCONEE PRA	--	3.0E-3	--	9.3E-4	1.1E-6	1.4E-7	8.6E-3
LIMERICK PRA	--	0.010	2.0E-3	4.0E-4	--	--	--
SHOREHAM PRA	--	8.0E-3	3.0E-3	7.0E-4	3.0E-7	1.8E-7	--
BWR-6 PRA	--	1.2E-3	6.7E-4	2.1E-4	--	1.7E-7	--
SEABROOK PRA	--	1.7E-2	4.7-4	2.0E-4	2.7E-7	1.8E-6	1.4E-2
RSS-PWR	--	2.7E-3	8.1E-4	2.7E-4	1.0E-6	1.1E-5	--
RSS-BWR	CRD Pump	2.7E-3	8.1E-4	2.7E-4	1.0E-6	--	--

*VS:Very Small - less than 1.5 inch diameter break; S:Small LOCA - less than 4 inch diameter break; L:Large LOCA - Greater than 4 inch diameter if "Medium" is not considered, otherwise greater than 8 inch diameter size.

Oconee[9] and Seabrook[17a] PRAs used experiential data for the evaluation of part of the LOCA frequencies rather than pipe break data used in the RSS. The Oconee PRA[9] considered the following events in a population of 35 plants:

Large LOCA (A) : no event occurred

Small LOCA (S) : one event that occurred at Zion Unit 1 in 1975

SGRupture (R) : Three events of SG tube ruptures with leakage rates greater than 100 gpm : Surry Unit 2 (Nov. 1972), Point Beach Unit 1 (Feb. 1975), and Prairie Island Unit 1, (Oct. 1979).

A two-stage Bayesian analysis was applied to the above generic data and to the Oconee plant-specific experience which reflects none of the above events in any of the three units on site. The BNL review of the Oconee PRA[10] added another event:

Very Small LOCA (VS): One event that occurred at H.R. Robinson Unit 2 (May 1975).

This has added a frequency of 3×10^{-3} (see Table 4) in addition to the NP-2230 transient no. 6. B/W owner group[18] based their estimate of "VS" on the precursor study (NUREG/CR-2497)[19] which introduced the "Robinson event" mentioned above.

Table 4 summarizes the list of initiating events that can be considered under the "LOCA" category and their frequencies according to several PRAs. It includes both PWR and BWR plants.

The Table includes LOCA frequencies from two types of origins:

a) Based on pipe break frequencies such as in the RSS.

b) Based on LERs such as in some of the newer PRAs.

The frequencies of Interfacing LOCA (Aout) has been derived in a process similar to the small LOCA and SG rupture cases. The RSS estimated its frequency based on generic check valve failure rates. Later PRAs, used generic valve and pipe break data[20-23] to estimate Interfacing LOCA frequencies[9,17,24,25]. The publication by NRC[26] of a number of casees in which a testable check valve was inadvertently activated while another check valve was leaking, lead to several recent studies which uses LER for a PWR[24] and for BWRs[25,27] experience to improve the valve and pipe break data in order to provide a more plant-specific frequency determination. The NRC data reported nine failures of air operated check valves in 1361 BWR valve years. Tables 5 and 6 present a comparison of several sources on valve failure, rupture and leakage rates as used in several PRAs and other special probabilistic studies. While the RSS used data from non nuclear sources, the NUREG/CR-1363 uses LERs for many plants, most of them include small leakages as well as very small ones (Tech spec exceedance). The last two sources (see Tables 5 and 6) uses more recent LERs. The data provided in the tables is used to estimate the Interfacing LOCA frequencies according to the number of valves available in every leak path identified in the particular plant under review, and on the basis of the testing intervals used on each leak path's valves.

Table 5: Motor Operated Valves Failure Rates

Component	Source	Failure Mode	Assessed Range	Mean Value
Motor Operated Valves (MOV)	RSS[1]	Failure to operate (include command)	$3 \times 10^{-4} - 3 \times 10^{-3}$/d	1.3×10^{-3}/d
	NUREG/CR-1363[23] (for BWRs)	Failure to operate (include command)	---	8×10^{-3}/d
	NUREG/CR-1363 (for BWRs)	Failure to operate (w/o command)	---	6×10^{-3}/d
	Command Failure of both MOVs (Inboard and	Failure of Inboard and Outboard MOVs	---	2×10^{-3}/d
	SNPS-PRA[11] App. A.2*	MOV Spurious Opening	---	1.6×10^{-7}/hr
	CHU-1987[25]	MOV Spurious Opening	---	9.2×10^{-8}/hr
	CHU-1987	Failure to reclose	---	3.9×10^{-6}/d
	CHU-1987	Inadvertent opening	---	1.2×10^{-3}/d

*Based on GE evaluation

Table 6: Valve Rupture or Excessive Leakage Rates

Component	Source	Failure Mode	Assessed Range $[hr^{-1}]$	Mean Value $[hr^{-1}]$
Check Valves	RSS[1]	Internal Leakage (severe)	10^{-6} -10^{-7}	$3.8x10^{-7}$
	NUREG/CR -1363[23]	Internal Leakage (all sizes)	---	$1x10^{-6}$
	SEABROOK PRA update[17]	Internal Leakage (severe)	$5x10^{-8}$ -$1x10^{-10}$	10^{-9}
		(all sizes)	$5x10^{-6}$ -$5x10^{-8}$	$5x10^{-7}$
	CHU-1987[25]	Internal Leakage (all sizes)	---	$3.4x10^{-7}$
Check Valves or Motor Operated Valves	RSS	Rupture	10^{-7} -10^{-9}	$2.7x10^{-8}$
	NUREG/CR -1363	External Leakage/Rupture	---	$7x10^{-8}$
	SEABROOK PRA update	Rupture	---	$<5x10^{-9}$*
	CHU-1987	Disk Separation	---	$1.4x10^{-7}$

*Never occurred in more than 10,000 valve-years

5. SPECIAL INITIATING EVENTS

The more recent PRAs do not solely use the transient data sources for identifying and evaluating IEs. They include a treatment to identify events of the loss of support systems. Some IEs of this kind are loss of instrument air, DC power buses, service water or component cooling water. These events, in spite of being relatively infrequent, have a large impact because several support and subsequently frontline systems fail as a result of the IE.

As a result of their low frequency of occurrence, there is no sufficient plant specific experience. Thus, their potential frequency is estimated by a special analysis; in most cases a fault tree analysis. Such an analysis was performed in the Oconee PRA[9]. Table 7 is an example of the evaluation of the loss of instrument air initiator frequency performed in Reference 9 by the fault tree method.

Table 7: Loss of Instrument-Air-Initiator Frequency Contributors

Event	Description	Dominant Cut Set	Initiator Frequency (yr^{-1}) OPRA	BNL**
Contamination	Inadvertent IA system contamination with water or oil	AIAPICF	0.102	0.133
Pipe rupture	IA pipe rupture not repaired in 10 minutes*	AIAPILF* AIAPIL10F	0.052	0.059
Loss of SA and one IA train	Pipe leak in SA system and failure of one IA compressor to run	ASAPILF* AIACPCF*3	0.006	0.007
Loss of SA and one IA in maintenance	Pipe leak in SA system and one IA compressor in maintenance	ASAPILF* AIACPAM*3	0.003	0.004
Loss of SA and loss of RCW to IA	Pipe leak in SA system and RCW valve to IA fails closed	ASAPILF* ARCWIASVO*3	0.002	0.003
One IA train fails and SA interconnect and fails too	IA fails mechanically to run, and SA interconnect fails	AIACPCF*3* (ASAIAVDO + ASAIAVVH)	~0	0.001
Total			0.17	0.21

*The ability to repair or isolate a major leak in the IA system is complicated by the fact that the system was not included in the detailed design drawings -- which make the recovery operation more difficult. Some pipes and valves are not visible or accessible (OPRA, page A.15-10).

**Differences in the BNL reevaluation are due to BNL's use of a factor of 0.8, rather than 0.7, to account for unit 3 being at power during the fault, and to correctly use the failure data given on page A15-19 of OPRA.

The data required for these special IEs are of two kinds:

a) Experiential data (LERs) on these special events in similar plants. This is used to identify the existance of the potential for these kinds of IEs.

b) Failure rates of equipment either generic or plant-specific for the quantification of the fault tree that was constructed for estimating their potential frequency.

6. INTERNAL FLOODING

Internal flooding is another initiating event that is treated in a detailed special study. The two common approaches that were referred to in the previous sections were also used for evaluating the internal flooding frequencies:

a) Estimation from NPP experienced flooding events combined with the plant-specific flooding-event experience.

b) Estimation from detailed fault-tree models for several internal flooding rates based on plant design and on the basis of pipe rupture rates, expansion joints rupture rates, valves' failure rates and human errors in maintenance.

Three important internal flooding studies were performed in the past(9,11,17). These three PRAs all used a combination of both approaches. In the case of the Oconee PRA(9,28) three iterations were carried out on the flooding analysis. The first was according to approach (a) above, the second was a combination of (a) and (b) and the third iteration was a combination of (a) and (b) with very detailed (b) approach. In the case of the Seabrook PRA(17), approach (a) was used more extensively than approach (b). However, pipe failure rate data was used to assess the frequency of the very large flooding category for which no event has been experienced by the industry. The approach in the Shoreham PRA(11) took into account the NP-801 transient IE frequencies in combination with unavailability of isolation system (due to maintenance and human errors) when the transient occurred, leading to a flooding event.

All the above application indicated a need for flooding event records from NPP operating experience, and for effective pipe and valve failure rates.

7. SUMMARY

A discussion of initiating event selection and frequency determination was presented by a breakdown into transients, Loss of Offsite Power, LOCAs, Special events (mainly infrequent support system failures) and internal flooding. The methodologies used were examplified from the treatment employed in recently published PRAs. There are two main approaches used:

a) Study of LERs and plant-specific occurrences which are combined by the two-stage Bayesian methodology.

b) Fault tree analyses and similar studies to probablistically evaluate the frequency of a certain initiating event.

Many studies combined the two approaches in their detailed analysis.

Our review reveals the need for accumulating LERs for use in PRA for the purpose of initiating event selection and frequency determination. It identifies the need for a master list of initiating event categories with well defined event descriptions to help the PRA analyst. In addition, the need for better failure rates in some particular areas - mainly for pipes and valves is identified for use in determination of LOCAs and internal flooding events.

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